Contents

Notice of the Minister of Science and Technology (I)

Nuclear Facility

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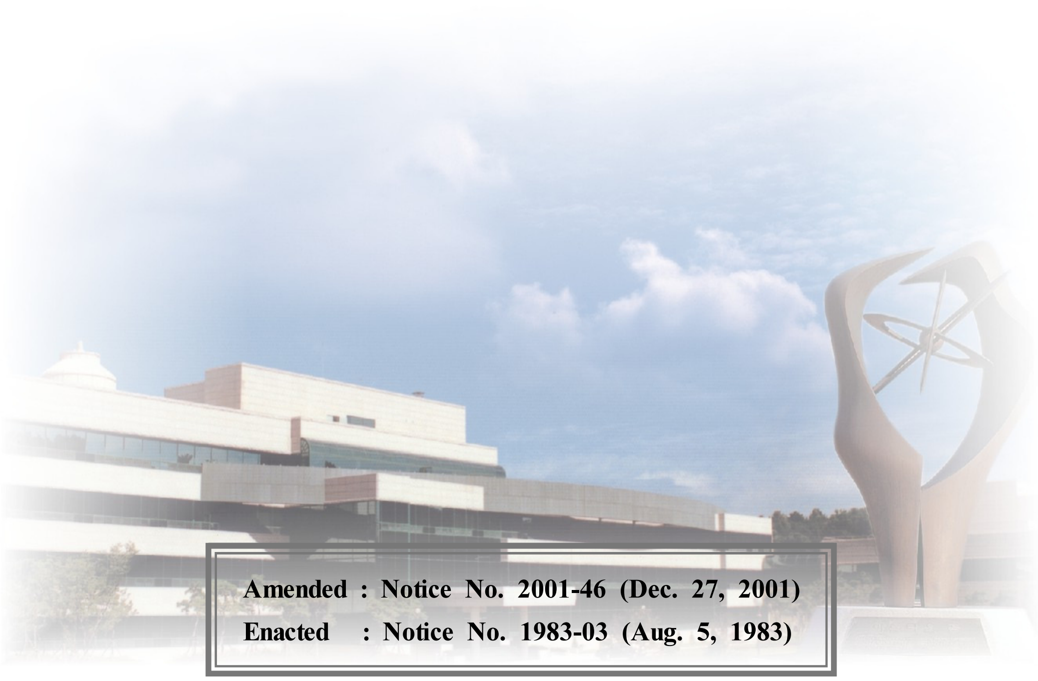
Notice of the Minister of Science and Technology No.2001-46 (MOST.reactor.003)

【 01 】

Standard Format and Content of Technical Specifications for Operation



Amended : Notice No. 2001-46 (Dec. 27, 2001)
Enacted : Notice No. 1983-03 (Aug. 5, 1983)

◎ **Notice of the Minister of Science and Technology No.2001-46 (MOST.reactor.003)**

The Standard Format and Content of Technical Specifications for Operation as provided for in Article 16 (1) of the Enforcement Regulation of the Atomic Energy Act is hereby notified publicly as follows:

December 27, 2001
Minister of Science and Technology

Standard Format and Content of Technical Specifications for Operation

Chapter 1 General Provisions

Article 1 (Purpose) The purpose of this notice is to prescribe detailed matters and guidelines for preparing the technical specifications for operation as provided for in Article 21 of the Atomic Energy Act and Article 16 (1) of the Enforcement Regulation of the Act.

Article 2 (Scope of Preparation) The technical specifications for operation shall be composed of and prepared in the areas of the operation of the nuclear reactor facility, the radiation and environment control of the nuclear reactor facility, and the operational management of the nuclear reactor facility.

Chapter 2 Operation of the Nuclear Reactor Facility

Article 3 (Use and Application) Matters of each of the following items shall be described in the technical specifications for operation:

1. Definitions of the terms;
2. Meanings of the logical connectors used to identify the relationships among unsatisfied conditions, required actions, completion times, and surveillance requirements and frequencies;
3. Guidelines for establishing the prescriptions and the use of completion times; and
4. Proper establishment and application methods for the surveillance frequencies.

Article 4 (Safety Limits) (1) The safety limits for the main process variables required to ensure the integrity of the physical barriers against the release of the radioactive materials shall be prescribed in the technical specifications for operation.

(2) In case the safety limit of the reactor exceeds, the operator shall shutdown the nuclear reactor and report to the regulatory authorities the details, the cause analysis of the event, and the actions taken. The reactor shall be put into operation after the approval of the regulatory authorities.

(3) In case the automatic protection system does not perform the required function, the operator shall take proper actions and then report to the regulatory authorities the details, the cause analysis of the event, and the actions taken.

Article 5 (Limiting Conditions for Operation) (1) The limiting conditions for operation which are the minimum required functions or performances to maintain the nuclear reactor facilities in a safe state shall be prescribed in the technical specifications for operation. In case the limiting conditions are not met, either the reactor shall be shut down or the required actions shall be taken until the conditions are met.

(2) The selection criteria of the limiting conditions for operation prescribed in the technical specifications for operation shall be as follows:

1. Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
2. Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
3. Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
4. Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Article 6 (Surveillance Requirements) The technical specifications for operation shall prescribe the surveillance requirements concerning tests, corrections, and inspections, which are performed to verify that acceptable level of qualities of systems and equipments are maintained, that the systems and equipments are operated within the safety limits, and that the limiting conditions for operation are abided by, and they shall prescribe the surveillance frequency reflecting operating experience or a probabilistic safety analysis, etc.

Article 7 (Design Characteristics) The technical specifications for operation shall prescribe the important design safety characteristics such as layout which is not included in the safety limits, limiting conditions for operation, or surveillance requirements, but causes a significant effect on the safety if it is changed or modified.

Chapter 3 Radiation and Environment Control of the Nuclear Reactor Facility

Article 8 (Radiation Protection) The following matters shall be prescribed in the technical specifications for operation to satisfy the requirements under the Article 96 of the Atomic Energy Act, Articles 300 and 301 of the Enforcement Decree of the Atomic Energy Act, and the Standards for Radiation Protection, etc. (Notice of the MOST No.2002-23, January 6, 2003):

1. Integrity of the nuclear reactor facility;
2. Radiation safety control; and
3. Management of radiation measuring equipment.

Article 9 (Management of Radioactive Materials, etc.) The following matters shall be prescribed in the technical specifications for operation to satisfy the requirements under the Standards for Radiation Protection, etc. (Notice of the MOST No.2002-23, January 6, 2003):

1. Management of radioactive waste;
2. Exhaust and drainage monitoring systems;
3. Matters on receipt and disbursement, transport, storage, and handling of nuclear fuel materials; and
4. Handling of radioisotope, etc.

Article 10 (Environment Preservation from the Nuclear Reactor Facilities) The matters on environment monitoring shall be prescribed in the technical specifications for operation to satisfy the requirements under the Regulation on Survey of Radiation Environment and Assessment of Radiological Impact on the Environment in the

Vicinity of Nuclear Power Utilization Facilities (Notice of the MOST No.2004-17, July 13, 2004).

Chapter 4 Operational Management of Nuclear Reactor Facility

Article 11 (Organization and Function) Operating organization, administrative organization, duty of each department, staffing, responsibility of operational management, KHNP Nuclear Review Board (KNRB), Plant Nuclear Safety Committee (PNSC), qualification for plant staff and procedures, etc. therefor shall be prescribed in the technical specifications for operation.

Article 12 (Surveillance of Nuclear Reactor Facilities) Precautions on surveillance, positioning of the patrols, and the items and actions for surveillance of equipments shall be prescribed in the technical specifications for operation.

Article 13 (Operator Action Items for Emergency) The matters related with the actions upon occurrence of an emergency, the actions to be taken after the reactor is tripped, the manual reactor trip, and the manual actuation of Emergency Core Cooling System (ECCS) shall be prescribed in the technical specifications for operation.

Article 14 (Programs and Manuals) Programs and manuals necessary for securing safety of the nuclear power plant shall be prescribed as follows:

1. Program of the technical bases control for the technical specifications for operation;
2. Safety function determination program;
3. Program of reactor coolant release sources outside containment;
4. In-service inspection (ISI) program;
5. In-service test (IST) program;
6. Fire protection program;
7. Testing program of diesel fuel oil;
8. Program of component cyclic and transient limits;
9. Offsite dose calculation manual (ODCM);
10. Atmospheric monitoring instrumentation management procedure;
11. Guideline for post-accident sampling analysis;
12. Radiation monitoring instrumentation management program;

13. Radioactive effluents control program;
14. Radioactive waste process control program (PCP);
15. Secondary water chemistry control program;
16. Ventilation filter testing program (VFTP);
17. Radioactivity monitoring program for explosive gas and liquid storage tank; and
18. Mobile testing program of hydrogen recombiner.

Article 15 (Reporting Requirements) The following reports necessary for securing safety of the nuclear power plant shall be prescribed in the technical specifications for operation:

1. Biannual and annual environmental radiation surveillance report;
2. Quarterly operation report;
3. Quarterly radiation control report;
4. In-service inspection report;
5. Accidents and incidents report; and
6. Reload safety evaluation report.

Addenda

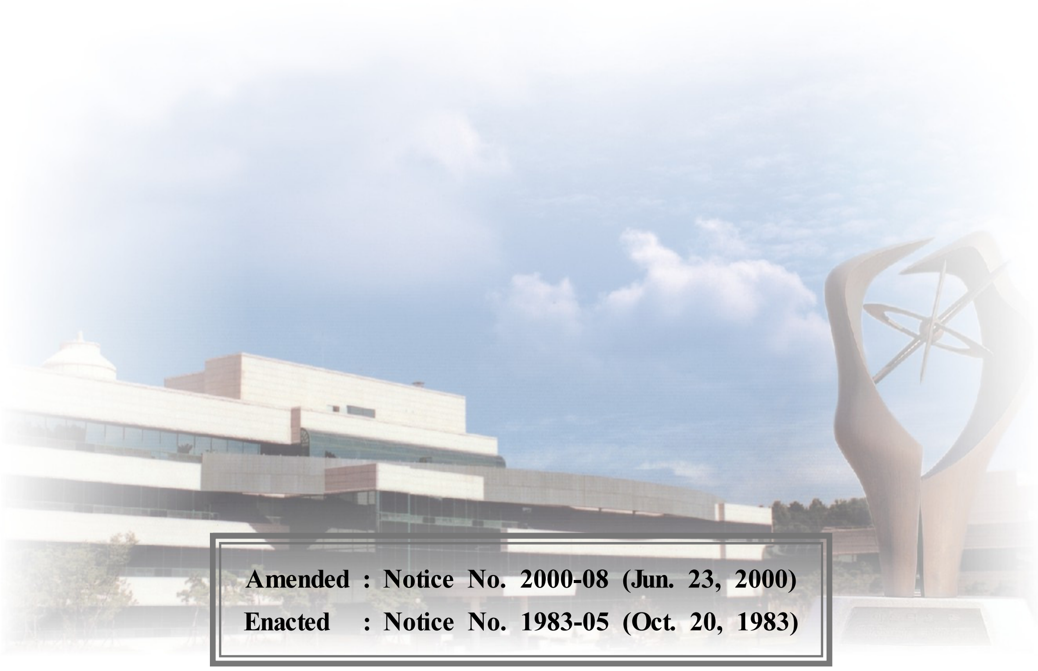
Article 1 (Enforcement Date) This notice shall enter into force on the date of its promulgation.

Article 2 (Repeal of Notice) Notice of the MOST No.1983-03 "The Standards for Preparing the Technical Specifications for Operation" shall be repealed at the time this notice enters into force.

Article 3 (Transitional Measures) The technical specifications for operation as provided for in Article 21 of the Atomic Energy Act approved before the enforcement date of this notice shall be regarded as valid.

【 02 】

Technical Standards for Locations, Structures and Equipment of Nuclear Reactor Facilities



Amended : Notice No. 2000-08 (Jun. 23, 2000)
Enacted : Notice No. 1983-05 (Oct. 20, 1983)

◎ **Notice of the Minister of Science and Technology No.2000-08 (MOST.reactor.004)**

The Technical Standards for the Locations, Structures, and Equipment of Nuclear Reactor Facilities as provided for in Articles 4 through 9 and 22 of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc. are hereby notified publicly as follows:

June 23, 2000
Minister of Science and Technology

Technical Standards for Locations, Structures, and Equipment of Nuclear Reactor Facilities

Article 1 (Purpose) The purpose of this notice is to prescribe details on "The Technical Standards for the Locations of Nuclear Reactor Facilities" as provided for in Articles 4 through 9 of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc. (hereinafter referred to as the “Regulations”), and "The Technical Standards for the Structures, and Equipment of Nuclear Reactor Facilities" as provided for in Article 22 of the Regulations.

Article 2 (Scope of Application) The standards prescribed in Table 1 shall apply to the technical standards for the locations of nuclear reactor facilities, and those prescribed in Table 2 shall apply to the technical standards for the structures and equipment of nuclear reactor facilities. Provided, that as regards the application of the standards prescribed in Table 2, in case the foreign country supplying the nuclear reactor facilities or technology has its own technical standards, if it is proven that neither safety nor performance of the nuclear reactor facilities is degraded by application thereof, such foreign standards may apply to the nuclear reactor facilities concerned upon approval of the Minister of Science and Technology.

Addenda

Article 1 (Enforcement Date) This notice shall enter into force on the date of its promulgation.

Article 2 (Repeal of Notice) Notice of the MOST No.83-05 (October 20, 1983) shall be repealed at the time this notice enters into force.

[Table 1]

Technical Standards for Locations of Nuclear Reactor Facilities

No.	Notice Subjects	Related Articles	Applicable Foreign Regulations
1	Guidelines for investigating and evaluating seismic and geologic characteristics of nuclear reactor facilities site	Regulations Article 4	1) 10CFR Part 100 Appendix A: "Seismic and Geologic Siting Criteria for Nuclear Power Plants" 2) R.G. 1.60: "Design Response Spectra for Seismic Design of Nuclear Power Plants" 3) R.G. 1.132: "Site Investigations for Foundations of Nuclear Power Plants" 4) R.G. 1.138: "Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants"
2	Guidelines for location restrictions of nuclear reactor facilities	Regulations Article 5	1) 10CFR 100.11: "Determination of Exclusion Area, Low Population Zone and Population Center Distance"
3	Guidelines for atmospheric conditions of the site and the surrounding area	Regulations Article 6	1) R.G. 1.4: "Assumption used for Evaluation the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors 2) R.G. 1.145: "Atmospheric Dispersion Models for Potential Accident Consequences at Nuclear Power Plants"
4	Guidelines for investigating and evaluating hydrologic characteristics of the site and the surrounding area	Regulations Article 7	1) R.G. 1.59: "Design Basis Floods for Nuclear Power Plants" 2) R.G. 1.102: "Flood Protection for Nuclear Power Plants" 3) R.G. 1.113: "Estimating Aquatic Dispersion Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I" 4) R.G. 1.132: "Site Investigations for Foundations of Nuclear Power Plants" 5) R.G. 1.135: "Normal Water Level and Discharge at Nuclear Power Plants"

No.	Notice Subjects	Related Articles	Applicable Foreign Regulations
5	Guidelines for investigating and evaluating man-made incidents for site selection	Regulations Article 8	1) R.G. 1.78: "Assumptions for Evaluating the Habitability of Nuclear Power Plant Control Room during a Postulated Hazardous Chemical Release" 2) R.G. 1.91: "Evaluations of Explosions Postulated to Occur on Transportation Routes near Nuclear Power Plants" 3) R.G. 1.95:"Protection of Nuclear Power Plant Control Room Operators against an Accidental Chlorine Release"
6	Guidelines for investigating and evaluating natural phenomena of the site and the surrounding area	Regulations Article 9	1) R.G. 1.59: "Design Basis Floods for Nuclear Power Plants" 2) R.G. 1.76: "Design Basis Tornado for Nuclear Power Plants" 3) IAEA 50-SG-S11A: "Extreme Meteorological Events in Nuclear Power Plant Siting" 4) IAEA 50-SG-S1: "Earthquakes and Associated Topics in Relation to Nuclear Power Plant Siting"

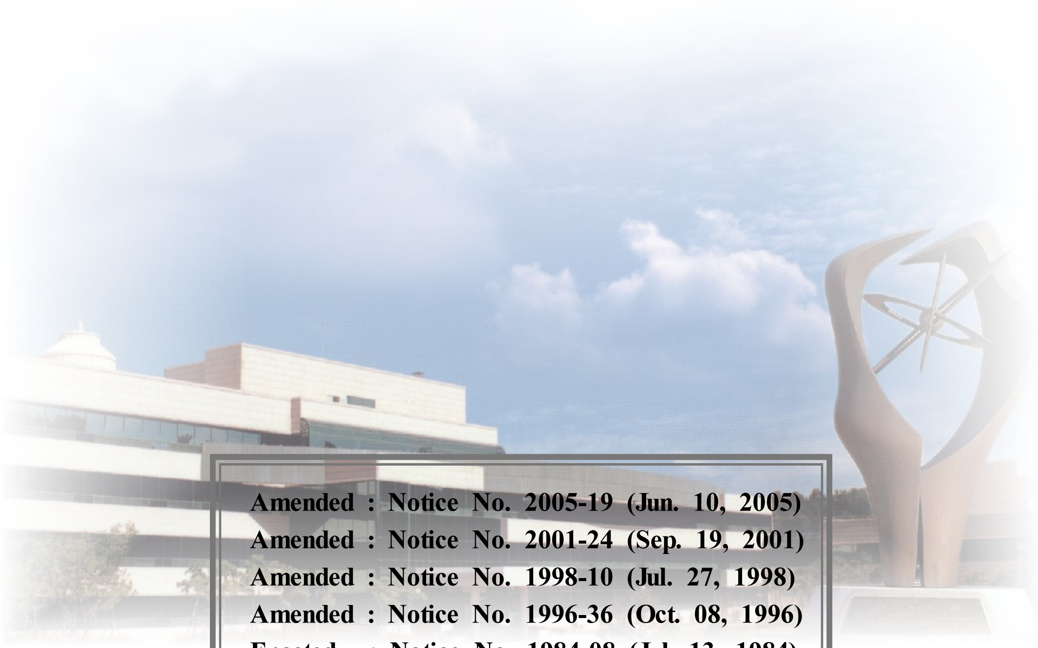
[Table 2]

Technical Standards for Structures and Equipment of Nuclear Reactor Facilities

No.	Notice Subjects	Related Articles	Applicable Foreign Regulations
1	Guidelines for safety and relief valves installed at the nuclear reactor facilities	Regulations Article 22	1) ANSI/ANS 51.1: "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants" 2) ASME Boiler and Pressure Vessel Code, Section III - Subsection NCA-General Requirements for Division 1 and 2 - Subsection NB-Class 1 Components - Subsection NC-Class 2 Components - Subsection ND-Class 3 Components

【 03 】

Standard Format and Content of Radiation Environmental Report for Nuclear Power Utilization Facilities



Amended : Notice No. 2005-19 (Jun. 10, 2005)
Amended : Notice No. 2001-24 (Sep. 19, 2001)
Amended : Notice No. 1998-10 (Jul. 27, 1998)
Amended : Notice No. 1996-36 (Oct. 08, 1996)
Enacted : Notice No. 1984-08 (Jul. 13, 1984)

◎ **Notice of the Minister of Science and Technology No.2005-19 (MOST.reactor.006)**

The Standard Format and Content of Radiation Environmental Report for Nuclear Power Utilization Facilities as provided for in Articles 7 (1), 28, 36, 40, 45, 79 (4), and 129 (2) of the Enforcement Regulation of the Atomic Energy Act is hereby revised and notified publicly as follows:

June 10, 2005
Minister of Science and Technology

Standard Format and Content of Radiation Environmental Report for Nuclear Power Utilization Facilities

Article 1 (Purpose) The purpose of this notice is to provide for entries, preparation methods, and other matters required for radiation environmental report (hereinafter referred to as "report") for the purpose of assessing the impacts of radiation or radioactivity on the neighboring environment as a result of construction and operation of nuclear power utilization facilities (hereinafter referred to as "environmental impacts") and for draft radiation environmental report (hereinafter referred to as "draft report") for the purpose of gathering residents' opinions.

Article 2 (Scope of Application) This notice shall apply to each of nuclear power utilization facilities as provided for in the following paragraphs according to the classification of draft report and report.

1. Draft report: nuclear power reactor and related facilities as provided for in Article 11 (1) of the Atomic Energy Act and radioactive waste disposal facilities and interim storage facilities of spent nuclear fuel among disposal facilities, etc. as provided for in Article 76 (1) of the Atomic Energy Act
2. Report: nuclear power reactor and related facilities, research reactor not less than 100 kW of thermal output, etc., nuclear fuel cycling facilities, disposal facilities of radioactive wastes

Article 3 (Definitions) (1) Definitions of the terms used in this notice shall be as follows:

1. The term "residents" means the inhabitants whose opinions have to be gathered in accordance with Article 104-5 (1) of the Atomic Energy Act and the inhabitants within emergency planning zone of the utilization facilities concerned, and the inhabitants of town/sub-county/village including the boundary of emergency planning zone; and
2. The term "enterpriser" means a person who submits the report to the Minister of Science and Technology with an attachment to the application for construction permit and/or operating license of nuclear power utilization facilities as prescribed in Article 2 of the Atomic Energy Act.

(2) Terms used herein other than those set forth in the foregoing Paragraph 1 shall have the same meanings as provided for in the Atomic Energy Act, the Enforcement Decree thereof, the Enforcement Regulation thereof, and related Ordinances and Notices of the Minister of Science of Technology.

Article 4 (General Requirements for Preparation of Report, etc.) Each of the following general requirements shall be considered in preparation of draft report and report (hereinafter referred to as "report, etc."):

1. Report, etc. shall be objectively and logically formulated based on scientific facts. For that purpose, natural science, social science, and applied science, etc. shall be comprehensively used;
2. Methods and techniques used in investigation/analysis of data, and prediction/assessment of environmental impacts shall be objectively acknowledged to be appropriate. Provided, that the existing cases or data may be used in case where the objective assessment methods are not established;
3. Environmental status shall be basically investigated by the enterpriser at the site. However, the existing literature or data of institution with public confidence, etc. may be quoted in case of necessity. Provided, that the source or basis of the quoted data shall be clearly presented;
4. Among the detailed items of preparation as provided for in this notice, in regard of excluded matters which are deemed to be difficult or unnecessary to be applied due to the installation location of nuclear power utilization facilities or characteristics of design, the reasons or appropriateness thereof shall be specified; and
5. In regard of technical terms used in report, etc., explanation of terms shall be formulated in a supplement for the purpose of understanding of the general public.

Article 5 (Composition and Preparation Guidelines of Report, etc.) (1) The report, etc. shall include each of the followings. Draft report and report shall be formulated based on Table 1 "Guidelines for preparation of draft radiation environmental report of nuclear power utilization facilities", and Table 2 "Guidelines for preparation of radiation environmental report of nuclear power utilization facilities", respectively.

1. Overview of construction plan

   The need for construction of nuclear power utilization facilities, construction plan, and reasons for the relevant site selection, etc. shall be described.

2. Status of environment

   Environmental characteristics of site and adjoining area thereof shall be described in order to predict/assess the environmental impacts due to construction and operation of nuclear power utilization facilities, and the data of investigation and measurement in the latest more than one year shall be used.

3. Status of facilities

   Facilities of nuclear power utilization facilities shall be described with focussing on radiation related systems and facilities.

4. Environmental impacts due to construction

   When nuclear power utilization facilities are constructed in the same site where the existing nuclear power utilization facilities are operated, the prediction/assessment of impacts of the existing nuclear power utilization facilities on construction workers during a construction period of relevant facilities thereof shall be described.

5. Environmental impacts due to operation

   Environmental impacts due to operation of nuclear power utilization facilities shall be described by prediction/assessment using quantitative methods as much as possible, and in the case where quantification is not easy, the objective and qualitative methods shall be used for prediction and assessment.

6. Impacts due to accidents

   Potential environmental impacts due to accidents of nuclear power utilization facilities shall be predicted and assessed, and the report shall describe matters including public protective measures.

7. Environmental monitoring plan

   In order to verify the environmental impacts and extents thereof due to operation of nuclear power utilization facilities, environmental monitoring plan shall be formulated and described according to the classification of pre-operation and in-operation of nuclear power utilization facilities.

8. Consensus of residents' opinions

Opinions of the residents and the heads of the related administrative agencies, etc. for draft report, and analysis and assessment of the result of holding a public hearing, and their specific and reflected details shall be described in the report.

9. Comprehensive assessment

Expected environmental impacts shall be comprehensively reviewed and final assessment for the justification of the relevant business shall be described.

10. Others

Matters that can be referred or related to perusal of draft report and review of report shall be described.

(2) When draft report is formulated, insignificant matters from a point of view of a consensus of residents' opinion may be briefly described or omitted.

(3) Contents of report shall not be contrary to those of draft report, and in those cases where there exists any change of contents, reasons thereof shall be presented.

(4) Report of research reactor and nuclear fuel cycle facilities shall be formulated in accordance with that of nuclear power plant in Table 2 "Guidelines for preparation of radiation environmental report of nuclear power utilization facilities", and the relevant items may be omitted in those cases where the application is difficult or unnecessary due to the purpose of use of facilities or fundamental difference of design among description items. Provided, that reasons and appropriateness thereof shall be presented in those cases.

Article 6 (Submission of Draft Report) Enterpriser shall promptly comply in those cases where the city mayors/county chiefs/district chiefs of competent jurisdiction verify whether the draft report is appropriately submitted to the heads of the administrative agencies as provided for in each Subparagraph of Article 130 (1) of the Enforcement Regulation of the Atomic Energy Act, and they take measures that enterpriser shall submit an additional draft report to an additional administrative agency.

Article 7 (Supplement of Report, etc.) Enterpriser shall take relevant supplementary measures in those cases where after the city mayors/county chiefs/district chiefs of competent jurisdiction review the draft report in accordance with Table 1, they acknowledge contents thereof not to be appropriate for a consensus of residents' opinions and request a supplement for draft report.

Article 8 (Impacts due to Operation of Complex Facilities) In those cases where numerous nuclear power utilization facilities are constructed and operated on the same site, assessment shall be done including impacts due to the existing nuclear power utilization facilities.

Article 9 (Provisions to be applied *Mutatis Mutandis*) Relevant domestic regulations shall apply mutatis mutandis to those matters not provided in this notice.

Addenda

Article 1 (Enforcement Date) This notice shall enter into force on the date of its promulgation.

Article 2 (Repeal Notice) Notice of the Minister of Science and Technology No. 2001-24 (September 19, 2001) "Regulation on Preparation of Radiation Environmental Report for Nuclear Power Utilization Facilities, etc." is repealed at the time this notice enters into force.

[Table 1]

Guidelines for preparation of draft radiation environmental report of nuclear power utilization facilities

Items	Description guidelines
1. Overview of construction plan	
1.1 Need for construction	o Background, purpose, and need, etc. of construction of nuclear power utilization facilities shall be described.
1.2 Basis of execution of environmental impacts assessment	o Related laws and based regulations of execution of environmental impacts assessment shall be described.
1.3 Circumstances of promotion of business	o Circumstances of promotion by related laws until the time of submission of draft report shall be described as schedule of promotion and major details according to each procedure. o Approval/permit procedures and future plan of promotion, etc. by related laws from the completion of submission of report to the stage of final administrative control shall be shortly described.
1.4 Construction plan	o General construction plan of nuclear power utilization facilities of name, location of the place, licensee, source of facilities, existing nuclear power utilization facilities, major facilities, and details of construction, etc. shall be roughly described.
1.5 Reasons for the relevant site selection	o Process and the reasons why the relevant site is selected as the site of nuclear power utilization facilities shall be described.
2. Status of environment	
2.1 Status of site	o Location of site, land estate, boundary of facilities, and exclusion area boundary, etc. shall be described and necessary drawings shall be attached.
2.2 Land-use	o Status of land-use, status of production of agricultural/livestock products, each used area, and future planned area within emergency planning zone shall be described, and necessary drawings shall be attached.
2.3 Ocean-use	o Status of ocean-use, status of production of marine products in major fisheries and nurseries, water activities, and future plan, etc. within the distance of emergency planning zone shall be described, and necessary drawings shall be attached.
2.4 Meteorology and atmospheric dispersion	o Regional climate, site meteorological observation plan and results of observation, and atmospheric dispersion/diffusion of radioactive materials within emergency planning zone in the event of normal operation and accidents shall be assessed and described, and necessary drawings shall be attached.
2.5 Hydrology and hydrological dispersion	o Status of surface water and ground water, hydrological dispersion of radioactive materials within emergency planning zone shall be assessed and described, and necessary drawings shall be attached.
2.6 Status of ocean and oceanic dispersion	o Status of ocean necessary for assessment of oceanic dispersion of radioactive materials and oceanic dispersion of radioactive materials within emergency planning zone shall be assessed and described, and necessary drawings shall be attached.

Items	Description guidelines
2.7 Population	o Status of population distribution within emergency planning zone, moving population, and future estimated population, etc. shall be described, and necessary drawings shall be attached.
2.8 Status of environmental radiation/radioactivity	o Background level of environmental radiation/radioactivity within emergency planning zone shall be investigated and described, and necessary drawings shall be attached.
3. Status of facilities	o The external appearance of facilities, radioactive waste treatment system, and other major facilities related with radiation shall be roughly described, and the related drawings shall be attached. o Radiation source due to leakage from the radiation related system shall be assessed and described.
4. Impacts due to construction	o In those cases where there are construction activities around the operating nuclear power utilization facilities, radiation source and exposure dose to construction worker caused by this from nuclear power utilization facilities thereof shall be assessed and described. o Exposure doses are calculated as maximum individual exposure dose and person-rem exposure. o Computational models and methods used in calculation of exposure dose shall be described.
5. Impacts due to operation	
5.1 Exposure pathways	o Direct/indirect exposure pathways for human and major organisms shall be described and diagram of exposure pathways shall be presented.
5.2 Assessment of exposure dose	
5.2.1 Exposure from gaseous pathways	o Annual maximum exposure dose to the whole body, thyroid, and other major organs of individual through gaseous pathways within emergency planning zone including exclusion area boundary shall be calculated and described. o Annual whole body exposure dose for the resident group within emergency planning zone shall be calculated and described. o Models and input data used in calculation shall be described.
5.2.2 Exposure from liquid pathways	o The same method with exposure from gaseous pathways shall be described.
5.2.3 Items for direct exposure from nuclear power related facilities	o Annual maximum external exposure dose to individual due to direct exposure within emergency planning zone including exclusion area boundary shall be calculated and described. o Models and input data used in calculation shall be described.
5.2.4 Summary of exposure dose	o Annual exposure dose to group within emergency planning zone from the facilities-related every member subject to exposure during the operation of nuclear power utilization facilities shall be summarized and diagram shall be used if necessary. o Calculated maximum individual exposure dose shall be compared with the related criteria and assessed. o In those cases where numerous nuclear power utilization facilities including the relevant nuclear power utilization facilities are constructed and operated within the same site, exposure dose from the whole of nuclear power utilization facilities thereof shall be calculated and compared with the related criteria and assessed.

Items	Description guidelines
6. Impacts due to accidents	
6.1 Assumption of accidents	o Potential accidents during the operation of nuclear power utilization facilities shall be assumed to be classified by types o Occurrence probability of postulated accidents by types shall be assessed. o Severe accidents shall be excluded from the subject of assessment.
6.2 Radiation source	o Radiation source of postulated accidents by types shall be described. o Assumptions used in the determination of radiation source shall be described.
6.3 Methods of assessment	o Computational models and input data used in calculation of exposure dose shall be described.
6.4 Assessment of exposure dose	o Maximum individual exposure dose to individual within the appropriate distance of emergency planning zone including exclusion area boundary shall be calculated and described by types of postulated accidents. o Person-rem whole body exposure dose within emergency planning zone shall be calculated and described.
7. Environmental monitoring plan	o Environmental monitoring plan prior to operation shall be described based on environmental status data o Environmental monitoring plan during operation for environmental matters subject to be impacted by operation shall be described based on environmental monitoring plan prior to operation.
8. Comprehensive assessment	o Expected environmental impacts and measures shall be comprehensively reviewed and final conclusive assessment for justification of the relevant business shall be described.
9. Others	o Following items that can be referenced to perusal of draft report shall be described. - Participant agencies and participants for preparation of draft report - Time required for preparation of draft report - Other references, etc.
10. References	o References used in preparation of draft report shall be mentioned.
Appendix 1 : Summary	o Business plan and results of environmental impacts assessment, etc. shall be comprehensively summarized.
Appendix 2 : Explanation of terms	o In regard of technical terms used in draft, explanation of terms thereof shall be formulated for the purpose of understanding of the public.

[Table 2]

Guidelines for preparation of radiation environmental report of nuclear power utilization facilities

1. Guidelines for preparation of Radiation Environmental Report of Nuclear Power Plant

Items	Description items	Description guidelines
1. Overview of construction plan		
1.1 Need for construction		o Background, purpose, and need, etc. of construction of plant shall be described.
1.2 Basis of execution of environmental impacts assessment		o Related laws and based regulations of execution of environmental impacts assessment shall be described.
1.3 Circumstances of promotion of business		o Circumstances of promotion by related laws until the time of submission of report shall be described as schedule of promotion and major details according to each procedure. o Approval/permit procedures and future plan of promotion, etc. by related laws from the time of submission of report to the stage of final administrative control shall be shortly described.
1.4 Construction plan	1.4.1 Name	o General construction plan of plant shall be roughly described.
	1.4.2 Location of the place	o Location of site and administrative district shall be described and their specified charts of administrative district shall be attached. o Boundary of land estate, exclusion area boundary, and area of site shall be displayed as drawings and mutual relationship shall be described.
	1.4.3 Licensee	
	1.4.4 Source of plant	o The outline of sources of plant under scheduled construction, such as reactor type, type of used fuel and annual amount used, maximum thermal power and maximum electric power, and annual possible amount generated, etc., shall be described.
	1.4.5 Reserve rate	o Reserve rate by long-term demand-supply program of electric power shall be described.
	1.4.6 Existing nuclear power related facilities	o In those cases where there are existing plant or nuclear power related facilities within site, their sources shall be shortly described.
	1.4.7 Major facilities	o Type and capacity of major facilities of plant shall be described.
	1.4.8 Details of construction	o Place of construction, methods of construction, and scale, etc. shall be described. o Period of construction, expense, input of man power, expected operation life, etc. shall be described.
1.5 Reasons for the relevant site selection		o Process and the reasons why the relevant site is selected as the site of nuclear power utilization facilities shall be described.

Items	Description items	Description guidelines
2. Status of environment		
2.1 Status of site	2.1.1 Location of site	o Location of site, important geographical/natural features, and administrative districts shall be described. o Blank map and topographical map shall be attached.
	2.1.2 Site area	o Drawing necessary for explanation shall be attached. o Mutual relationship among boundary of land estate, boundary of plant, and exclusion area boundary, etc. shall be described in detail. o Occupancy area by types of major facilities in site shall be described.
2.2 Land-use	2.2.1 Status of land-use	o The present extent of land-use within a 10-km radius of site by types of use and land category, and future plan of land-use shall be described and special aspect of change at present shall be described. o Map of land-use which shows exclusion area boundary, boundary of plant, boundary of land estate, and status of land-use shall be attached.
	2.2.2 Production of livestock products	o Period of grazing and types of feed stuff, etc. for livestock within a 80-kim radius of site shall be described, and kinds and number of livestock and location of stock farm and poultry farm, etc. within a 10-kim radius shall be described. o Concentric circles of 2, 4, 6, 8, 10, and 20-km radius shall be divided into 16 directions, respectively, and annual production of meat and milk by types shall be described. o Concentric circles with intervals of 15-km between 20-km and 80-km radius shall be divided as provided for in the foregoing paragraph and their production shall be described. o Present used area of livestock/dairy farming and future planned area shall be described.
	2.2.3 Production of agricultural products	o Distribution and area by types of vegetable gardens within a 10-km radius of site shall be described. o Production of agricultural products by distance and types shall be described according to description form of livestock products. o Present used area of farm land and future planned area shall be described.
2.3 Ocean-use	2.3.1 Status of ocean-use	o Status of distribution and plan of use of fishing ports, fisheries, and nurseries, etc. within a 80-km radius of site shall be described, and the latest special aspects of change shall be described. o Status of distribution of leisure facilities such as swimming beach, etc. within a 80-km radius of site shall be described. o Drawing necessary for explanation shall be attached.

Items	Description items	Description guidelines
	2.3.2 Production of marine products	o Production by types of marine products produced in major fisheries and nurseries, etc. within a 80-km radius of site shall be described. o Production of marine products shall be divided into self-supporting use and commercial use and described accordingly.
	2.3.3 Water activities	o Activities of fisheries in the sea within a 80-km radius of site, activities of leisure, etc. shall be described. o Annual time of water activities by age and activities shall be described. o The total man-days participating in water activities shall be described.
2.4 Meteorology and atmospheric dispersion	2.4.1 Regional climate	o Climate around the site shall be roughly described. o Following local data of weather which are observed at a regional meteorological station around the site during the past more than 30 years shall be investigated and presented after analysis and arrangement. - Temperature, rainfall, humidity (average by month, the highest and lowest temperature of the year and month) - Direction and velocity of wind (yearly/monthly average velocity of wind and monthly maximum velocity of wind by 16 directions, yearly/monthly average occurrence frequency of wind, the prevailing wind, occurrence frequency of calmness, etc.) - Snowfall, sunlight, cloudiness, etc. - Occurrence frequency of fog and time of continuation, ect.
	2.4.2 Meteorology of site	o Plan of meteorological observation of site including methods of observation, and installation, correction, repair of observation instruments, procedures of data analysis, etc. shall be described. o Following data of weather which are observed at the site during the last more than one year at least shall be presented. - Temperature, rainfall, humidity (average by month, the highest and lowest temperature of the year and month) - Direction and velocity of wind (yearly/monthly average velocity of wind and monthly maximum velocity of wind by 16 directions, yearly/monthly average occurrence frequency of wind, the prevailing wind, occurrence frequency of calmness, etc.) - Appearance height and frequency of inversion layer and atmospheric mixing height, etc. - Occurrence frequency of fog and time of continuation, etc. - Atmospheric stability, distribution of direction and velocity of wind by season and year thereof o Topographical map within a 80-km radius shall be attached.
	2.4.3 Atmospheric dispersion during normal operation	o Computational models used in assessment of concentration of gaseous effluents discharged during the operation of plant, computer codes, computational methods, conditions of assumption of models, accuracy, and limitations, etc. shall be described. o Annual average of atmospheric dispersion factor (χ/Q) and deposition ratio (D/Q) within exclusion area boundary, the nearest residential area of population, and the area classified by distance and direction during investigation of the distribution of population shall be described.

Items	Description items	Description guidelines
	2.4.4 Atmospheric dispersion in the event of accidents	o Computational models used in assessment of concentration of gaseous effluents discharged in the event of accidents of plant, computer codes, computational methods, conditions of assumption of models, accuracy, limitations, etc. shall be described. o Atmospheric dispersion factor (χ/Q) at the exclusion area boundary and at the outer boundary of low-populated area by the time elapsed of accidents and by direction in the event of accidents shall be described. o Topographical longitudinal section by direction within a 10-km radius shall be attached.
2.5 Hydrology and hydrological dispersion	2.5.1 Status of surface water	o In regard of surface water system which is expected to be affected by execution of business, following investigation data along with charts shall be presented. - Characteristic data of watercourse such as location of river and size of basin, etc. - Characteristic data of reservoir such location of reservoir and size of catchment area, etc.
	2.5.2 Status of ground water	o In regard of ground water system which is expected to be affected by execution of business, following investigation data along with charts shall be presented. - Main aquifer and characteristic data of underground media (storage coefficient, transmissivity, hydraulic conductivity, porosity, adsorption coefficient by nuclide, etc.) - Direction and speed of ground water, and data of change of water level by season - Data for status of use of ground water
	2.5.3 Hydrological dispersion	o Computational models used in assessment of concentration of effluents discharged from plant, computer codes, computational methods, conditions of assumption of models, accuracy, limitations, etc. shall be described. o Transport and dispersion of effluents discharged from plant into environment shall be described. o Discharge point, distance of utilized point, traveling time, and dilution factor of ground water around the site shall be described including charts.
2.6 Status of ocean and oceanic dispersion	2.6.1 Status of ocean	o Period of investigation for items of investigation, location, summit, and observation instruments, etc. shall be described. o Following data which is investigated within a 16-km radius along the coast shall b e presented and attached with the related charts. - Characteristic data of ocean current, tide, tidal current, wind-driven current - Data of vertical structure of seawater and characteristic data of dispersion of material - Data of seafloor and topography of seashore including water depth chart - Characteristic data of floating sediment and submarine sediment

Items	Description items	Description guidelines
	2.6.2 Oceanic dispersion	o Computational models used in assessment of concentration of liquid phase effluents discharged during normal and abnormal operation of plant, computer codes, computational methods, conditions of assumption of model, conditions of input of model, boundary conditions, and limitations, etc. shall be described. o In regard of liquid phase effluents discharged during normal and abnormal operation, dilution factor and deposition factor within a 80-km radius along the coast shall be seasonally assessed and described including distribution charts and diagrams of major locations. o Distance, traveling time, and working time of the resident at major places such as fishing port, swimming beach, nursery, fishing grounds, and exclusion area boundary, etc. within a 80-km radius along the coast shall be described.
2.7 Population	2.7.1 Distribution of population	o Total number of population within a 80-km radius of site, density of population, an overpopulated area, etc. shall be described. o Status of constituent of population by age, sex, and industry shall be described.
	2.7.1.1 Resident Population within 20 km	o Concentric circles of 2, 4, 6, 8, 10, and 20-km radius with reactor in the center shall be divided into 16 directions, respectively, and the number of population by age group shall be described including charts. o Map that shows major overpopulated areas within a 20-km radius with reactor in the center shall be attached.
	2.7.1.2 Resident population between 20 and 80 km	o Concentric circles with intervals of 15 km between 20-km and 80-km radius with reactor in the center shall be divided as provided for in the foregoing paragraph and the number of population by age group shall be described including charts.
	2.7.1.3 Moving population	o Seasonal and monthly change of moving population within a 20-km radius with reactor in the center shall be described.
	2.7.2 Estimation of population	o The estimated population for the first year of operation of plant using the latest data of census and the estimated population for every ten years during the lifetime of plant shall be described including chart by types of distance and direction. o The estimation of population shall be predicted using the latest technique for estimation of population and natural and social increase and decrease of population.
2.8 Status of environmental radiation/radioactivity	2.8.1 Environmental radiation	o When the plant thereof is constructed at the new site, background level of environmental radiation around the area shall be investigated and described. o Following paragraphs shall be investigated and described when the plant thereof is constructed at the site where the existing nuclear power utilization facilities is operated.

Items	Description items	Description guidelines
		- Environmental radiation level at major positions and comparative positions within a 20-km radius of site - Set up of base-line level by the results of data analysis of environmental radiation which is continuously monitored more than one year
	2.8.2 Environmental radioactivity	o When the plant thereof is constructed at the new site, background level of environmental radioactivity around the area shall be investigated and described. o Following paragraphs shall be investigated and described when the plant thereof is constructed at the site where the existing nuclear power utilization facilities is operated. - Environmental radioactivity level at major positions and comparative positions within a 20-km radius of site shall be investigated and described. - Data of environmental radioactivity which is monitored more than one year shall be analyzed and base-line level shall be set up.
3. Status of plant 3.1 Appearance		o Layout of building including the boundary of plant shall be attached and described. o Discharge points of liquid and gas wastes shall be described and indicated on the layout of building.
3.2 Reactor and steam-electricity system		o Type, capacity, and power, etc. of reactor and turbine generator shall be described. o Using fuel shall be described in detail.
3.3 Fuel storage facilities		o Flow chart for cooling system of fuel storage, cleanup system, and exhaust system shall be presented. o Capacity of fuel storage pool shall be described. o Management of water inventory during refueling shall be described and source of make-up water shall be presented. o Concentration of radioactive materials in water of fuel storage pool after refueling shall be estimated. o Release rate of effluents which are evaporated and released as gases from the surface of water of fuel storage pool during normal operation and refueling shall be estimated.
3.4. Radioactive wastes treatment system	3.4.1 Gaseous radioactive wastes treatment system	o Treatment system of gas radioactive wastes shall be explained.

Items	Description items	Description guidelines
	3.4.1.1 Wastes treatment system	o Flow chart of treatment system of gas wastes shall be attached. o Facilities that are used continually and only in special case shall be described, respectively. o Facilities that are used in common with those of other reactors in site and with discharge points shall be described. o Capacity of treatment facilities and component equipment shall be described. o Time of stay, attenuation, and storage capacity, etc. of storage system shall be described.
	3.4.1.2 Treatment of radioactive materials	o Collection, processing, handling, storage, and discharge, etc. of gas radioactive wastes shall be described. o Estimated amount of gas wastes by types of source flowing into treatment system, discharge rate, planned decontamination factor, and time of stay, etc. shall be described.
	3.4.2 Liquid radioactive wastes treatment system	o Treatment system of liquid radioactive wastes shall be explained.
	3.4.2.1 Wastes treatment system	o Flow chart of treatment system of liquid wastes shall be attached. o Facilities that are used independently and in common with those of other reactors in site shall be separately described. o Capacity of storage tank, flow rate of system, and design capacity of each component equipment shall be described.
	3.4.2.2 Treatment of radioactive materials	o Collection, processing, handling, storage, and discharge, etc. of liquid radioactive wastes shall be described. o Estimated amount of liquid wastes by types of source flowing into treatment system, discharge rate, planned decontamination factor, and time of stay, etc. shall be described.
	3.4.3 Solid radioactive wastes treatment system	o Treatment system of solid radioactive wastes shall be explained.
	3.4.3.1 Wastes treatment system	o Flow chart of treatment system of solid wastes shall be attached. o Whether treatment system of solid radioactive wastes is used in common with that of other reactors in site or not shall be described. o Storage facilities of packaged solid wastes shall be explained. o Capacity of treatment system by types of components shall be described. o Facilities and operation modes for solidifying liquid radioactive wastes shall be described.

Items	Description items	Description guidelines
	3.4.3.2 Treatment of radioactive materials	o Handling, storage, and package for transport of solid radioactive wastes, etc. shall be described. o Types of packaged solid wastes and decay during storage thereof shall be described. o Expected amount of production of solid wastes by types of source and amount of radioactivity thereof shall be displayed as chart and described.
	3.4.4 Effluents monitoring	
	3.4.4.1 Discharge point of effluents	o Pathways of effluents discharged during operation shall be described. o Monitoring by continuous measurements and discharge points monitored by sampling analysis shall be separately described. o Drawings which show monitoring points shall be attached.
	3.4.4.2 Monitoring instruments	o Installed location, types, function, and characteristics of operation of monitoring instruments of effluents shall be described. o Setpoint of alarm, display equipment of warning, and warning sound of monitoring instruments of effluents, and matters of actions in case of alarm shall be described.
	3.4.4.3 Sampling	o Sampling methods, period, and analysis methods of effluents prior to their discharge into environment shall be described.
3.5 Radiation source	3.5.1 Source of radiation/radioactivity	o Source of radioactive nuclides flowing into treatment systems of gas, liquid, and solid radioactive wastes during normal operation and anticipated abnormal operation shall be described. o Concentration of activation corrosion product used in the calculation of source term shall be estimated and assumptions used in estimation thereof shall be described. o Activation phenomena of coolant, etc. in coolant system shall be described. o Sources and each concentration of radioactive isotopes in reactor coolant shall be described.
	3.5.2 Environmental release of radioactivity	o Possibilities that radioactivity is released into environment shall be described. o Amount of leakage of radioactive gas, particle, and especially radioactive iodine shall be estimated for each release source, and release pathways and transport process shall be described. o Release rate into building, etc. where is equipped with exhaust system shall be estimated.

Items	Description items	Description guidelines
	3.5.3 Gaseous radiation source	o Discharge amount and concentration of radioactive materials by nuclide discharged as gaseous phases shall be described. o Discharge height, discharge velocity, and discharge temperature shall be described and discharge point shall be specified. o In the case where there exists reference plant, accomplishment of the operation of reference plant shall be analyzed and annual average of discharge amounts shall be described.
	3.5.4 Liquid radiation source	o Discharge amount and concentration of radioactive materials by nuclide discharged as liquid phases shall be described. o In the case where there exists reference plant, accomplishment of the operation of reference plant shall be analyzed and annual average of discharge amounts shall be described.
4. Impacts due to construction		o In those cases where there are construction activities around the operating nuclear power utilization facilities, radiation source and exposure dose to construction worker caused by this from nuclear power utilization facilities thereof shall be assessed and described. o Exposure doses are calculated as maximum individual exposure dose and person-rem exposure. o Computational models and methods used in calculation of exposure dose shall be described.
5. Impacts due to operation		
5.1 Exposure pathways		o Direct/indirect exposure pathways for human and major organisms shall be described and diagram of exposure pathways shall be presented.
5.2 Assessment of exposure dose	5.2.1 Exposure through gaseous pathways	o Annual maximum exposure dose to individual through gaseous pathways within exclusion area boundary shall be calculated and described. o Annual person-rem exposure dose to the resident group within a 80-km radius of site shall be calculated and described. o Models and input data used in calculation shall be described.
	5.2.2 Exposure through liquid pathways	o Annual maximum exposure dose to individual through liquid pathways within exclusion area boundary shall be calculated and described. o Annual person-rem exposure dose to the resident group within a 80-km radius of site shall be calculated and described. o Models and input data used in calculation shall be described.
	5.2.3 Direct exposure from plant facilities	o Annual maximum external exposure dose to individual due to direct exposure within exclusion area boundary shall be calculated and described. o Models and input data used in calculation shall be described.

Items	Description items	Description guidelines
	5.2.4 Summary of exposure dose	o Annual exposure dose to group within the area thereof from the plant-related every member subject to exposure during the operation of plant shall be summarized and diagram shall be used if necessary. o Calculated maximum individual exposure dose shall be compared with the related criteria and assessed. o In those cases where numerous nuclear power utilization facilities including the relevant plant are constructed and operated within the same site, exposure dose from the whole of nuclear power utilization facilities thereof shall be calculated and compared with the related criteria and assessed.
6. Impacts due to accidents		
6.1 Assumption of accidents		o Potential accidents during the operation of plant shall be assumed to be classified by types and occurrence probability of postulated accidents by types shall be assessed. o Severe accidents shall be excluded from the subject of assessment.
6.2 Radiation source		o Radiation source of postulated accidents by types shall be described. o Assumptions used in the determination of radiation source shall be described.
6.3 Methods of assessment		o Computational models and input data used in calculation of exposure dose shall be described.
6.4 Assessment of exposure dose		o Maximum individual exposure dose in exclusion area boundary shall be calculated and described by types of postulated accidents. o Person-rem exposure dose within a 80-km radius of site shall be calculated and described.
6.5 Measures of public protection		o Overview of emergency plan for public protection, overview of organization, overview of activities for public protection (protection measures, notification, guideline of action, shelter and evacuation, etc.), environmental monitoring plan, etc. shall be described.
7. Environmental monitoring plan		
7.1 Environmental monitoring prior to operation		o Environmental monitoring plan for actual environmental impacts assessment due to construction of plant shall be described based on base-line data of pre-operation of plant which is a standard for environmental impacts assessment during operation of plant. o Technical matters such as measurement location classified by item, period, methods, and procedures, etc. shall be described.

Items	Description items	Description guidelines
7.2 Environmental monitoring during operation		o Environmental monitoring plan during operation of plant for environmental matters subject to be impacted by operation of plant shall be described based on environmental monitoring plan prior to operation. o Technical matters such as measurement location classified by item, period, methods, and procedures, etc. shall be described.
8. Consensus of public opinion		
8.1 Summary of opinion consensus		o The city mayors/county chiefs/district chiefs of competent jurisdiction (in those cases where business is extended over more than two administrative districts, related city mayors/county chiefs/district chiefs are included) shall be recorded. o The place and date of public inspection (in those cases where public hearing is executed, those are included) shall be recorded. o Occupation, name, and address of public, and name of agency that present their opinion shall be recorded.
8.2 Result of opinion consensus		o Opinion presenter (agency) by items of assessment which is classified by public inspection and public hearing, point of opinion, whether the opinion is reflected or not (reasons why the opinion is not reflected), number of page for the reflection, etc. shall be described in the order named thereof and using the chart. o Other appended materials related with consensus of public opinion such as a copy of notice for the result of draft report public inspection, a copy of public notification, and public hearing, etc. which are notified from the city mayors/county chiefs/district chiefs of competent jurisdiction shall be presented as appendixes.
9. Comprehensive assessment		o Expected environmental impacts and measures shall be comprehensively reviewed and final conclusive assessment for justification of the relevant business shall be described.
10. Others		o Following items, etc. that can be referenced to review of other reports shall be described. - Participant agencies and participants for preparation of report - Time required for preparation of report - Other references, etc.
11. References		o References used in preparation of report shall be mentioned.
Appendix : Summary		o Business plan and results of environmental impacts assessment, etc. shall be comprehensively summarized.

2. Guidelines for preparation of Radiation Environmental Report of Spent Nuclear Fuel Interim Storage Facilities

Items	Description items	Description guidelines
1. Overview of construction plan		o 1.1 Need for construction, 1.2 Basis of execution of environmental impacts assessment, 1.3 Circumstances of promotion of business, 1.5 Reasons for the site selection, etc. shall be described in accordance with those of plant.
1.4 Construction plan		o General construction plan of interim storage facilities shall be roughly described, and 1.4.1 Name, 1.4.2 Location of the place, 1.4.4 details of construction, etc. shall be described in accordance with those of plant.
	1.4.3 Overview of facilities	o Condition fo spent nuclear fuel, transportation method and plan, and types and capacity, etc. of major facilities (reception, storage, and cooling, etc.) shall be roughly described.
	1.4.5 Existing nuclear power utilization facilities	o In those cases where interim storage facilities are installed within the site of existing nuclear power utilization facilities, those shall be roughly described thereof.
2. Status of environment		o This shall be described in accordance with that of plant, and range of assessment of investigation shall be within a 10-km radius of site.
3. Status of facilities		
3.1 Appearance		o Layout of major buildings including boundary of interim storage facilities shall be attached and appearance of facilities shall be described. o Location of storage facilities of gas wastes, liquid wastes, and solid wastes shall be indicated and described on the layout of building.
3.2 Water		o Location and flow rate of intake and drain by water system shall be described. o Annual maximum demand of water and average of monthly and yearly demand of water shall be predicted and described by types of water source .
3.3 State of spent nuclear fuel		o Following paragraphs shall be described by types. - Spent nuclear fuel assembly and structure and weight of fuel rod - The elapsed time and expected period of storage after withdrawal from reactor on the basis of the date received. - Values on the time received and state of change during expected period of storage for composition of spent nuclear fuel, radioactivity (radioactivity by the whole and major nuclides), generation of decay heat, and temperature distribution of fuel assembly, etc.
3.4 Transportation of spent nuclear fuel		o Structure, weight, capacity, and cooling mode of transport cask, and temperature distribution and surface radiation dose rate of transport cask shall be described if spent nuclear fuel is included. o Structure, weight, and capacity, etc. of transportation vehicle shall be described. o Types and size of transportation vehicle and loading facilities shall be described.

Items	Description items	Description guidelines
3.5 Reception facilities		o Operation from loading the transport cask of spent nuclear fuel to storing fuel assembly shall be described by process. o Types, structure, and size, etc. of the related facilities in each process shall be described.
3.6 Storage facilities		o Storage method and structure and size of related facilities, and relationship between major subsystems related with storage facilities shall be described. o Range of water temperature of storage pool and radioactivity level in water during normal operation shall be described.
3.7 Cooling facilities		o Structure, size, and operation modes, etc. of cooling facilities related with storage pool of spent nuclear fuel and reception facilities shall be described. o Related equipments and instruments shall be described. o Others 3.8 Treatment system of radioactive wastes and 3.9 Radiation source shall be described in accordance with those of plant.
4. Impacts due to construction		o This shall be described in accordance with that of plant.
5. Impacts due to operation		o This shall be described in accordance with that of plant and the range of impacts assessment shall be within a 10-km radius of site.
6.～9.		6. Impacts due to accidents, 7. Environmental monitoring plan, 8. Consensus of public opinion, 9. Item of comprehensive assessment shall be described in accordance with that of plant.
10. Others		o This shall be described in accordance with that of plant.
11. References		o References used in preparation of report shall be mentioned.
Appendix : Summary		o Business plan and results of environmental impacts assessment, etc. shall be comprehensively summarized.

3. Guidelines for preparation of Radiation Environmental Report of Radioactive Waste Disposal Facilities

Items	Description items	Description guidelines
1. Overview of construction plan		o 1.1 Need for construction, 1.2 Basis of execution of environmental impacts assessment, 1.3 Circumstances of promotion of business, 1.5 Reasons for the site selection, etc. shall be described in accordance with that of plant.
1.4 Construction plan		o General construction plan of disposal facilities shall be roughly described, and 1.4.1 Name, 1.4.2 Location of the place, 1.4.4 Details of construction, etc. shall be described in accordance with those of plant.
	1.4.3 Overview of facilities	o State of low- and intermediate-level radioactive wastes, transportation method and plan, and types and capacity of major facilities (handling, ventilation, drain, etc.), etc. shall be roughly described.
	1.4.5 Existing nuclear power utilization facilities	o In those cases where disposal facilities are installed within the site of existing nuclear power utilization facilities, those shall be roughly described thereof.
2. Status of environment		o 2.5 Matters excluding hydrological and hydraulic characteristics shall be described in accordance with those of plant, range of assessment of investigation shall be within a 10-km radius of site.
2.5 Hydrological and hydraulic characteristics	2.5.1 Status of surface water	o Status of surface water system of site of disposal facilities and adjoining area shall be described, and the related charts of water system shall be attached. o Characteristics of botany and soil around the site of disposal facilities shall be investigated and described. o Control structure of surface water, waterway, and matters of artificial change shall be described. o Hydrological data of drain area, slope of the surface of the earth, flooding area, etc. shall be described. o Status of use of the existing surface water and future plan of use of the expected surface water shall be described. o Owner, location, type of use, quantity of use, source of supply, etc. shall be described.
	2.5.2 Status of ground water 2.5.2.1 General status	o Hydrological data of status such as state of fluctuation of ground water, size, and morphology, etc. at the site of disposal facilities and adjoining area shall be measured and presented during the last more than one year at least. o Boundary and structure of wide ground water system (excess of a 10-km radius of site) and local ground water system (within a 10-km radius of site) of the area including disposal facilities shall be hydrologically and geologically investigated, indicated as plane figure and cross section, and range of charging and drain area shall be illustrated and described. o In the case where aquifer is fractured aquifer, direction, density of distribution, and connectivity of fractured zone and joint which can be a main pathway of ground water shall be described.

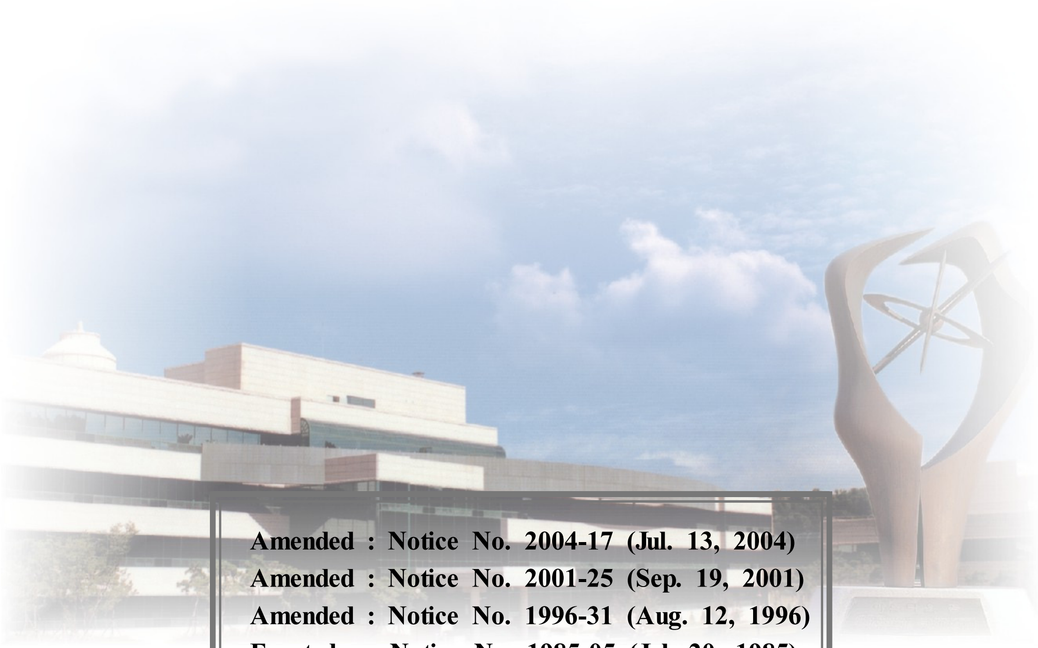
Items	Description items	Description guidelines
		o Status of use of ground water (user, location, source of water, quantity of use, etc.) and future status of the expected use within a 10-km radius shall be described. o Phenomena of intrusion of salt water around the site of disposal facilities shall be investigated and described, and the related data shall be attached. o Expected change of pathway of ground water after construction of disposal facilities shall be described. o In regard of all aquifers, following paragraphs shall be investigated and described. - Horizontal range and thickness - Permeability-related characteristics - Charging and drain area - Velocity of ground water - Conceptual model of other saturated zones and data that can be use in preparation of numerical model o In regard of all pumping wells and observation wells, following paragraphs shall be described. - Location, altitude, source and installed status of screen - Details of construction and closing - Hydrological and geological unit for observation o Depth and discharge of submerged pump of pumping well shall be described and their influential range shall be indicated on a plane figure. o Testing data for test of all aquifers, assumption, analysis and testing procedures shall be described. o Following paragraphs for underground media shall be described. - Storage Coefficients - Transmissivity - Hydraulic conductivity and porosity
	2.5.2.2 Unsaturated zone	o Following paragraphs for disposal site and adjoining area shall be investigated and described. - Thickness of horizontal range of permeable zone and impermeable zone - Potential pathways that can cause a large release abnormally - Change of water contents during wet and dry period o In the case of near surface disposal located at unsaturated zone, following paragraphs shall be additionally investigated and described. - Velocity of ground water in unsaturated zone - Spatial and layered distribution of porosity - Saturated hydraulic conductivity, interrelationship among water contents, pressure head, and hydraulic conductivity - Spatial change of hydrological variables, cross section, columnar section, and related map which can show horizontal range and pathway of ground water, etc. of impacted layers

Items	Description items	Description guidelines
3. Status of facilities 3.1 Appearance		o Layout of major buildings including control area of disposal facilities, boundary of site, and exclusion area boundary shall be attached and described. o Discharge points of liquid and gaseous wastes shall be described and indicated on the layout of building.
3.2 Disposal facilities		o Structure of facilities, capacity of facilities, facilities of intake and drain, facilities of ventilation, etc. shall be described. o Accessory facilities of disposal facilities (reception facilities, treatment facilities, package facilities, storage facilities, and decontamination facilities, etc.) and methods for final disposal, etc. shall be described. o Related drawings and system diagram, etc. necessary for explanation shall be attached.
3.3 Characteristics of wastes		o Characteristics of types, forms, and number, etc. of wastes which are expected to be undertaken presently or in the future shall be described.
3.4 Radiation source	3.4.1 Inventory by nuclide	o Inventory of important nuclides in wastes which are expected to be undertaken presently or in the future shall be described.
	3.4.2 Radiation source during operation	o Gaseous and liquid radiation source shall be described in accordance with those of nuclear power plant.
	3.4.3 Radiation after closure	o Leakage mechanism of major nuclides from disposal facilities after closure of disposal facilities shall be explained and described including corrosion rate of container and leaching rate of solid form.
4. ~ 5.		o 4. Impacts due to construction and 5. Impacts due to operation shall be described in accordance with those of plant, and range of impacts assessment shall be within a 10-km radius of site.
6. Impacts due to closure		o Expected pathways of radionuclides (air, ground water, surface water, absorption of plant, ocean, absorption of oceanic plant, etc.) that can be released from disposal facilities shall be described and diagram of pathways shall be presented. o Exposure dose (mSv/reasonal period) to individual by pathway of exposure within a 10-km radius of site due to potential leakage of radionuclides shall be calculated, and methods and models used in computation shall be described. o This shall be compared and assessed with the related criteria. o Discharge points of ground water located at downstream of disposal site (river, well, spring, pumping well, and sea, etc.) and concentration of important elements and nuclides at possible contact regions of human and livestock shall be predicted, assessed, and presented. o Related materials of assessment for concentration of nuclides by each distance, traveling time of ground water, pathways, etc. shall be presented in the foregoing data. o Movement of ground water including radioactive materials and influence of geochemical characteristics of rock and soil on concentration of radioactive element shall be analyzed and described. o Applied numerical or conceptual model shall be presented, characteristics of each model and input data such as dispersion coefficients and distribution coefficients of each nuclide, etc. shall be described.

Items	Description items	Description guidelines
7. Impacts due to accidents		o This shall be described in accordance with that of plant.
8. Environmental monitoring plan		o 8.1 Environmental monitoring prior to operation and 8.2 Environmental monitoring during operation shall be described in accordance with those of plant.
8.3 Environmental monitoring after closure		o Environmental monitoring plan during and after the institutional controlled period shall be distinguished and described each other. o Environmental monitoring plan after closure shall be formulated in order that the principle of preceding environmental monitoring plan is maintained.
9. Consensus of public opinion		o This shall be described in accordance with that of plant.
10. Others		o This shall be described in accordance with that of plant.
11. References		o References used in preparation of report shall be mentioned.
Appendix : Summary		o Business plan and results of environmental impacts assessment, etc. shall be comprehensively summarized.

【 04 】

Regulation on Survey of Radiation Environment and Assessment of Radiological Impact on Environment in Vicinity of Nuclear Power Utilization Facilities



Amended : Notice No. 2004-17 (Jul. 13, 2004)
Amended : Notice No. 2001-25 (Sep. 19, 2001)
Amended : Notice No. 1996-31 (Aug. 12, 1996)
Enacted : Notice No. 1985-05 (Jul. 20, 1985)

◎ **Notice of the Minister of Science and Technology No.2004-17 (MOST.reactor.007)**

The regulation on Survey of Radiation Environment and Assessment of Radiological Impact on the Environment in the Vicinity of Nuclear Power Utilization Facilities as provided for in Article 133 (2) of the Enforcement Regulation of the Atomic Energy Act is hereby notified publicly as follows:

July 13, 2004
Minister of Science and Technology

Regulation on Survey of Radiation Environment and Assessment of Radiological Impact on Environment in Vicinity of Nuclear Power Utilization Facilities

Article 1 (Purpose) The purpose of this notice is to prescribe the detailed matters required for implementation of survey of radiation environment (hereinafter referred to as "environmental survey") and assessment of radiological impact on the environment (hereinafter referred to as "environmental impact assessment") as provided for in Article 133 (2) of the Enforcement Regulation of the Atomic Energy Act.

Article 2 (Scope of Application) This notice shall apply to the implementation of environmental survey and environmental impact assessment in the vicinity of the relevant facilities by the installer and operator (hereinafter referred to as "enterpriser") of facilities (hereinafter referred to as "relevant facilities") as provided for in Subparagraphs 1 through 5 of Article 104-6 (1) of the Atomic Energy Act.

Article 3 (Definitions) Definitions of the terms used in this Notice shall be as follows:

1. The term "lowest limit of detection" means the minimum concentration of radioactivity which is measurable by the methods used for environmental survey;
2. The term "target value of detection" means the target value of concentration of radioactivity which is set by enterpriser himself in order to fulfill the lowest limit of detection; and

3. The term "minimum detectable concentration" means the lowest detectable concentration of radioactivity which is determined by conditions of measurement such as instruments of radioactivity, quantity of sample, rate of recovery, and time of measurement, etc.

Article 4 (Environmental Survey Plan) (1) The enterpriser shall establish the plan to perform the environmental survey as provided for in Subparagraph 1 of Article 133 (1) of the Enforcement Regulation of the Atomic Energy Act and then submit the plan to the Minister of Science and Technology one year prior to the enforcement of environmental survey.

(2) The environmental survey plan as provided for in the foregoing Paragraph 1 shall be formulated as follows:

1. The organization that executes the environmental survey and environmental impact assessment and its responsibility shall be described;
2. Characteristics of the relevant facilities and site shall be roughly described from the viewpoint of environmental survey;
3. Items and contents of environmental survey shall be described in detail in comparison with the purpose of use for the result of environmental survey;
4. Types and their characteristics of instruments for measuring radiation and radioactivity, methods of measurement, and procedures, etc. shall be described;
5. Methods such as sampling for the measurement of radioactivity, sample preparation, analysis, and verification of reliability for the result of measurement, etc. shall be briefly described, and the target value of detection of survey item shall be set up and described;
6. Statistical handling and assessment of the result of environmental survey, and record keeping shall be described;
7. Method of environmental impact assessment shall be described; and
8. Quality control plan for environmental survey shall be described.

(3) The environmental survey plan as provided for in the foregoing Paragraph 1 shall be re-examined every period of not in excess of three years. When the contents of environmental survey plan are about to be changed, the reason for change and environmental survey plan to be changed shall be submitted to the Minister of Science and Technology six months prior to the enforcement of environmental survey. Provided, that minor change may be reported after execution.

(4) The Minister of Science and Technology may give orders that amendment or supplement of the environmental survey plan shall be done by enterpriser when the

contents of environmental survey plan submitted by the regulation as provided for in the foregoing Paragraphs 1 or 3 are deemed to be insufficient for assessing the environmental impact by the relevant facilities.

Article 5 (Quality Control) In regard of quality control as provided for in Subparagraph 2 of Article 133 (1) of the Enforcement Regulation of the Atomic Energy Act, plan for each of the followings shall be formulated and verification for the result of environmental survey shall be performed more than once a year:

1. Sampling and conveyance;
2. Sample preparation;
3. Measurement of radiation and analysis of radioactivity;
4. Analysis of the result of survey and statistical handling; and
5. Report on the result of survey.

Article 6 (Guideline of Environmental Survey) Environmental survey as provided for in Subparagraph 3 of Article 133 (1) of the Enforcement Regulation of the Atomic Energy Act shall conform to the followings:

1. Survey item, survey period, and survey location shall conform to "Guidelines for survey of environmental radiation/radioactivity" in Table 1, and may be adjusted according to characteristics of the relevant facilities and site;
2. The starting of environmental survey shall be a point of time when basic environmental survey data, a standard for assessing environmental impact by operation of the relevant facilities, can be sufficiently secured, and the time shall be at least two years before the operation of the relevant facilities;
3. The ending of environmental survey shall be a point of time when environmental impact is deemed not to exist along with the assessment of the result of environmental survey after closure of the relevant facilities; and
4. Sampling, sample preparation, analysis and measurement for environmental survey shall be performed according to method and technique to satisfy "The lowest limit of detection for analysis of environmental radioactivity" in Table 2, and procedures that describe each process classified by nuclide and sample of the subject of survey shall be formulated and peformed accordingly.

Article 7 (Environmental Impact Assessment) Environmental impact assessment as provided for in Subparagraph 4 of Article 133 (1) of the Enforcement Regulation of the Atomic Energy Act shall be performed according to the followings:

1. Radiation exposure dose to residents by radioactive materials or radiation discharged from the relevant facilities shall be calculated and compared/assessed with the criteria. In this case, computational codes for radiation exposure dose to residents shall reflect characteristics of site and peripheral environmental factors, and the reliability of assessment model shall be secured;
2. Long-term trends of accumulation and fluctuation of radioactive materials around the relevant facilities shall be assessed on the basis of the result of environmental survey, and short-term fluctuation due to the unexpected discharge of radioactive materials from the relevant facilities shall be assessed; and
3. On the basis of the result of survey for environmental samples subject to ingestion of human, radiation exposure dose shall be assessed in case of ingestion of those sample.

Article 8 (Processing of Environmental Survey Data) (1) Average value and fluctuating range (minimum～maximum) in normal times shall be established by survey location and environmental survey item.

(2) Average value and fluctuating range in the Paragraph 1 shall be established by the result of environmental survey for the relevant facilities in pre-operation and by the result of environmental survey which has been performed around the relevant facilities in normal operation during the latest more than three years at least. In this case, the result of environmental survey by abnormal cause shall be excluded.

(3) The result of environmental survey shall be concisely recorded using a proper prefix and the international standard unit.

(4) The result of environmental survey shall be marked in consideration of a number of significant figures in order to watch the degree of measurement and analysis, and its consistency shall be maintained by item of the subject of survey.

(5) In case where radionuclides are not detected, minimum detectable concentration shall be specified and indicated as below the concentration, and minimum detectable concentration shall be used in the calculation of average value as a result of environmental survey.

(6) In case where all the results of survey are less than minimum detectable concentration, average value in the Paragraph 1 shall not be indicated, and the fluctuating range in normal times shall be specified as the lowest value among the minimum detectable concentrations and indicated as below the lowest value.

Article 9 (Record Keeping) (1) The result of environmental survey and environmental impact assessment shall be kept until the end of the business of environmental survey.

(2) All circumstantial conditions for sampling, sample preparation, measurement, and analysis shall be recorded in detail by survey item and kept for more than five years so that it is possible to examine in the reverse order of environmental survey in preparation for the case where reassessment for the result of environmental survey is necessary.

(3) Environmental samples for which measurement/analysis is completed shall be preserved per sample during the determined period in preparation for the case where reassessment is necessary.

Article 10 (Report) Enterpriser's report on the result of environmental survey and environmental impact assessment as provided for in Article 104-6 (1) of the Atomic Energy Act shall conform to the followings:

1. In those cases where matters which fall under each of the followings are detected from the result of environmental survey, those cases shall be reported according to the form of Table 3 within one week since the detection:
   a. The case where average value in one hour for ambient gamma dose rate in the middle of continuous monitoring at a fixed location is in excess of an average value of data in the latest more than three years by 10 μR/h (only the secured data if less than that period);
   b. The case where the result of analysis of radioactivity at sampling locations of survey plan proves that it is in excess of five times of average value of data in the latest more than three years (only the secured data if less than that period); and
   c. The case where an artificial radionuclide is detected in environmental samples which are measured less than minimum detectable concentration during the latest three years.
2. The result of environmental survey which is performed during the first half of a year shall be formulated and submitted as a report until September 30 of the year concerned.
3. Comprehensive results for environmental survey and environmental impact assessment performed every year shall be formulated and submitted as a report until March 31 of the next year.

Article 11 (Application to the Site of Complex Facilities) (1) In case where there exist more than one relevant facility on the same site, one environmental survey plan for unified environmental survey and environmental impact assessment may be established and implemented.

(2) In case where there exist more than one enterpriser when Paragraph 1 is applied, unified environmental survey plan may be established and implemented upon deliberation among enterprisers.

Addenda

Article 1 (Enforcement Date) This notice shall enter into force on the date of its promulgation.

Article 2 (Transitional Measures) Enterpriser who performs the environmental survey around the relevant facilities as provided for in Notice of the MOST No.2001-25 at the time of the enforcement of this notice shall formulate the environmental survey plan as provided for in Article 4 or Article 11 within three months from the enforcement date of this notice and submit it to the Minister of Science and Technology.

Article 3 (Repeal of Notice) Notice of the MOST No.2001-25 "Regulation on Radiation Environmental Survey and Radiation Environmental Impacts Assessment around Nuclear Power Utilization Facilities" is repealed at the time this notice enters into force.

[Table 1]

Guidelines for Survey of Environmental Radiation/Radioactivity

a. Survey item and selection of period

Class	Survey item		Survey period	
	Environmental media	Monitored nuclide	Frequency of sampling	Frequency of analysis
Radiation	environmental radiation	ambient gamma dose rate	continuous monitoring	once a month
		collective dose (TLD)		once a quarter
Sample of land	air	gross beta, ^{14}C, ^{131}I, U, gamma isotopes	continuous sampling	once a month
	moisture in air	^{3}H		twice a month
	drinking water / ground water	^{3}H, U, gamma isotopes	once a quarter	once a quarter
	rainwater / surface water	gross beta, ^{3}H, U, gamma isotopes	once a month	once a month
	surface soil	^{90}Sr, Pu, U, gamma isotopes	twice a year	twice a year
	river soil	U, gamma isotopes	once a quarter	once by season
	agricultural product	^{3}H, ^{14}C, ^{90}Sr, U, gamma isotopes	harvesting season	twice a year
	meat	^{14}C, gamma isotopes	twice by year	twice a year
	milk	^{3}H, ^{90}Sr, ^{129}I, ^{131}I, gamma isotopes	once a month	once a month
		^{14}C		once a quarter
	biological indicator	^{90}Sr, gamma isotopes	twice a year	twice by year
Sample of sea	sea water	gross beta, ^{3}H	once a week	once a month
		^{90}Sr, Pu, gamma isotopes		once a quarter
	submarine sediment	^{90}Sr, Pu, gamma isotopes	twice a year	twice a year
	fish and shellfish	^{90}Sr, Pu, gamma isotopes	twice a year	twice a year
	seaweed	^{90}Sr, ^{99}Tc, ^{129}I, ^{131}I, gamma isotopes	twice a year	twice a year

b. Selection of survey location

Class	Survey location
Environmental radiation	① Location for continuous monitoring and periodic measurement of environmental radiation shall be selected in consideration of distance from the relevant facilities, direction of wind, and density of population, etc., and overpopulated area shall be preferentially selected. ② Measurement of environmental radiation shall be done in principle at a height of 1 m from the earth's surface in consideration of the effect of earth's radiation, and especially, measurement shall be done above the general soil or lawn of a vastly open land.
Sample of land	① Sampling location shall be equally distributed by distance and direction around the facilities, area of high possibility of contamination shall be preferentially selected in consideration of regional meteorological data, regional characteristics, and assessment of atmospheric dispersion, etc. ② Optimal sampling location shall be selected in consideration of survey item, geological characteristics, and possibility of sampling, ect., and then sampling shall be done at the selected location.
Sample of sea	① Area of high possibility of contamination shall be preferentially selected in consideration of flow of sea. ② Intake and drain shall be included as survey locations. ③ The location shall be selected in consideration of status of use of sea (fishing grounds, nursery, and recreation area, etc.).
Comparative location	① The location which is expected to be not affected by the relevant facilities shall be selected in consideration of minimum downwind area, clean sea area, and distance from the relevant facilities, etc. ② Comparative location over one point shall be set up by each environmental survey item.

[Table 2]

The Lowest Limit of Detection for Analysis of Environmental Radioactivity

Class	Air (Bq/m^3)	Water (Bq/L)	Soil (Bq/kg-dry)	Milk (Bq/L)	Agricultural/livestock product (Bq/kg-fresh)	Sea water (Bq/L)	Marine organism (Bq/kg-fresh)	Submarine sediment (Bq/kg-dry)
gross β	3×10^{-4}	0.15	-	-	-	3	-	-
^{3}H	0.1	5	-	5	5^{*}	5	-	-
$^{14}C^{**}$	0.25	-	-	-	0.25	-	-	-
^{51}Cr	5×10^{-3}	1	15	1	3	5×10^{-2}	6	15
^{54}Mn	8×10^{-5}	0.5	2	0.2	1	5×10^{-3}	2	2
^{58}Co	3×10^{-4}	5×10^{-2}	2	0.2	1	5×10^{-3}	2	2
^{60}Co	8×10^{-5}	2×10^{-2}	2	0.2	1	5×10^{-3}	2	2
^{59}Fe	5×10^{-4}	3×10^{-2}	5	0.5	2	5×10^{-3}	3	5
^{65}Zn	5×10^{-4}	5×10^{-2}	5	0.5	2	2×10^{-2}	3	5
^{90}Sr	5×10^{-6}	1×10^{-3}	0.5	0.1	0.1	1×10^{-3}	0.1	0.5
^{95}Zr-Nb	5×10^{-4}	0.5	5	0.3	0.5	6×10^{-3}	1	5
^{99}Tc	2×10^{-4}	1	0.1	-	5×10^{-3}	-	-	0.1
^{106}Ru	-	5×10^{-2}	15	-	-	-	1	15
^{129}I	5×10^{-2}	-	1×10^{-2}	0.5	2×10^{-5}	1	-	1×10^{-2}
^{131}I	1×10^{-2}	0.1	3	0.2	0.5	0.1	1	1.5
^{134}Cs	8×10^{-5}	8×10^{-3}	5	0.2	0.1	3×10^{-3}	0.1	5
^{137}Cs	8×10^{-5}	8×10^{-3}	5	0.2	0.1	3×10^{-3}	0.1	5
^{140}Ba-La	0.1	10	70	10	2	0.1	5	70
^{238}U	3×10^{-6}	0.1	20	-	5×10^{-2}	-	-	-
$^{239+240}Pu$	2×10^{-6}	2×10^{-4}	0.1	-	1×10^{-3}	5×10^{-6}	1×10^{-3}	0.1

*) unit of ^{3}H in food : Bq/L (based on concentration of ^{3}H in tissue free water)

**) unit of ^{14}C : Bq/g-C

Footnote 1) Low limit of detection for biological indicator shall follow that of food.

Footnote 2) Low limit of detection above shall be applied only to monitored nuclides provided in environmental survey plan

Footnote 3) In case of river soil, low limit of detection for submarine sediment shall be applied.

[Table 3]

Report on Temporary Increase of Environmental Radiation/Radioactivity

1. Name of business :
2. Content of report on temporary increase

Class	Content of report				
	Environmental media	Reported nuclide	Sampling date	Location (inside and outside of site, comparative)	Date of detection
Radiation	Environmental radiation	Ambient gamma dose rate			
Sample of land					
Sample of sea					

Class	Content of report		More than 3 years lately		
	Measured value	Unit	Average value at the location	Fluctuating range at the location (min. ~ max.)	Maximum value throughout the locations
Radiation					
Sample of land					
Sample of sea					

3. Cause for temporary increase
   - ☐ Effect of facilities ☐ Natural phenomena ☐ Sampling environment
   - ☐ Unknown cause ☐ Others( )
4. Matters of actions
   - ☐ Unnecessary additional inquiry of cause
   - ☐ Documentary report after additional inquiry of cause
5. Reasons why additional inquiry of cause is unnecessary
   - ☐ Appending documentary evidence for natural phenomena
   - ☐ Appending document for inquiry of cause
   - ☐ Substituted by analysis of cause and assessment
6. Analysis of cause and assessment

To Minister of Science and Technology :

I hereby submit a report on temporary increase of environmental radiation/radioactivity as provided for in Subparagraph 1 of Article 10 of Notice of the MOST No. 2004-17.

Date (month/day/year) : ____________________

Submitted by : ____________________ Signature: ____________________

※ **Annotation of Table 3**

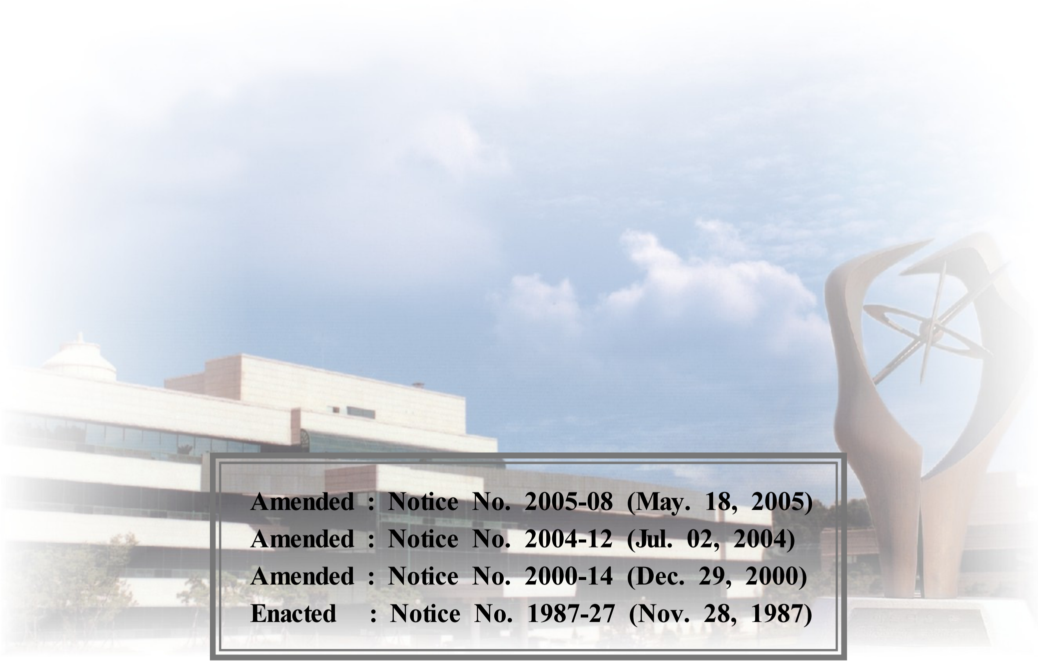
Matters to be attended to formulation of report on temporary increase of environmental radiation/radioactivity

1. In those cases where the cause of temporary increase is subject to complex factors such as the effect of facilities and natural phenomena, etc., all cases shall be recorded.
2. In cases where there exist contents which conform to the following paragraphs in analysis of cause and assessment, all cases shall be recorded and comprehensively assessed:
   a. Artificial radionuclides was not detected as a result of an analysis of gamma nuclides for the relevant sample;
   b. Artificial radionuclides besides ^{137}Cs was not detected as a result of an analysis of gamma nuclides for the relevant sample;
   c. There is a circumstance when radiation/radioactivity is temporarily increased due to environmental conditions such as sampling location, and sampling time, etc., and the increase has no connection with the operation of nuclear power utilization facilities;
   d. Investigated values at the entire places besides locations where there exists temporary increase maintain the fluctuating range of normal times, and the investigated level is less than maximum value measured at the entire places during the latest past three years;
   e. Investigated concentration is measured more than minimum detectable concentration, but less than low limit of detection for analysis of environmental radioactivity of Table 2;
   f. Investigated data at the same place is not secured during more than the latest past three years;
   g. Investigated data is less than maximum value measured at the same place and in the relevant year; and
   h. Others.

Notice of the Minister of Science and Technology No.2005-08 (MOST.reactor.009)

【 05 】

Regulation on Other Facilities related to Safety of Nuclear Reactor



Amended : Notice No. 2005-08 (May. 18, 2005)
Amended : Notice No. 2004-12 (Jul. 02, 2004)
Amended : Notice No. 2000-14 (Dec. 29, 2000)
Enacted : Notice No. 1987-27 (Nov. 28, 1987)

◎ **Notice of the Minister of Science and Technology No.2005-08 (MOST.reactor.009)**

The Regulation on Other Facilities related to Safety of Nuclear Reactor (Notice of the MOST No.2004-12, July 2, 2004) is hereby amended and notified publicly as follows:

May 18, 2005
Minister of Science and Technology

Regulation on Other Facilities related to Safety of Nuclear Reactor

Article 1 (Purpose) The purpose of this notice is to prescribe regulation on "the other facilities related to safety of nuclear reactor" as provided for in Subparagraph 8 of Article 9 of the Enforcement Decree of the Atomic Energy Act.

Article 2 (Other Facilities related to Safety of Nuclear Reactor) (1) "The other facilities related to safety of nuclear reactor " as provided for in Subparagraph 8 of Article 9 of the Enforcement Decree of the Atomic Energy Act shall be the following facilities:

1. Structures;
2. Service water system facilities;
3. Heating, ventilation and air conditioning system facilities;
4. Electric power system facilities;
5. Auxiliary system facilities; and
6. Power conversion system facilities.

(2) The details of each facility as provided for in Paragraph 1 are specified in the Table.

Article 3 (Facilities subject to Application) If it is difficult to apply the provisions in Article 2 to a nuclear reactor and its related facilities due to the difference in its utilization purpose or design features, then the provisions in Article 2 can be applied to the facilities having the equivalent functions to the facilities specified in Article 2.

Addendum

This notice shall enter into force on the date of its promulgation.

【Table】

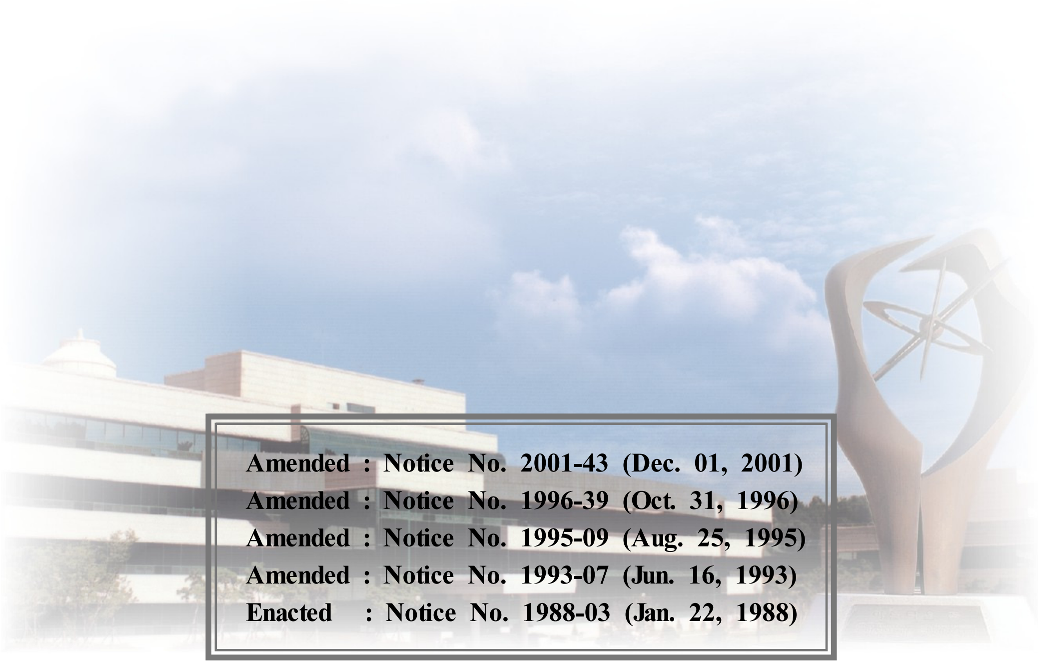
Other Facilities related to Safety of Nuclear Reactor
(related to Article 2 (2))

Name of facility	Items
1. Structures	A. Control building B. Auxiliary building C. Nuclear fuel handling building D. Emergency diesel generator building E. Component cooling water heat exchanger building F. Primary-side component cooling sea-water intake/discharge structures G. Emergency feedwater storage structures H. Class 1E under ground conduit (Cable tunnel) I. Radioactive waste management building J. Turbine building
2. Service water system facilities	A. Component cooling sea-water system B. Component cooling water system C. Condensate water storage and transport system D. Refueling water storage system E. Cooling water system F. Reactor make-up water system
3. Heating, ventilation and air conditioning system facilities	A. Control building B. Auxiliary building C. Nuclear fuel handling building D. Reactor containment building E. Engineered safety features F. Radioactive waste management building
4. Electric power system facilities	A. Off-site electric power system B. On-site AC power system C. On-site DC power system

Name of facility	Items
5. Auxiliary system facilities	A. Chemical and volume control system B. Compressed air system C. Radioactive drainage system D. Sampling system E. Fire protection system F. Emergency diesel generator fuel storage and transport system G. Safety and de-pressurization valve H. Piping, support, and snubber
6. Power conversion system facilities	A. Main steam system B. Steam generator blowdown system C. Feedwater and condensate system D. Auxiliary feedwater system E. Turbine and auxiliary system F. Generator and related system G. Auxiliary steam system

【 06 】

Regulation on Disposition and Control of Inspection Findings of Nuclear Power Utilization Facilities



Amended : Notice No. 2001-43 (Dec. 01, 2001)
Amended : Notice No. 1996-39 (Oct. 31, 1996)
Amended : Notice No. 1995-09 (Aug. 25, 1995)
Amended : Notice No. 1993-07 (Jun. 16, 1993)
Enacted : Notice No. 1988-03 (Jan. 22, 1988)

◎ **Notice of the Minister of Science and Technology No.2001-43 (MOST.reactor.010)**

The regulations on disposition and control of inspection findings pointed out in various inspections of nuclear power utilization facilities and nuclear licensees under the Atomic Energy Act are hereby notified publicly as follows:

December 1, 2001
Minister of Science and Technology

Regulation on Disposition and Control of Inspection Findings of Nuclear Power Utilization Facilities

Article 1 (Purpose) The purpose of this notice is to provide regulations on affairs pertaining to the disposition and control of the findings pointed out in inspections of nuclear power utilization facilities and nuclear licensees as provided for in Articles 16, 23-2, 36, 45, 59, 78 and 103 of the Atomic Energy Act (hereinafter referred to as "Act") (including the orders issued as the inspection findings under Articles 30, 36 and 54 of the Act).

Article 2 (Definitions) The terms used herein are defined as follows:

1. The term "entrusted institution" means an institution delegated by the Minister of Science and Technology (hereinafter referred to as "Minister") to perform affairs related with inspections under Article 111 of the Act;
2. The term "inspection officer" means an employee of the Ministry of Science and Technology (including a resident officer) who performs inspections under the Act;
3. The term "resident officer" means an employee of the Ministry of Science and Technology who is stationed at a nuclear power plant or radioactive waste management facilities in order to carry out affairs related with safety control of nuclear facilities according to Article 12 of the Office Regulations of the Ministry of Science and Technology and Organizations under Its Command;
4. The term "inspector" means a person holding inspector certificate who performs inspections as commissioned by the Minister according to the regulations on handling of entrusted affairs as provided for in Article 111 (5) of the Act;
5. The term "inspecting person" means an inspection officer or an inspector;

6. The term "inspection findings" means matters that are pointed out due to violations of the laws, permit conditions, technical standards, procedures, drawings and so forth related with nuclear safety or due to non-performance of orders from the Minister; and
7. The term "recommendations" means matters requiring improvement or complementation by nuclear licensees for the purpose of enhancing nuclear safety, which do not fall under inspection findings.

Article 3 (Preparation of Inspection Findings Form) (1) When an inspecting person recognizes inspection findings, such person shall immediately prepare Inspection Findings Form as provided for in Attached Form 1. Provided, however, that the blank for "Corrective Action Required" in the Inspection Findings Form shall be filled in by the authorities that issue the Inspection Findings Form (hereinafter referred to as "issuing authorities").

(2) The Inspection Findings Form as provided for in the foregoing Paragraph 1 shall be prepared according to each of the followings:

1. The blank for "Inspecting Person" shall be filled with the organization and name of the inspecting person who recognized inspection findings, and the signature of such person shall be attached thereto;
2. The blank for "Inspected Person" shall be filled with the organization and name of the inspected person, with the signature of such inspected person attached thereto. In such case, the inspecting person shall fully explain to the inspected person the details of findings and bases for the findings;
3. In the blank for "Title", a simple description to promote understanding of the contents of findings shall be provided;
4. In the blank for "Contents of Findings", the details of matters pointed out shall be described clearly;
5. In the blank for "Bases for Findings", the details of the rationale that served as the basis for findings (laws, technical standards, procedures and so forth) shall be described specifically;
6. In the blank for "Corrective Action Required", specific matters that the Inspected Institution should rectify or improve shall be described clearly; and
7. Matters other than those set forth in the foregoing Subparagraphs 1 through 6 shall be prepared according to Schedule 1 and Schedule 2.

(3) The head of the entrusted institution shall send the Inspection Findings Form to the issuing authorities within 7 days from the date of preparation thereof. Provided,

however, that as regards any matters that fall under any Subparagraph of Article 4 (1) hereof, the Inspection Findings Form shall be promptly sent upon preparation thereof.

(4) The "Opinion of the Inspecting Authorities" prepared according to the due form as provided for in Attached Form 3 shall be attached to the Inspection Findings Form as provided for in the foregoing Paragraph 3.

Article 4 (Issuance of Inspection Findings Form) (1) The Minister shall issue an Inspection Findings Form to a nuclear licensee who underwent an inspection. Provided, however, that as regards minor inspection findings that do not fall under any of the followings, Article 5 hereof shall apply:

1. Matters subject to a penalty or a fine for negligence according to Articles 114 through 120-2 of the Act;
2. Matters that fail to meet the standards for permit, license or designation as provided for in the Act;
3. Matters in violation of the conditions imposed on permit, license or designation under Article 104 (1) of the Act;
4. Matters that need to be ordered according to the Act by the Minister for the sake of safety including suspension of use, modification, repair, transfer, designation of operation method, change of technical specifications for operation, and decontamination; or
5. Other common important matters that must be rectified for the purpose of nuclear safety.

(2) The issuing authorities shall issue the Inspection Findings Form within 7 days from the date of preparation of such Inspection Findings Form or receipt of Inspection Findings Form from the inspecting authorities.

(3) The Minister may take necessary measures including corrective actions and requests for supplementation by means of official documents instead of issuing the Inspection Findings Form.

Article 5 (Issuance of Minor Inspection Findings Form) (1) In a place of business where a resident officer is stationed, the resident officer shall issue the Inspection Findings Form to the licensee subject to an inspection. In such case, the licensee refers to the head of the place of business.

(2) In a place of business where a resident officer is not stationed, the head of an entrusted institution shall issue the Inspection Findings Form to the licensee subject

to an inspection.

(3) If any person to whom the Inspection Findings Form is issued as provided for in the Paragraphs 1 and 2 has an objection to the findings and so forth, such person shall file the objection to the issuing authorities within 7 days from its receipt.

(4) Any authorities that receive the objection in accordance with the Paragraph 3 shall promptly send to the Minister the relevant Inspection Findings Form and the contents of such objection.

(5) If the results of corrective actions as provided for in Article 9 (2) hereof are unsatisfactory, the issuing authorities pursuant to the foregoing Paragraphs 1 and 2 shall make a report thereof to the Minister.

Article 6 (Control and Utilization of Inspection Findings) (1) The issuing authorities of Inspection Findings Form shall record and maintain the "Book of Controlling Inspection Findings" in such due form as set out in Attached Form 4.

(2) When issuing Inspection Findings Form in accordance with Articles 4 and 5 hereof, the issuing authorities shall send a copy thereof to the head of the entrusted institution.

(3) When approving requested extension of the period as provided for in Article 8 (2) hereof or closing inspection findings according to Article 10 hereof, the issuing authorities shall give notice thereof to the head of the entrusted institution.

(4) In order to enhance the safety of nuclear facilities, etc. and ensure efficient performance of entrusted affairs, the head of the entrusted institution shall conduct general computerized management of the status of the disposition of inspection findings and periodically make a comprehensive analysis and use of the status of issuance of Inspection Findings Form.

Article 7 (Recommendations) (1) If an inspecting person discovers any matters for recommendations, he/she may issue a Recommendations Form in such due form as provided for in Attached Form 5. The issuing authorities of the Recommendations Form shall be the same as those issuing Inspection Findings Form as provided for in Articles 4 and 5 hereof.

(2) A nuclear licensee shall exert its endeavor to implement the recommendations.

Article 8 (Submission of Report on Corrective Actions of Inspection Findings) (1) A nuclear licensee shall take corrective or complementary measures by the requested date in accordance with the Contents of Required Corrective Actions and submit to the issuing authorities a "Report on Corrective Actions of Inspection Findings" in such due form as provided for in Attached Form 6.

(2) If there exist any circumstances preventing completion of corrective or complementary measures by the requested date, a nuclear licensee may apply for a postponement prior to the requested date. The issuing authorities may approve such application if deemed justifiable in its reasonable discretion.

Article 9 (Review and Supplementation of Report on Corrective Actions of Inspection Findings) (1) The issuing authorities shall review and confirm the adequacy of the results of corrective actions. In such case, the authorities may receive technical advice from the institution that prepared the Inspection Findings Form or from other organizations.

(2) The issuing authorities, when finding any unsatisfactory matters, shall request supplementation thereof by designating a requested date. The licensee that receives such request shall submit the results of relevant corrective actions in accordance with Article 8 hereof.

Article 10 (Closing of Inspection Findings) If the results of corrective actions are appropriate, the issuing authorities shall close inspection findings and give notice thereof to the relevant nuclear licensee.

Article 11 (Relations with Report on Entrusted Affairs) Any matters submitted or reported by the head of the entrusted institution hereunder shall not be deemed a report under Article 312 of the Enforcement Decree of the Act.

Addenda

Article 1 (Enforcement Date) This notice shall enter into force on January 1, 2002.

Article 2 (Repeal of Notice) Notice of the MOST No.96-39 (October 31, 1996) is repealed at the time this notice enters into force.

[Attached Form 1]

Inspection Findings Form

Ministry of Science and Technology [ ]
Inspected Institution [ ]
Inspecting Authorities [ ]

① Control Number	- -	② Inspection Type	-	③ Finding Type	
④ Inspected Institution		⑤ Reactor Type		⑥ Inspected System	
⑦ Inspecting Person	*(Signature)*	⑧ Inspection Date	. .	⑨ Issue Date	. .
⑩ Inspected Person	*(Signature)*	⑪ Requested Date	. .	⑫ Closing Date	. .
⑬ Title					
⑭ Contents of Findings					
⑮ Bases of Findings					
⑯ Corrective Actions Required					

Approved on June 10, 1993 (A4), ≪If you have any objection to the inspection findings, you may file an application for objection within seven days.≫

Inspection Findings Form(continued)	① Control No.			-		-			
⑬ Title									

Approved on June 10, 1993 (standard size: A4)

[Attached Form 3]

Opinion of the Inspecting Authorities	① Control No.			-		-			
⑬ Title									

Contents of Corrective Actions Required

Special Issue or Problem Noted during Inspection

Approved on June 10, 1993 (standard size: A4)

[Attached Form 4]

Book of Controlling Inspection Findings

Control No. ☐☐-☐-☐☐☐	Inspection Type ☐☐-☐	Finding Type ☐	Inspected Institution ☐☐	Reactor Type ☐☐☐☐	Title	Inspection Date	Issue Date	Requested Date	Closing Date	Remarks

Approved on June 10, 1993 (standard size: B4 horizontal)

[Attached Form 5]

Recommendations Form

Issue No.	☐☐ - ☐ - ☐☐☐	Inspected Institution: Inspection Type: Inspecting Person: *(Signature)* Issue Date:

Title:

<u>Inadequacies</u>

<u>Bases for Recommendation</u>

<u>Contents of Recommendation</u>

Approved in August 1995 (standard size: A4)

[Attached Form 6]

<table>
<tr><td colspan="4" align="center">Report on Corrective Actions
of Inspection Findings</td></tr>
<tr><td>Related Document</td><td>(. .)</td><td>① Control No.</td><td>[ ][ ]-[ ]-[ ][ ][ ]</td></tr>
<tr><td>⑬ Title</td><td colspan="3"></td></tr>
<tr><td colspan="4"><u>Contents of Corrective Actions (Summary)</u></td></tr>
<tr><td colspan="4"><u>Measures for Prevention of Recurrence</u></td></tr>
<tr><td>Confirmation by the Inspected Institution</td><td colspan="3">(Title) (Name) (Signature) 200 . .</td></tr>
<tr><td colspan="4"><u>Review</u></td></tr>
<tr><td>Review by the Inspecting Authorities</td><td>Reviewed by (Signature)</td><td colspan="2">Confirmed by (Signature)</td></tr>
</table>

Approved on June 10, 1993 (standard size: A4)

[Schedule 1]

Composition and Contents of Entry Codes regarding Inspection Findings Form

Classification	Code	Contents
① Control Number	□□ - □ - □□□ (Year) (Issuer) (Serial Number)	◦ As regards the Year, enter the last two digits of the issuing year of the Inspection Findings Form. ◦ Issuer of the Inspection Findings Form (Issuing Institution) 1 Minister of Science and Technology 2 Head of the Entrusted Institution 3 Resident Officer at Kori 4 Resident Officer at Wolsong 5 Resident Officer at Yonggwang 6 Resident Officer at Uljin ◦ As regards the Serial Number, start from 001 every issuing year.
② Inspection Type	□□ - □ (Inspection Type) (Inspecting Authorities)	◦ Inspection Type 01 Manufacturing inspection 02 Inspection during construction (Resident Officer) 03 Pre-operational (facility) inspection 04 Pre-operational (performance) inspection 05 Inservice inspection (Resident Officer) 06 Periodical inspection 07 Quality assurance inspection 08 Special inspection/ irregular inspection 09 Decommissioning facility inspection/ Decommissioning confirmation inspection ◦ Inspecting authorities refer to the institution that actually performs inspections. Code is the same as that of the issuing authorities in Control No.
③ Finding Type	□□ (Code)	◦ Type of Inspection Finding (Refer to Schedule 2.) 01 Inadequacy of equipment performance 02 Inadequacy of measures taken 03 Inadequacy of tests or inspections 04 Violation or inadequacy of procedures 05 Inadequacy of construction or maintenance 06 Inadequacy of design or manufacturing 07 Non-compliance with standard requirements 08 Others

<table>
<tr><th>Classification</th><th>Code</th><th>Contents</th></tr>
<tr>
<td>④ Inspected Institution</td>
<td><table><tr><td></td><td></td><td></td></tr></table>(Code) (Name of Inspected Institution)</td>
<td>
<table>
<tr><th>Institution Type</th><th>Code</th><th>Name of Inspected Institution</th></tr>
<tr><td>Nuclear power reactor</td><td>11</td><td>Korea Hydro & Nuclear Power Co., Ltd.</td></tr>
<tr><td>Research reactor</td><td>21</td><td>Korea Atomic Energy Research Institute (KAERI)</td></tr>
<tr><td>Reactor for education</td><td>31</td><td>Kyunghee University</td></tr>
<tr><td>Nuclear fuel cycle business</td><td>41
42

43

44

45</td><td>KAERI (refining facilities)
KAERI (conversion facilities)
KAERI (fabrication facilities)
KAERI (spent nuclear fuel processing facilities)
Korea Nuclear Fuel Co., Ltd. (fabrication facilities)</td></tr>
<tr><td>Nuclear power plant design</td><td>51
52</td><td>KAERI
Korea Power Engineering Company Inc.</td></tr>
<tr><td>Nuclear power equipment production</td><td>60</td><td>(Insert the name of the licensee)</td></tr>
<tr><td>Nuclear power plant construction</td><td>70</td><td>(Insert the name of the licensee)</td></tr>
<tr><td>Nuclear power plant maintenance</td><td>80</td><td>(Insert the name of the licensee)</td></tr>
<tr><td>Disposal business</td><td>91

92

93</td><td>Spent nuclear fuel storage facilities
Low and intermediate level radioactive waste disposal facilities
Radioisotope waste disposal facilities</td></tr>
<tr><td>Others</td><td>99</td><td></td></tr>
</table>
- As regards Institution Type, if there exists no relevant institution, the second digit shall be 9.
</td>
</tr>
</table>

Classification	Code	Contents
⑤ Reactor Type	| | | | | (Code)	◦ Applicable only to operators of nuclear reactor. ◦ Nuclear power reactor facilities (1000)

1100	Head office and all places of business
1200	Kori Power Plant Site
1201	Kori Unit 1
1202	Kori Unit 2
1203	Kori Unit 3
1204	Kori Unit 4
1300	Wolsong Power Plant Site
1301	Wolsong Unit 1
1302	Wolsong Unit 2
1303	Wolsong Unit 3
1304	Wolsong Unit 4
1400	Yonggwang Power Plant Site
1401	Yonggwang Unit 1
1402	Yonggwang Unit 2
1403	Yonggwang Unit 3
1404	Yonggwang Unit 4
1405	Yonggwang Unit 5
1406	Yonggwang Unit 6
1500	Uljin Power Plant Site
1501	Uljin Unit 1
1502	Uljin Unit 2
1503	Uljin Unit 3
1504	Uljin Unit 4
1505	Uljin Unit 5
1506	Uljin Unit 6

- Fill in the blank as follows if commonly applicable to two units of a subject facility
  e.g.) Kori Unit 1 and 2: 1212
  Yonggwang Unit 3 and 4: 1434
- For units added to a single site, an additional number shall be granted in sequence, based on the existing classification number.
- For nuclear power plants newly constructed and operated, a new number shall be granted.
  e.g.) New Wolsong 1600, New Kori 1700, New Uljin 1800, etc.

◦ Research reactor facilities (2000)

2101	Hanaro research reactor
2102	TRIGA Mark Ⅱ
2103	TRIGA Mark Ⅲ

◦ Education reactor facilities (3000)

3101	AGN-201 (Kyunghee University)

Classification	Code	Contents
⑥ Inspected System	(Code)	◦ Fill in the blank if the target of inspection is a nuclear reactor and related facilities: 01 Nuclear reactor (including nuclear fuel) 02 Reactor coolant system facilities 03 Instrumentation and control system facilities 04 Handling and storage facilities of nuclear fuel materials 05 Radioactive wastes disposal facilities 06 Radiation control facilities 07 Nuclear reactor containment facilities 08 Nuclear reactor safety system facilities 09 Safety-related electric power supply system facilities 10 Radiological emergency response facilities 18 Commonly applicable to not less than two systems 19 Other facilities related to safety of a nuclear reactor
⑧ Inspection Date	. . .	◦ Enter the date on which the Inspecting Person prepared Inspection Findings Form. Enter the last two digits for *(Year)* and enter respectively two digits for *(Month)* and *(Date)*.
⑨ Issue Date	. . .	◦ Enter the date on which the Issuing Authorities executed Inspection Findings Form, in writing, toward the Inspected Institution in the same manner as specified in the foregoing ⑧, i.e. how to enter the Inspection Date.
⑪ Requested Date	. . .	◦ Enter the date on which the Issuing Authorities executed Inspection Findings Form, in writing, toward the Inspected Institution in the same manner as specified in the foregoing ⑧, i.e. how to enter the Inspection Date.
⑫ Closing Date	. . .	◦ Enter the date on which the Issuing Authorities give a notice to the licensee of the closing of Inspection Findings according to Article 10 in the same manner as specified in the foregoing ⑧, i.e. how to enter the Inspection Date.

※ The blanks of ① Control No., ⑨ Issue Date, ⑪ Requested Date, and ⑫ Closing Date shall be filled in by the Issuing Authorities of the Inspection Findings Form.

[Schedule 2]

Types and Description of Inspection Findings

01 Inadequacy of equipment performance

When equipment or components fail to perform their intended function in their design due to under-performance and aging thereof.

02 Inadequacy of measures taken

When operational measures taken by the nuclear licensee and the results thereof fail to satisfy the regulatory requirements or technical standards.

03 Inadequacy of tests or inspections

When a periodic test of equipment and systems is not conducted, or a part thereof is omitted, or relevant records are omitted despite the requirement that such equipment and systems should be subject to a test periodically in accordance with test procedures.

04 Violation or inadequacy of procedures

When matters set forth in test, maintenance, operation, and work procedures related with operation and management of nuclear power plants are inconsistent with applicable technical standards or authorization/ permission requirements or when such procedures are unsatisfactory.

05 Inadequacy of construction or maintenance

When matters taken are not complete since construction has not been performed in accordance with a relevant design or maintenance has not been carried out in compliance with maintenance procedures.

06 Inadequacy of design or manufacturing

When the details and results of the design of structures, systems, components, and equipments fail to satisfy relevant regulatory requirements and technical standards or when purchase and manufacturing of materials, components, and equipment fail to meet the relevant regulatory requirements, design standards, and quality assurance requirements.

07 Non-compliance with standard requirements

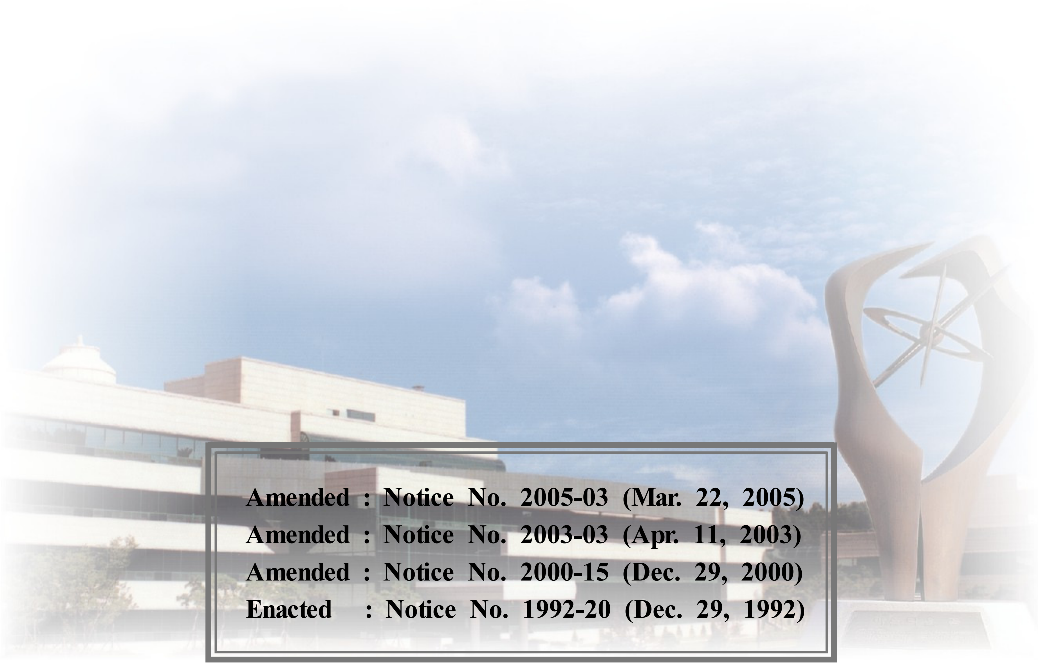
In the case of a failure to comply with a final safety analysis report, technical specifications for operation, domestic standards for permit/license, designation or foreign technical standards that apply *mutatis mutandis*.

08 Others

Matters other than the above types of Inspection Findings, i.e. unsatisfactory matters regarding document management, human factor, safety culture, and safety control.

【 07 】

Material Surveillance Criteria for Reactor Pressure Vessel



Amended : Notice No. 2005-03 (Mar. 22, 2005)
Amended : Notice No. 2003-03 (Apr. 11, 2003)
Amended : Notice No. 2000-15 (Dec. 29, 2000)
Enacted : Notice No. 1992-20 (Dec. 29, 1992)

◎ **Notice of the Minister of Science and Technology No.2005-03 (MOST.reactor.014)**

The Material Surveillance Criteria for Reactor Pressure Vessel as provided for in Subparagraph 7 of Article 102 (1) of the Enforcement Decree of the Atomic Energy Act and Article 21 (4) and Subparagraph 3 of Article 63 (1) of the Regulation on Technical Standards for Nuclear Reactor Facilities, etc. is hereby notified publicly as follows:

March 22, 2005
Minister of Science and Technology

Material Surveillance Criteria for Reactor Pressure Vessel

Article 1 (Purpose) The purpose of this notice is to prescribe requirements of material surveillance program for pressurized water reactor vessel as provided for in Subparagraph 7 of Article 102 (1) of the Enforcement Decree of the Atomic Energy Act and Article 21 (4) and Subparagraph 3 of Article 63 (1) of the Regulation on Technical Standards for Nuclear Reactor Facilities, etc.

Article 2 (Scope of Application) (1) This notice is applied to ferritic materials in the beltline of pressurized water reactor vessel.

(2) This notice is applied to all pressurized water reactor vessels for which the predicted maximum neutron fluence at the end of the design lifetime exceeds 1×10^{17} n/cm^2 (E > 1 MeV) at the inside surface of the reactor vessel.

Article 3 (Definitions) The definitions of terms used in this Notice are as follows:

1. The term "surveillance tests" means a series of tests to monitor changes in the mechanical properties of ferritic materials in the reactor vessel beltline region which result from exposure of these materials to neutron irradiation and the thermal environment;
2. The term "surveillance program" means all the plans to systematically perform the surveillance tests of Subparagraph 1;
3. The term "reference nil-ductility transition temperature (RT_{NDT})" means the reference temperature defined in accordance with the ASME Boiler and Pressure

Vessel Code Sec. III, NB-2330;

4. The term "transition temperature shift ($\triangle RT_{NDT}$)" means the difference in the 41 J index temperatures for the best fit Charpy impact test curve measured before and after irradiation;
5. The term "adjusted reference temperature (ART)" means the reference temperature adjusted for irradiation effects by adding to the initial RT_{NDT}, the transition temperature shift $\triangle RT_{NDT}$, and an appropriate margin as described in the U.S. NRC Regulatory Guide 1.99, Rev.2;
6. The term "upper-shelf energy" means the average energy value for all Charpy specimen tests (normally three) whose test temperature is above the onset of upper-shelf behavior. For specimens tested in sets of three at each test temperature, the set having the highest average may be regarded as defining the upper-shelf energy;
7. The term "transition region" means the region on the Charpy transition temperature curve in which toughness increases rapidly with rising temperature;
8. The term "lead factor" means the ratio of the neutron flux density at the location of the specimens in the surveillance capsule to the neutron flux density at the reactor pressure vessel inside surface at the peak fluence location;
9. The term "effective full-power years (EFPY)" means the accumulated total power divided by the power which can be achieved when the reactor is operated 24 hours a day for one year with the rated power given by the design;
10. The term "accelerated irradiation capsules" means test specimens inserted additionally on the location where neutron flux density is high;
11. The term "beltline" or "beltline region" means the irradiated region of the reactor vessel (shell material including welds, heat affected zones, and plates or forging) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage;
12. The term "heat-affected zone (HAZ)" means plate material or forging material extending outward from, but not including, the weld fusion zone in which the microstructure of the base metal has been altered by the heat of the welding process;
13. The term "surveillance capsule" means a sealed container with the inserted surveillance test specimens which is attached inside the reactor vessel; and

14. The term "pre-irradiation tests" means the tests performed with the unirradiated reactor vessel materials to acquire the baseline data of surveillance tests.

Article 4 (Test Specimens) (1) The installer or operator of nuclear power reactor shall prepare the following specimens in each surveillance capsule to perform surveillance tests for the beltline of the reactor vessel:

1. Test specimens to measure the changes in mechanical properties in the beltline of the reactor vessel;
2. Enough number of neutron dosimeters to determine neutron fluence in the inside of the reactor vessel; and
3. Enough number of temperature monitors to confirm the maximum temperature to which the test specimens inside a surveillance capsule are exposed.

(2) The test specimens shall consist of the base metal from one plate or forging used in fabricating the beltline of the reactor vessel, weld metal made with the same welding practice and lot of flux and by the same welding practice as that used for the selected beltline weld, and the heat-affected-zone associated with the base metal.

(3) To prepare for the cases when the shape or size of test specimens should be changed or accelerated irradiation specimens should be fabricated, archive materials which meet the following requirements shall be retained with full documentation and identification throughout the life of the reactor vessel:

1. Test stock sufficient to perform at least 2 additional surveillance tests shall be retained;
2. This test stock shall be in the form of full-thickness sections of the original materials; and
3. The fabrication history (including lot number, chemical composition, and heat treatment) and mechanical properties shall be recorded.

Article 5 (Types and Quantities of Specimens) Test specimens shall consist of Charpy V-notch impact specimens, tension specimens, and fracture toughness test specimens. And the minimum number of test specimens for each irradiation exposure set (capsule) shall be as follows:

Material \ Specimen Type	Charpy	Tension	Fracture Toughness
Base metal			
◦ parallel to major working direction	12	3	4
◦ normal to major working direction	12	3	4
Weld metal	12	3	4
Heat-affected-zone (HAZ)	12	-	-

Article 6 (Standard for Test Specimens) (1) The standard shape and size of specimens shall be as follows:

1. Charpy V-notch impact specimens corresponding to KS (Korean Industrial Standard) B 0809 "Test Pieces for Impact Test of Metallic Materials";
2. Tension specimens corresponding to KS (Korean Industrial Standard) B 0801 "Test Pieces for Tensile Test of Metallic Materials" or the specimens approved by the Minister of Science and Technology; and
3. Fracture toughness test specimens approved by the Minister of Science and Technology.

(2) Different types of specimens can be used if the standard specimens of Subparagraphs 1 and 2 of Paragraph 1 are not available. Provided, that it shall be demonstrated that they have the same test results as those obtained from using the standard specimens.

Article 7 (Specimen Orientation and Location) Test specimens shall be selected as follows:

1. The test specimens for base metal shall be removed from about the quarter-thickness (1/4T) locations. The specimen orientations parallel and normal to the major working direction shall be selected and the axis of the notch of the Charpy impact specimen for base metal shall be oriented perpendicular to the surface of the material;
2. Specimens representing weld metal may be removed at all locations throughout the thickness with the exception of locations within 12.7 mm (1/2 in.) of the root or surface of the weld as indicated in the Attached Figure; and
3. Charpy specimens representing the weld heat-affected-zone shall be removed from about the quarter-thickness (1/4T) locations and the notch roots in the HAZ Charpy specimens shall be at a standard distance of approximately 0.8 mm (1/32 in.) from the weld fusion line as the Attached Figure.

Article 8 (Encapsulation of Specimens and Location of Capsules) (1) Specimens shall be fixed not to move within the surveillance capsule, and then they shall be encapsulated.

(2) Specimens shall be maintained in an inert environment within a corrosion-resistant capsule.

(3) Surveillance capsules shall be located within the reactor vessel so that the specimen irradiation history duplicates neutron spectrum, temperature history, and maximum neutron fluence as closely as possible. It is recommended that the surveillance capsule lead factors be in the range of one to three.

Article 9 (Test Type and Method) (1) Surveillance tests shall be performed in such a manner that the changes and anticipated effects due to neutron irradiation can be compared and evaluated, and shall be tested in accordance with the following methods:

1. A full transition temperature curve shall be obtained by performing Charpy impact tests in accordance with KS B 0810 "Method of Impact Test for Metallic Materials";
2. Tension tests shall be performed in accordance with KS B 0802 "Method of Tensile Test for Metallic Materials" and KS D 0026 "Method of Elevated Temperature Tensile Test for Steels and Heat-Resisting Alloys" at a temperature in the vicinity of the upper end of the Charpy energy transition region, the mid-transition temperature, and the service temperature; and
3. Fracture toughness tests shall be performed in accordance with the method approved by the Minister of Science and Technology.

(2) Different test methods can be used if the tests can not be performed in accordance with the requirements of Paragraph 1 or are difficult to be carried out. Provided, that it shall be demonstrated that they have the same test results as those obtained from using the test methods given in Paragraph 1.

Article 10 (Pre-service Tests) (1) For the pre-service tests which is performed in unirradiated condition, the test material shall be selected in accordance with Article 4 (2), the types of test specimens shall be prepared in accordance with Article 6, and the specimen orientation and location shall meet the requirements of Article 7.

(2) The specimen type and material for pre-service tests shall follow the requirements of Article 5 and the number of specimens shall be enough to decide the material properties depending on temperature.

(3) The methods of pre-service tests shall be as follows:

1. The Charpy impact test method shall follow the requirements of Subparagraph 1 of Article 9 (1);
2. The tension tests method shall follow the requirements of Subparagraph 2 of Article 9 (1). Provided, that the test temperatures shall include room temperature, service temperature, and one intermediate temperature; and
3. Fracture toughness tests shall be performed in accordance with the method approved by the Minister of Science and Technology and the same temperatures as those of the tension tests shall be selected.

Article 11 (Submission of Surveillance Program) The installer of nuclear power reactors shall submit the surveillance program providing the following information as an attachment document of the Final Safety Analysis Report:

1. Fabrication history including material specification and heat treatment in the beltline of the reactor vessel;
2. Drawings showing the locations of reactor internals, the locations and structures of reactor core and surveillance capsules, etc;
3. Drawings indicting location(s) of weld metal and the peak vessel fluence;
4. Test type and method (In case of Article 9 (2), the explanation documents shall be attached.);
5. Specimen type and fabrication method (In case of Article 6 (2), the explanation documents shall be attached.);
6. Pre-service test results;
7. Withdrawal schedule; and
8. Evaluation method of test results.

Article 12 (Implementation of Surveillance Tests) (1) The number of capsules and the withdrawal sequence are based on the predicted transition temperature shift at the end of life as presented in the Attached Table. The first surveillance test should be performed using the capsule with the highest lead factor.

(2) All the standby capsules inserted in addition to the minimum number of capsules specified in the Attached Table shall be removed and stored at a refueling outage when the capsule receives the accumulated neutron fluence equivalent to 1.5 times of the peak end-of-life vessel fluence.

(3) After all the remaining capsules are removed, an adequate dosimetry program and a periodical inspection schedule shall be established to ensure that the exposure

conditions of the reactor vessel continue to be consistent with those used to project the effects of embrittlement to the end of design life.

(4) Any changes to the reactor vessel exposure conditions and their effects shall be reported to the Minister of Science and Technology prior to changing the plant licensing basis.

(5) The surveillance specimens withdrawn in accordance with Paragraph 1 shall be retained in a container so that they can be available for reconstitution to cope with reestablishment of surveillance programs, etc.

Article 13 (Record and Report, etc.) (1) The operator of nuclear power reactor shall make records of matters concerning surveillance program, test procedure, and test results and keep such records with the used surveillance specimens during the operating life of the reactor vessel.

(2) The following matters shall be reported in the document of Paragraph 1:

1. Surveillance program;
2. Test results; and
3. Evaluation of test results and measures taken.

(3) The operator of nuclear power reactor shall submit the report to the Minister of Science and Technology on the subject of Subparagraphs 2 and 3 of Paragraph 2 within one year from the date of capsule withdrawal.

Article 14 (Determination of Operating Condition) (1) The operator of nuclear power reactor shall confirm the appropriateness of the existing operating condition of the reactor vessel by using the adjusted reference temperature estimated from the surveillance tests, and determine the operating condition which can be used until the next surveillance test results are available and their corresponding operating condition is determined.

(2) The operating condition of Paragraph 1 shall be determined in accordance with the U.S. 10CFR Part50, Appendix G, IV "Fracture Toughness Requirements."

Article 15 (Surveillance Test Result Requirements) (1) The nuclear reactor shall not continue to be operated if the surveillance test results do not meet the requirements of the U.S. 10CFR Part 50, Appendix G, IV "Fracture Toughness Requirements."

(2) Regardless of the requirements of Paragraph 1, the nuclear reactor is allowed to be operated during the period when the following items are met, in case all the requirements below are demonstrated to be satisfied and the submitted documents

are approved by the Minister of Science and Technology:

1. When the integrity of reactor vessel is confirmed by performing a volumetric examination of 100% of the related region of reactor vessel which do not meet the fracture toughness requirements in Paragraph 1;
2. When safety margins are guaranteed from the results of supplemental fracture toughness tests; and
3. When the integrity and safety of the reactor vessel are demonstrated by performing the fracture mechanics analysis and safety evaluation with appropriate safety margins.

(3) The documents of Paragraph 2 shall be submitted to the Minister of Science and Technology at least three years prior to the date when the requirements of Paragraph 1 are predicted to be no longer satisfied.

(4) The reactor vessel whose Charpy upper-self energy at the end of life is evaluated to be less than 68 J or the adjusted reference temperature at the 1/4T position in the vessel wall at the end of life is calculated to be more than 93℃ (200˚F) should be designed to make possible a thermal annealing treatment.

(5) In case the fracture toughness requirements of Paragraph 1 or the requirements of Paragraph 2 are not satisfied, the reactor vessel beltline may be given to a thermal annealing treatment to recover the fracture toughness of the material. In this case, the thermal annealing plan shall be submitted to the Minister of Science and Technology at least three years prior to the date when the requirements of Paragraph 1 or Paragraph 2 are predicted to be no longer satisfied.

Addenda

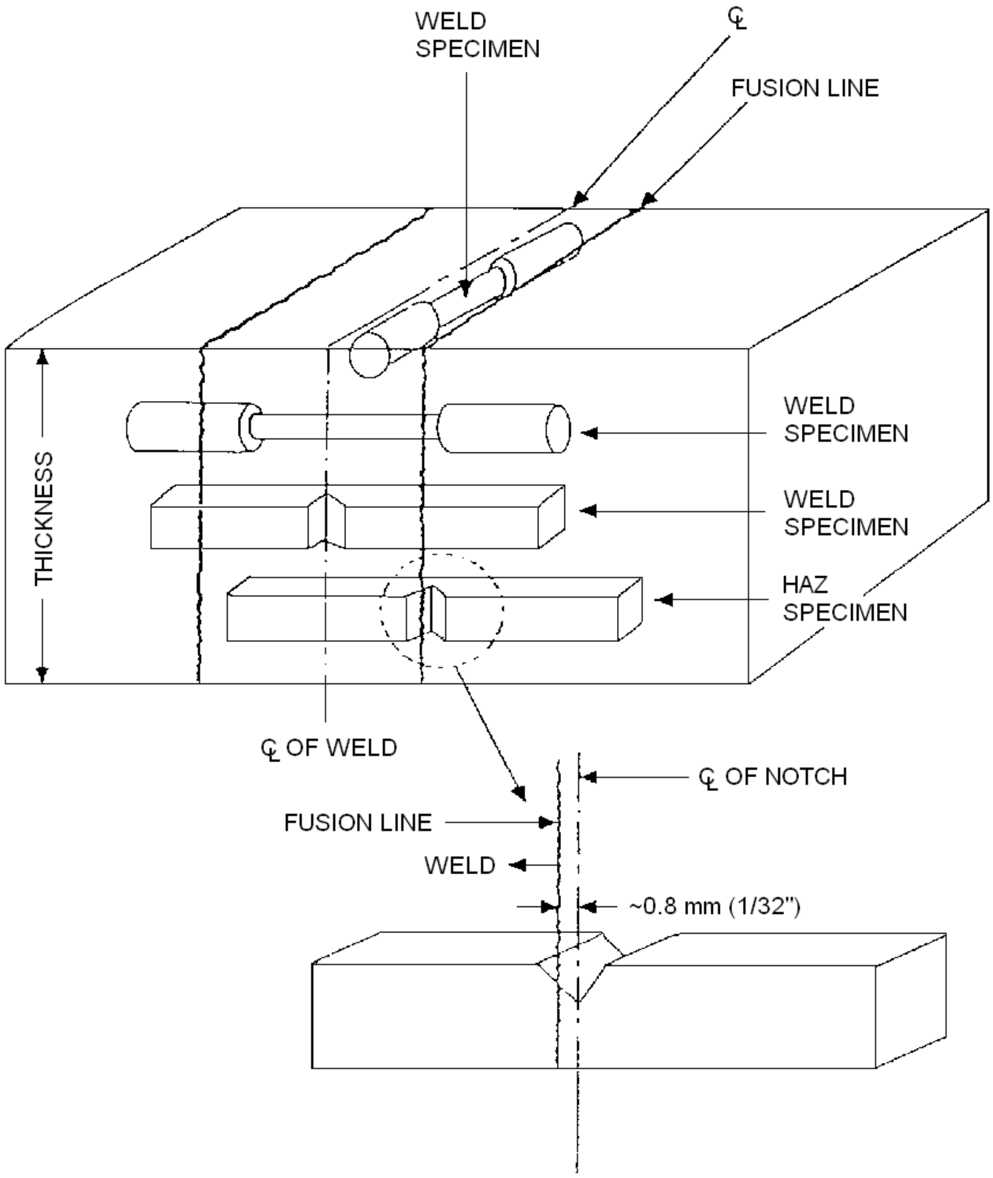
Article 1 (Enforcement Date) This notice shall enter into force on the date of its promulgation.

Article 2 (Interim Measures) The reactor vessel surveillance tests for a nuclear power facility whose construction permit or operating license of nuclear power reactor was issued before the enforcement of this Notice may be subject to the requirements having been effective at issuance time of construction permit or operating license. But the implementation of the surveillance tests shall follow the requirements of Article 12 and the record and report shall follow the requirements of Article 13.

Article 3 (Repeal of Notice) Notice of the MOST No.2003-03 "Material Surveillance Criteria for Reactor Vessel" is repealed at the time this Notice enters into force.

[Attached Figure]

Location of Test Specimens within Weld and HAZ Test Material



[Attached Table]

Minimum Recommended Number of Surveillance Capsules and Their Withdrawal Schedule

Items		Predicted Transition Temperature Shift at Vessel Inside Surface ($\triangle RT_{NDT}$)		
		$\triangle RT_{NDT} \leq 56℃$	$56℃ < \triangle RT_{NDT} \leq 111℃$	$\triangle RT_{NDT} > 111℃$
Minimum number of Capsules		3	4	5
Withdrawal Sequence	First	6[A]	3[A]	1.5[A]
	Second	15[B]	6[C]	3[D]
	Third	32[E]	15[B]	6[C]
	Fourth	-	32[E]	15[B]
	Fifth	-	-	32[E]

* The withdrawal schedule of this Table is given in terms of effective full-power years (EFPY) of the vessel with a design life of 32 EFPY. Surveillance tests can be performed at the nearest refueling outage.

A. Or at the time when the accumulated neutron fluence of the capsule exceeds 5×10^{18} n/cm^2, or at the time when the highest predicted $\triangle RT_{NDT}$ of all encapsulated materials is approximately 28℃, whichever comes first.

B. Or at the time when the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor vessel inner wall location, whichever comes first.

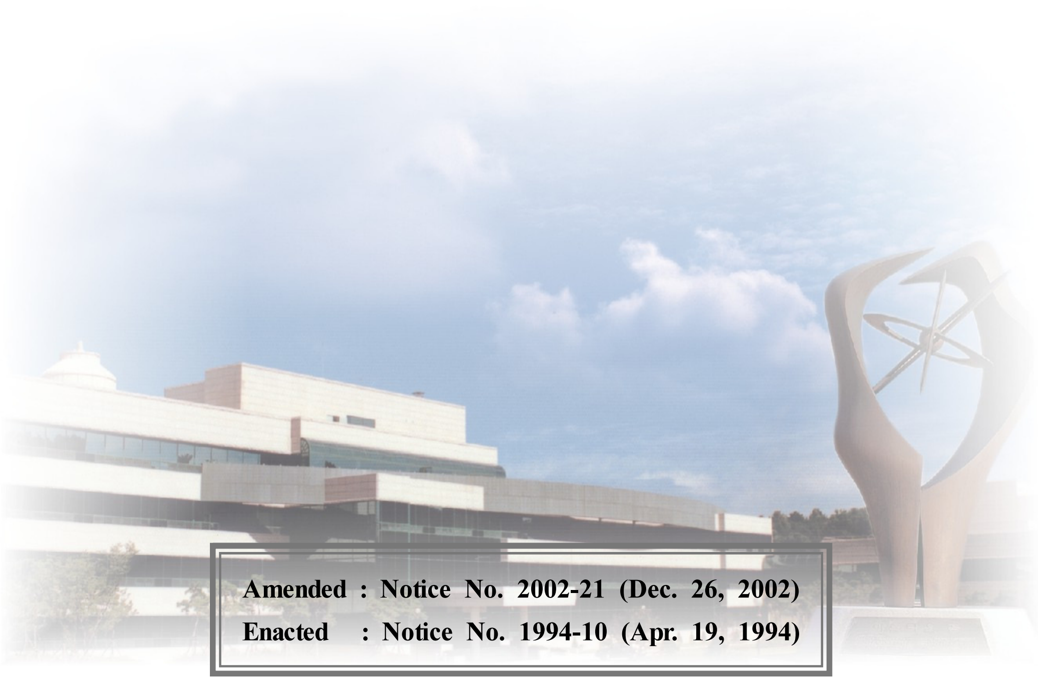
C. Or at the time when the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor vessel 1/4T location, whichever comes first.

D. Or at the time when the accumulated neutron fluence of the capsule corresponds to a value midway between that of the first and third capsules.

E. At the time when the capsule receives the accumulated neutron fluence equivalent to 1.5 times of the peak end-of-life vessel fluence. Based on the previous test results, the test using this capsule may be retained.

【 08 】

Regulation on Safety Classification and Applicable Codes and Standards for Nuclear Reactor Facilities



Amended : Notice No. 2002-21 (Dec. 26, 2002)
Enacted : Notice No. 1994-10 (Apr. 19, 1994)

◎ **Notice of the Minister of Science and Technology No.2002-21 (MOST.reactor.015)**

The regulation on safety classification for the structures, systems and components important to safety and the applicable codes and standards in accordance with Article 12 (1) of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc. is hereby notified publicly as follows:

December 26, 2002
Minister of Science and Technology

Regulation on Safety Classification and Applicable Codes and Standards for Nuclear Reactor Facilities

Article 1 (Purpose) The purpose of this notice is to prescribe the safety classification for the structures, systems and components important to safety and the applicable codes and standards in accordance with Article 12 (1) of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc.

Article 2 (Scope of Application) This notice shall be applied to the structures, systems and components important to safety in the pressurized water reactor. Provided, that CSA/CAN3-N285.0 and CSA/CAN3-N285.1 are applied to those in the pressurized heavy water reactor.

Article 3 (Definitions) Definitions of the terms used in this notice shall be as follows:

1. The term "safety class" means classified safety of structures, systems, or components of nuclear reactor facilities based on their nuclear safety function. It is classified as Safety Class 1 (SC-1), 2 (SC-2) or 3 (SC-3);
2. The term "non-nuclear safety (NNS)" means classified safety of structures, systems, or components of nuclear reactor facilities that are not in Safety Class 1, 2 or 3;
3. The term "facility" means structures, systems, or components which have one or more functions;
4. The term "safety function" means any function that is necessary to ensure: (a) the integrity of the reactor coolant pressure boundary, (b) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (c) the capability to

prevent or mitigate the consequences of plant conditions that could result in potential off-site exposures defined in the "Technical Standards for the Locations, Structures and Installations of Nuclear Reactor Facilities";

5. The term "applicable codes and standards" means requirements which are applied to materials, design, manufacture, construction, inspection, and test of equipment in accordance with the safety class assigned; and
6. The term "interface boundary" means the connected interface portions when equipment assigned as Safety Class 1, 2, 3 or NNS are connected to any equipment assigned as other Safety Class.

Article 4 (General Requirements) General requirements for Safety Class shall be as follows:

1. Safety Class 1, 2, 3 or NNS shall be assigned to all the equipment of nuclear reactor facilities in accordance with this Notice;
2. Where more than one system is capable of accomplishing a nuclear safety function and one of the systems, on its own, satisfies all the nuclear safety-related requirements (e.g. redundancy, diversity, and capacity), the designer may classify the latter to the corresponding Safety Class and the others as NNS;
3. The designer may impose more stringent design requirements than those imposed upon one facility corresponding to the applicable class. If this option is chosen, the designer shall retain the original class designation;
4. The designer may optionally provide two or more pieces of equipment to separately meet multiple environmental qualification requirements of SC-2 or SC-3 equipment for events, where one equipment need not meet such separated qualification requirements simultaneously;
5. In case the function of an equipment is degraded due to failure of the support, the support shall be classified to a class corresponding to the function of the equipment, and in case the failed support does not affect the function of the equipment, the support shall be classified to a class corresponding to its own function;
6. The interface boundary connected to the equipments which have different Safety Class shall be assigned to the more stringent Safety Class corresponding to functions of the connected equipment. In case, however, the result of analyses of interface boundary interaction confirms its appropriateness, the interface boundary may be assigned to the lower Safety Class;
7. If the failure of Safety Class or NNS equipment connected to other Safety Class

equipment could prevent the latter equipment from accomplishing its nuclear safety function, an interface barrier or isolation device shall be provided to protect the latter equipment; and
8. Where a single item of equipment, or a portion thereof, provides two or more functions of different classes, it shall be classified to the more stringent class.

Article 5 (Safety Class 1) Safety Class 1 (SC-l) shall be assigned to pressure-retaining portions and supports of mechanical equipment that form part of the reactor coolant pressure boundary (RCPB) whose failure could cause a loss of reactor coolant exceeding the reactor coolant normal makeup capability.

Article 6 (Safety Class 2) Safety Class 2 (SC-2) shall be assigned to pressure-retaining portions and supports of primary containment, and to pressure-retaining portions and supports that are not included in SC-1 of Article 5 and are designed to accomplish the following nuclear safety functions that:
1. provide fission product barrier or primary containment radioactive material holdup or isolation;
2. provide emergency heat removal for the primary containment atmosphere to an intermediate heat sink, or emergency removal of radioactive material from the primary containment atmosphere (e.g. containment spray);
3. introduce emergency negative reactivity to make the reactor subcritical (e.g. boron injection system), or restrict the addition of positive reactivity via pressure boundary equipment;
4. ensure emergency core cooling where the equipment provides coolant directly to the core (e.g. residual heat removal and emergency core cooling); or
5. provide or maintain sufficient reactor coolant inventory for emergency core cooling (e.g. refueling water storage tank).

Article 7 (Safety Class 3) Safety Class 3 (SC-3) shall be assigned to equipment, not included in SC-1 or SC-2, that is designed to accomplish the following nuclear safety functions that:
1. ensure hydrogen control of the primary containment atmosphere to acceptable limits, except for primary containment boundary extension function;
2. remove radioactive material from the atmosphere of confined spaces outside primary containment. (e.g. control room or fuel building) containing SC-1, SC-2, or SC-3 equipment;

3. introduce negative reactivity to achieve or maintain subcritical reactor conditions (e.g. boron makeup);
4. provide or maintain sufficient reactor coolant inventory for core cooling (e.g. reactor coolant normal makeup system);
5. maintain geometry within the reactor to ensure core reactivity control or core cooling capability (e.g. core support structures);
6. structurally load-bear or protect SC-l, SC-2, or SC-3 equipment (This applies to concrete or steel structures that are not within the scope of the KEPIC MN (nuclear mechanical) in accordance with Notice of MOST No.2005-04, and its corresponding code (ASME Boiler and Pressure Vessel Code Section III);
7. provide radiation shielding for the control room or off-site personnel;
8. ensure required cooling for liquid-cooled stored fuel (e.g. spent fuel storage pool and cooling system);
9. ensure nuclear safety functions provided by SC-1, SC-2, or SC-3 equipment (e.g. provide heat removal for SC-l, SC-2, or SC-3 heat exchangers, provide lubrication of SC-2 or SC-3 pumps, provide fuel oil to the emergency diesel engine);
10. provide actuation or motive power for SC-1, SC-2, or SC-3 equipment;
11. provide information or controls to ensure capability for manual or automatic actuation of nuclear safety functions required of SC-1, SC-2, or SC-3 equipment;
12. supply or process signals or supply power required for SC-l, SC-2, or SC-3 equipment to perform their required nuclear safety functions;
13. provide a manual or automatic interlock function to ensure or maintain proper performance of nuclear safety functions required of SC-l, SC-2, or SC-3 equipment;
14. provide an acceptable environment for SC-1, SC-2, or SC-3 equipment and operating personnel; or
15. provide for functions defined in SC-2 where equipment, or portions thereof, is not within the scope of the code for design and manufacture of pressure vessel (KEPIC MN(nuclear mechanical) in accordance with the Notice of MOST No.2005-04, and its corresponding code (ASME Boiler and Pressure Vessel Code Section III).

Article 8 (Non-Nuclear Safety) Non-nuclear safety (NNS) shall be assigned to equipment not included in SC-1, SC-2, or SC-3 prescribed in Articles 5, 6 and 7 and is designed to accomplish the following functions that:

1. process, extract, encase, or store radioactive waste;
2. provide cleanup of radioactive material from the reactor coolant system or the fuel storage cooling system for normal operations;
3. extract radioactive waste from, store, or transport for reuse irradiated neutron absorbing materials (e.g. boron compounds);
4. monitor radioactive effluents to ensure that release rates or total releases are within limits established for normal operations and transient events;
5. resist failure that could prevent any SC-l, SC-2, or SC-3 equipment from performing its nuclear safety function;
6. structurally load-bear or protect NNS equipment providing any of the NNS functions;
7. provide permanent shielding for proof SC-l, SC-2, or SC-3 equipment or of onsite personnel;
8. provide operational, maintenance or post-accident recovery functions involving radioactive materials without undue risk to the health and safety of the public;
9. provide an acceptable environment for SC-1, SC-2, or SC-3 equipment required to achieve or maintain a safe shutdown condition, following a control room evacuation in emergency;
10. handle spent fuel, the failure of which could result in fuel damage such that significant quantities of radioactive material could be released from the fuel;
11. ensure reactivity control of stored fuel;
12. protect SC-2 or SC-3 equipment necessary to attain or maintain safe shutdown following fire; or
13. monitor variables to:
    a. verify that plant operating conditions are within technical specification limits (e.g. refueling water storage tank level, safety-related cooling water temperature);
    b. indicate the status of protection system bypasses that are not automatically removed as a part of the protection system operation;
    c. indicate status of SC-l, SC-2, or SC-3 equipment; or
    d. aid in determining the cause or consequences of events for post-accident survey.

Article 9 (Applicable Codes and Standards) The equipment assigned as Safety Class in accordance with Articles 5, 6 and 7 shall correspond to the following codes and standards. Provided, that in case some portion of each of the following codes and standards is to be applied, it shall be reviewed and evaluated technically by the Minister of Science and Technology.

1. Safety Class 1, 2, or 3 mechanical equipment : the KEPIC MN (nuclear mechanical) in accordance with Notice of the MOST No.2005-04, and its corresponding code (ASME Boiler and Pressure Vessel Code Section III)
2. Safety Class 3 electrical equipment : the KEPIC EN (nuclear electrical) in accordance with Notice of the MOST No.2005-04, and its corresponding code (IEEE 279, 308 and 603)
3. Safety Class 2, or 3 structures : the KEPIC SN (nuclear structure), MN (nuclear mechanical) in accordance with Notice of the MOST No.2005-04, and its corresponding code (ASME Section III, ACI 349 and ANSI/AISC N-690)

Addenda

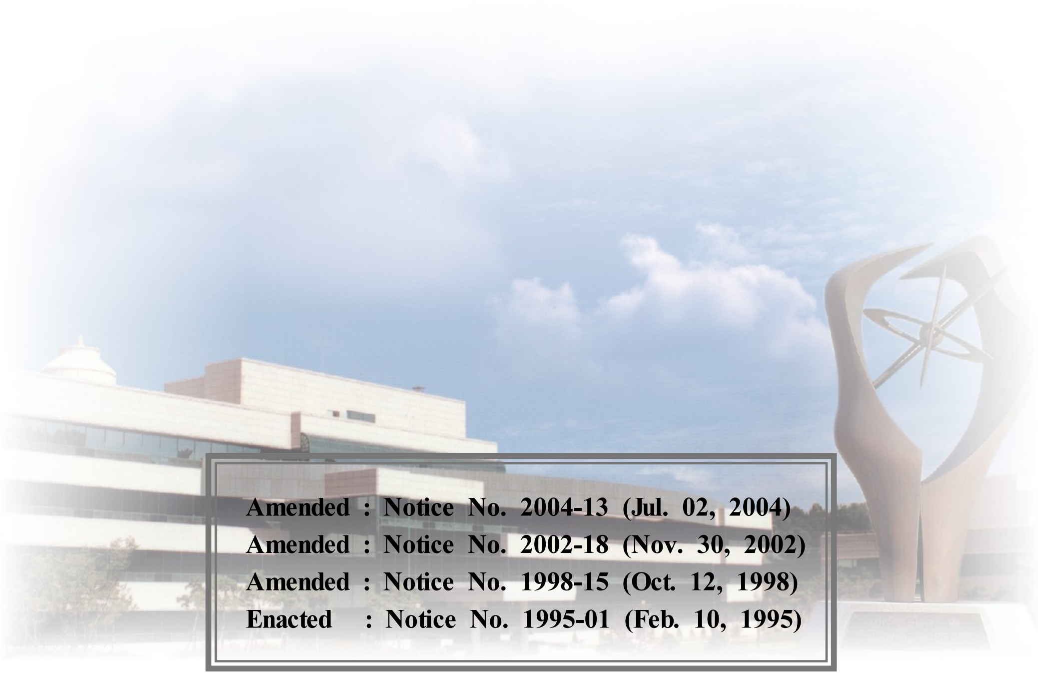
Article 1 (Enforcement Date) This notice shall enter into force on the date of its promulgation.

Article 2 (Repeal of Notice) Notice of the MOST No.1994-10 "Regulation on Safety Classification and Applicable Codes and Standards for Nuclear Reactor Facilities" is repealed at the time this notice enters into force.

Notice of the Minister of Science and Technology No.2004-13 (MOST.reactor.016)

【 09 】

Regulation on In-Service Inspection of Nuclear Reactor Facilities



Amended : Notice No. 2004-13 (Jul. 02, 2004)
Amended : Notice No. 2002-18 (Nov. 30, 2002)
Amended : Notice No. 1998-15 (Oct. 12, 1998)
Enacted : Notice No. 1995-01 (Feb. 10, 1995)

◎ **Notice of the Minister of Science and Technology No.2004-13 (MOST.reactor.016)**

The regulation on in-service inspection of the safety-related facilities as provided for in Article 63 (1) 1 and (2) of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc. is hereby notified publicly as follows:

July 2, 2004
Minister of Science and Technology

Regulation on In-Service Inspection of Nuclear Reactor Facilities

Article 1 (Purpose) The purpose of this notice is to prescribe the requirements of the in-service inspection to monitor and assess the degradation of performance and materials of safety-related facilities due to aging in accordance with Article 63 of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc.

Article 2 (Scope of Application) This notice shall be applied to the safety-related facilities classified in accordance with the "Regulation on Safety Classification and Applicable Codes and Standards for Nuclear Reactor Facilities" under Article 12 of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc.

Article 3 (Definitions) The definitions of terms used in this notice shall be as follows:

1. The term "in-service inspection (ISI)" means the activities related to non-destructive testing, pressure testing, repair and replacement, assessment of unanticipated operating events, etc. of safety related facilities performed by the operator of nuclear power reactor (hereinafter referred to as "operator") in order to monitor and assess the degradation of performance and materials of safety related facilities due to aging during the life time. It contains pre-service inspection (PSI) performed prior to operation of nuclear power plant;
2. The term "safety-related facility" means structures, systems and components classified as safety related items and containing the safety functions such as the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents that could result in potential

off-site exposures;

3. The term “entrusted institution” means the institution performing the activities entrusted in accordance with Article 111 (1) of the Atomic Energy Act; and
4. The term “repair and replacement" means welding, brazing, metal removal of and removing, adding, or modifying of items or systems of safety-related facility.

Article 4 (Inspection Intervals and Initial Date) (1) The operator shall perform in-service inspection every 10 years. Provided, that in case of pressurized heavy water reactor, the initial inspection interval shall be within 5 years and successive inspection intervals shall be 10 years.

(2) Initial date of the first in-service inspection interval shall correspond to commercial operating date.

Article 5 (Inspection Baseline Data) The installer of nuclear power reactor (hereinafter referred to as "installer") shall perform pre-service inspection to obtain the baseline data for evaluation and assessment of aging degradation, and provide those to the operator.

Article 6 (Inspection Standards) (1) The following standards and guidelines shall apply to the inspection:

1. To the pressurized water reactor, MI (In-Service Inspection) of KEPIC (Korea Electric Power Industry Code) defined in "Guideline for Application of the Korea Electric Power Industry Code as the Technical Standards of Nuclear Reactor Facilities" or equivalent standard (ASME Code Sec. XI) shall apply. Provided, that volumetric examination in addition to surface examination shall be performed to the butt welds of class 1 piping not less than 2 inch nominal size unisolable to reactor coolant pressure boundary during normal operation.
2. To the pressurized heavy water reactor, “CAN/CSA-N285.4, Periodic Inspection of CANDU Nuclear Power Plant Components” and “CAN/CSA-N285.5, Periodic Inspection of CANDU Nuclear Power Plant Containment Components” shall apply. Provided, that volumetric examination and surface examination at the level equivalent to Subparagraph 1 shall be performed to the butt welds of class 1 piping not less than 2 inch nominal size unisolable to reactor coolant pressure boundary during normal operation.
3. US-NRC Regulatory Guides 1.14, 1.65, 1.83, 1.147 and 1.150 shall apply to the followings:

(a) Reactor coolant pump fly wheel integrity;

(b) Materials and inspections for reactor vessel closure studs;

(c) In-service inspection of pressurized water reactor steam generator tubes. Provided, that the Implementation Guideline for Steam-Generator Tube Inspection in the Appendix of this notice shall apply in preference to the above Regulatory Guide;

(d) Inservice Inspection Code Case Acceptability, ASME Section Xi, Division 1; and

(e) Ultrasonic testing of reactor vessel welds during pre-service and in-service examinations.

4. US-NRC Regulatory Guides 1.35 and 1.90, CAN/CSA-N287.7, and RCC-G Part 3 shall apply to the inspection of tendons and structures of containment building.

(2) Effective code cut-off date of the standards applicable for each in-service inspection interval defined in Article 4 shall be as follows:

1. To the pressurized water reactor, effective code cut-off date of KEPIC MI or equivalent standard (ASME Code Sec. XI) designated 1 year prior to commencement of the inspection interval shall apply;
2. To the pressurized heavy water reactor, the latest version of CSA/CAN-N285.4, N285.5 and N287.7 issued 1 year prior to commencement of the inspection interval shall apply;
3. In case the effective code cut-off date of the other standards referred to standard for in-service inspection is not defined, the date shall be equal to that of the standard for in-service inspection; and
4. To the pre-service inspection, the effective standards in Subparagraphs 1 and 2 shall apply in principle. Provided, that if it is difficult to apply the effective cut-off date due to design condition, it is allowable to use the code cut-off date applied for construction permit.

Article 7 (Application of Augmented Inspection) The Minister of Science and Technology (hereinafter referred to as "Minister") may require the operator to reflect the operating experience and lessons learned from accidents at domestic and foreign nuclear power plants.

Article 8 (Submission of Inspection Program) (1) The operator shall prepare the long term in-service inspection program for every 10 years and submit it to the Minister 3 months prior to commencement date of the 10 years interval. Provided, that the installer shall submit the pre-service inspection program to the Minister 3 months prior to commencement date of the inspection.

(2) The following items shall be included in the long term in-service inspection program:

1. Inspection frequency and schedule
2. The following items for the inspection standards:
   (a) Standards used in the in-service inspection, and the effective date;
   (b) List of modification and limitation to the standards;
   (c) Alternative and relief requests (including ASME Code Case); and
   (d) Other items related to inspection standards.
3. The following items related to non-destructive examination (NDE):
   (a) Matters for inspection of structures, systems and components, such as selected items, scope, and details of NDE, in accordance with standards by safety class;
   (b) Piping & Instrument drawings (P&ID) identifying the scope of NDE;
   (c) Drawings identifying components and area to be examined or tested; and
   (d) Matters for calibration blocks.
4. The following items related to system pressure testing:
   (a) Name of system subject to system pressure testing;
   (b) Drawings identifying pressure boundary; and
   (c) Frequency and method for system pressure testing .
5. Inspection standards, components subject to inspection, inspection frequency, methods and contents for snubber
6. Matters related to repair and replacement of safety related facilities
7. Matters related to assessment of un-anticipated events during operation
8. List of examination and test procedures, and its related reference documents
9. Other matters related to in-service inspection

(3) In case of changing the long term in-service inspection program, the operator shall submit the changed program to the Minister together with the reason for changes prior to commencement of inspection concerned.

(4) The operator shall prepare the inspection program for each outage and submit it to the head of the entrusted institution prior to the inspection. In-service inspection program shall include the items to be performed during the outage concerned among those of Paragraph 2. If the in-service inspection program for the outage is different from the long term in-service inspection program, the operator shall describe the reason thereof.

(5) The Minister may require complement of in-service inspection program if the program does not meet requirements of Articles 6 and 7.

Article 9 (Report for Inspection Results) (1) The operator shall submit to the head of the entrusted institution the in-service inspection report including the following items within 3 months after completion of the inspection for the outage concerned:

1. Inspection and evaluation results for each component and system;
2. The details of non-conformances to the inspection standards of Article 6 and corrective actions thereof;
3. Details, reasons, and measures to be taken for uncompleted portion of inspection, if any; and
4. Inspection items to be added at the next outage, if any, and reasons thereof.

(2) After completion of in-service inspection for each outage provided for in Paragraph 1, the operator shall submit the summary report of the corrective actions for the indications exceeding acceptance criteria to the head of the entrusted institution, considering review period, prior to criticality of reactor.

(3) The installer shall submit the summary report of the pre-service inspection containing items of Paragraph 1 to the head of the entrusted institution prior to the issue date of operating license.

(4) The installer or operator shall submit to the Minister the integrated report within 6 months after completion of pre-service or long term in-service inspection.

Article 10 (Repair and Replacement) (1) The installer or operator shall submit the repair or replacement program containing the followings to the head of the entrusted institution when he intends to remove defects in the safety related facilities, or replace, modify or newly install a part or whole of such facilities:

1. Applicable edition and addenda of standards for repair or replacement;
2. Description of repair or replacement activities (including P&ID);
3. Matters related to work procedure for repair or replacement, qualification for workers, welding procedure and qualification, heat treatment, NDE, pressure test, etc.; and
4. Other technical matters related to repair or replacement.

(2) The requirements of Paragraph 1 are applicable after structures, systems and components have met all requirements of construction codes and standards.

(3) The technical standards for repair or replacement shall correspond with the codes and standards used for construction or those identified in the long-term

in-service inspection program. Provided, that if the requirements of the technical standards need to be changed or can not be met, approval of the Minister shall be obtained in accordance with Article 13.

Article 11 (Performance Demonstration of Non-Destructive Examination) (1) Among NDE methods performed during in-service inspection, the installer or operator shall conduct the performance demonstration of ultrasonic testing (UT) for the safety related facilities and eddy current testing (ECT) for steam generator tubes.

(2) The technical standards of the UT performance demonstration for the PWR and PHWR shall apply to Article 6 (1) 1 and 2, and cut-off date of the technical standards shall comply with Article 6 (2).

(3) The test specimen for UT performance demonstration shall be selected among those representing characteristics of size, materials, configuration, etc. of safety related facilities of domestic PWR and PHWR. In case of ECT for steam generator tubes, the performance demonstration shall be supplemented for each nuclear power plant by using data of characteristics of the plant

(4) The operator shall submit the report containing the followings to the Minister 3 months prior to application of performance demonstration. When the operator needs to change the performance demonstration requirements or to use a foreign performance demonstration system, he shall submit the report 3 months prior to the application:

1. Name and address of the organization carrying out performance demonstration;
2. Applicable technical standards for performance demonstration and their edition;
3. Test specimen or test data of performance demonstration;
4. Evaluation of performance demonstration;
5. Quality assurance of performance demonstration;
6. Security related to performance demonstration;
7. Operation of performance demonstration system; and
8. Other related items.

(5) The Minister may require supplementation if he deems the report of Paragraph 4 unsatisfactory to the requirements.

(6) The organization carrying out performance demonstration shall periodically report the status of the demonstration to the Minister. The Minister may audit the performance demonstration activities if necessary.

Article 12 (Certification) (1) Welding and NDE shall be performed by the accredited personnel in accordance with inspection standards defined in Article 6.

(2) Personnel qualified in accordance with performance demonstration shall be re-qualified every 5 years. Provided, that re-qualification for UT performance demonstration is required only once.

(3) Practical training for maintaining the qualification required in accordance with limitations of Table 2 in "Guideline for Application of the Korea Electric Power Industry Code as the Technical Standards of Nuclear Reactor Facilities" shall be performed at the organization of performance demonstration.

Article 13 (Alternative Application or Relief Request) (1) The operator may apply the other standards or guidelines equivalent to those defined in the Notice or utilize the new methods or procedures appropriate for the in-service inspection (alternative application), and also, he may request relief or exemption from any part of the standards (relief request), in one of the following cases not reducing reliability of nuclear reactor facilities and integrity of related facilities:

1. In case that the inspection can neither be conducted nor meet the standards in Article 6 due to characteristics of design, configuration and materials of the components or systems subject to examination or test;
2. In case that the workers are anticipated to be over-exposed to high radiation in spite of considerable protective action taken;
3. In case that alternative application or relief request does not reduce appropriate quality and safety; or
4. In case that quality and safety improvement to be achieved by the inspection according to the standards of Article 6 do not compensate for the difficulty and risk of the working activity.

(2) For the alternative application or relief request of Paragraph 1, the operator shall submit to the Minister the application containing the followings for approval:

1. Name of plant;
2. Unit, and name of components and systems subject to alternative application or relief request, and document number;
3. Standards and guidelines for the alternative application or relief request;
4. The reasons and contents for alternative application or relief request;
5. Technical justifications for alternative application or relief request; and
6. Other related matters.

(3) The alternative application or relief request shall be effective only during the long-term in-service inspection interval applied to each reactor. The application may

be submitted together with the inspection program or its revision.

(4) The P&I drawings and repair history of the component and system shall be attached to the alternative application program of Paragraph 2.

(5) In case of application of ASME Code Cases, the alternative application shall be submitted to the Minister for approval in accordance with Paragraphs 1, 2 and 3. Alternative application related to ASME Code Case may be submitted together with long-term in-service inspection program or separately.

Addenda

Article 1 (Enforcement Date) This notice shall enter into force on the date of its promulgation. Provided, that performance demonstrations of UT for piping welds and bolts and ECT for steam generator tubes shall apply from July 1, 2004. Provided, that performance demonstration of UT for reactor vessel of PWRs shall enter into force on the date of July 1, 2005; and performance demonstration of UT for nozzle welds and inner radius sections of reactor vessel, dissimilar metal welds of piping, and structural overlaid welds of austenitic piping shall enter into force on the date of July 1, 2006.

Article 2 (Transitional Measures for Certification of the Foreign Performance Demonstration) In application of Article 12 (1), US EPRI certificate of UT performance demonstration shall be effective for 1 year from the enforcement date of this notice, and US EPRI certificate of ECT performance demonstration shall be effective during its certified period.

Article 3 (Repeal of Notice) The provisions related to in-service inspection in Notice of the MOST No.2002-18 "Regulation for In-service Inspection and In-service Testing of Nuclear Reactor Facilities" are repealed at the time this notice enters into force.

[Appendix]

Implementation Guideline for Steam-Generator Tube Inspection (related to Article 6 (1) 3 c)

1. Minimum requirements of inspection[1]

Reactor type (Unit No.) / Category		WH, Framatome (Kori 2,3,4, Yonggwang 1,2, Uljin 1,2) KSNP (Yonggwang 3,4,5,6 and Uljin 3,4,5,6)	WH (Kori 1)	FW, B&WI (Wolsong 1,2,3,4)
Tube materials		Alloy 600 (TT & HTMA) Alloy 690 TT (Explosive Expansion)	Alloy 690 TT	Alloy 800M
Bobbin		100 %	100 %	50 %
MRPC	H/L	100 %[2)]	≥20 %	≥10 %
	C/L	20 %[2)]	≥10 %	-
Profilometry		100% (H/L and C/L) - first inspection and tubes in which crack was detected	- tubes subject to bobbin test	- tubes subject to bobbin test
The 3rd party evaluation		100%	100%	100%

※ 1) It may be modified according to the results of the previous in-service inspection after completion of one cycle of inspection for all the plants.
2) Up to 5 inch above the top of tube-sheet for KSNP, Up to 3 inch above the top of tube-sheet for the other type of plant.

2. Plugging criteria for the worn tubes

Unit / Category	Yonggwang 3,4,5,6 and Uljin 3,4,5,6 (KSNP)		The other plants (WH, Framatome, CANDU)
	Vicinity of Stay Cylinder	The other area	
plugging criteria (flaw depth)	≥ 30%	≥ 36%	≥ 40%

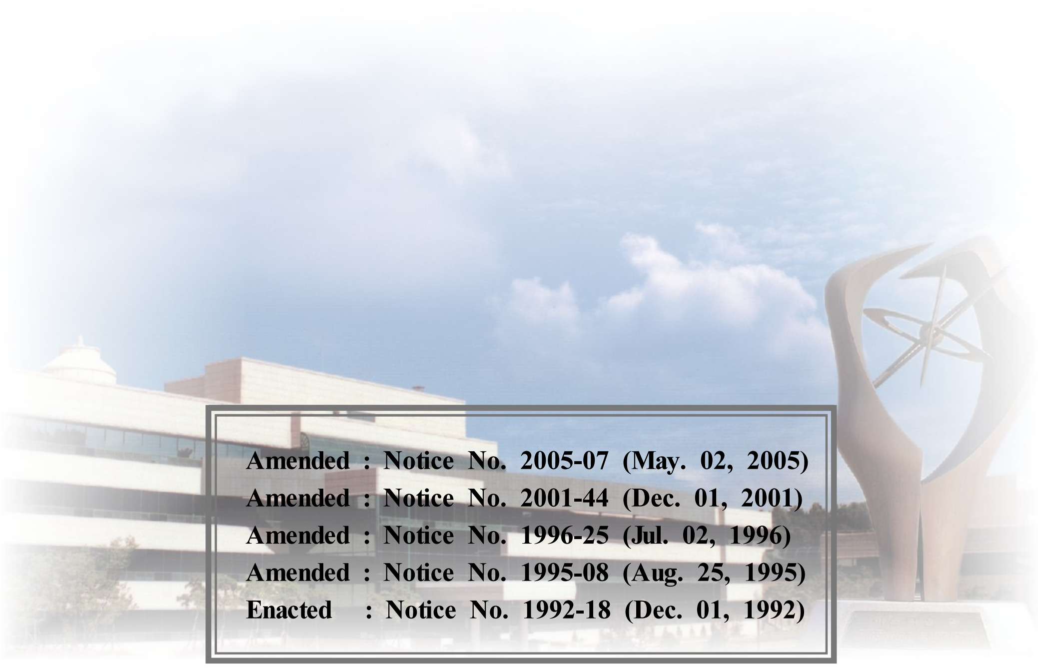
3. Collection of records during tube manufacturing stage

○ The following inspection data shall be obtained in advance for reference data of ISI and PSI:

- ECT results of each tube;
- Visual examination results of tube assembling, welding, and inside examination of tubes; and
- NDE for the expansion area of each tube, and inspection result of cleaning after tube expansion.

【 10 】

Regulation on Reporting and Public Announcement of Accidents and Incidents for Nuclear Power Utilization Facilities



Amended : Notice No. 2005-07 (May. 02, 2005)
Amended : Notice No. 2001-44 (Dec. 01, 2001)
Amended : Notice No. 1996-25 (Jul. 02, 1996)
Amended : Notice No. 1995-08 (Aug. 25, 1995)
Enacted : Notice No. 1992-18 (Dec. 01, 1992)

◎ **Notice of the Minister of Science and Technology No.2005-07 (MOST.reactor.019)**

The Regulation on Reporting and Public Announcement of Accidents and Incidents for Nuclear Power Utilization Facilities as provided in the Atomic Energy Act is hereby revised and promulgated as follows:

May 2, 2005
Minister of Science and Technology

Regulation on Reporting and Public Announcement of Accidents and Incidents for Nuclear Power Utilization Facilities

Chapter 1 General Provisions

Article 1 (Purpose) The purpose of this Notice is to prescribe matters regarding items to report and make open, procedures according to Articles 89, 98, 102 of the Atomic Energy Act (hereinafter referred to as "Act"), and the Technical Specifications for Operation submitted by nuclear licensees (hereinafter referred to as "licensees") according to Articles 21 (2) and 33 (2) of the Act, and evaluations thereof.

Article 2 (Scope of Application) This Notice shall be applied to items to be reported to the government by the licensees when accidents or incidents (hereinafter referred to as "events") occur during operation of nuclear power utilization facilities (hereinafter referred to as "facilities").

Article 3 (Definitions) (1) Definitions of the terms used in this Notice shall be as follows:

1. The term "accidents" means the events, of level 4 or higher according to the Event Rating Classification Criteria of this Notice, resulting in radiological hazards to human body, significant damage to the facilities, or radiological contamination to the environment;
2. The term "incidents" means events, of level 3 or below according to the Event Rating Classification Criteria of this Notice, which do not result in radiological hazards to human body, significant damage to the facilities, or radiological

contamination to the environment;

3. The term "reactor shut down" means a entry into sub-criticality of the reactor from power operation above the zero (0) power;
4. The term "leakage" means uncontrolled release of radioactive material inside or outside a facility, excluding release by a pre-planned procedure;
5. The term "release" means all leakage and discharge of liquid or airborne radioactive material; and
6. The term "working day" means the time from 09:00 to 18:00 of a day, except legal holidays, off-Saturdays and Sundays.

(2) Definitions of the terms used in this notice shall be the same as those in the Atomic Energy Act, the Enforcement Decree thereof, the Enforcement Regulation thereof, the Regulations on Technical Standards for Nuclear Reactor Facilities, etc. and the Regulations on Technical Standards for Radiation Safety Control, etc. except otherwise specified in Paragraph 1.

Chapter 2 Items to Report

Article 4 (Items to report) Cases that a licensee shall report according to this Notice shall be as follows, and details thereof are on the Tables:

1. Accidents occurred as prescribed in Articles 89 and 102 of the Act;
2. Situations in which safety measures need to be taken as prescribed in Article 98 (1) of the Act; and
3. Violations of safety limits, limits for safety systems settings or limiting condition for operation of the requirements of Technical Specifications for Operation specified in Article 16 (1) of the Enforcement Regulation of the Act.

Chapter 3 Methods and Procedures of Report

Article 5 (Report) (1) Licensee, when events addressed in Article 4 occur, shall make a Verbal Report to the Minister of Science and Technology using available communication means within times specified in the Table and shall submit an Initial Written Report on the Attached Form 1 within the next working day.

(2) In case of a verbal report as per Paragraph 1, the report shall be made either to the relevant desk during work-time or to the night duty desk during off-time. The nuclear power reactor licensee shall report to the resident officer of the Ministry of Science and Technology on the site.

(3) The written report as per Paragraph 1 shall include materials informed to the press.
(4) When the safety state of the facility concerned deteriorates after the time of report of Paragraph 1, follow-up reports shall be made.
(5) Licensee shall submit a Detailed Report to the Minister within times specified in the Table on the Attached Form 2.

Article 6 (Report on Equipment Failure) (1) The nuclear power reactor licensee or research reactor licensee shall submit a Report on Equipment Failure on the Attached Form 3 to the Minister within 30 days when equipment failure specified in the Table occurred.
(2) Equipment failure in Paragraph 1 shall not applied to the report in Article 5 (1), the provisional rating in Article 8 and internet publicity in Article 11.

Article 7 (Notification) When, among accidents specified in Subparagraph 1 of Article 4, theft or loss of radioactive material occurs, or emergency evacuation of neighboring inhabitant is needed due to release of radioactive material, licensees shall notify the police authorities in charge of the region without delay thereof.

Chapter 4 Rating of Events

Article 8 (Provisional Rating) (1) In case rating is needed in some of the events to report as specified in the Table, licensee shall make a provisional scale rating according to Guideline for Evaluation of Event Scale in Appendix 2 (hereinafter referred to as "Provisional rating") and fill out the Attached Form 1.
(2) Licensees shall add the Attached Form 4 in case the provisional rating is Class 2 or higher.

Article 9 (Comprehensive Evaluation) (1) The President of the Korea Institute of Nuclear Safety (KINS) shall hold the Event Scale Evaluation Committee (hereinafter referred to as "Evaluation Committee") quarterly, evaluate and confirm the final scale, and submit the results to the Minister of Science and Technology within 15 days after the next quarter begins.
(2) The President of the KINS shall release the results of event scale evaluation in Paragraph 1.

Article 10 (Composition of Evaluation Committee) (1) The Evaluation Committee in Article 9 is composed of the staff of the KINS and relevant experts, and operated under the direction of the President of the KINS.

(2) The President of the KINS shall report to the Minister of Science and Technology within 30 days when Committee members are appointed or changed.

Chapter 5 Event Information Release

Article 11 (Information Release to the Public) (1) Licensees shall release via internet within the following working day, the content reported to the Minister of Science and Technology according to this Notice.

(2) Licensee (Minister of Science and Technology as needed) shall inform the press within the following working day after the occurrence of event, of the following cases among event to report in the Table:

1. when the result of provisional event rating is Level 1 or higher;
2. when release, fire or other accidents occur during the transportation or packaging of radioactive materials, etc. in Article 89 of the Act;
3. when theft, loss, fire or other accidents occur to radiation generating devices or radioactive materials, etc. in Article 102 of the Act;
4. when power reduction for safe operation of facility occurs due to external effect such as natural disasters, etc. in Article 98 of the Act;
5. when emergency measures are necessary to protect the personnel and residents in relation to events of the facility and radiological hazards in Article 98 of the Act;
6. when leakage of radioactive materials inward or outward facility occurs and exceeds the limits specified in the Table;
7. when any human being is killed in relation to the operation of the facility; and
8. In other cases when the Minister of Science and Technology or licensee deems necessary.

(3) In case there is any change in the content released to the press according to Paragraph 2, licensee shall report to the Minister of Science and Technology and inform the press thereof.

Article 12 (Information Release Abroad) (1) In each of the following cases, the Minister of Science and Technology shall send the content of Event Rating according to the Attached Form 4 to the International Atomic Energy Agency:

1. the result of Provisional Rating in Article 8 is Level 2 or higher;
2. cases in which the IAEA is interested; or

3. when the Minister of Science and Technology deems necessary.

(2) In case the result of the comprehensive evaluation is class 2 or higher, the Minister of Science and Technology may send a report according to the "Incident Reporting System of the IAEA (IAEA-IRS)" to the IAEA.

Chapter 6 Others

Article 13 (Submission and Collection of Additional Data and Information) (1) The Minister of Science and Technology may request the submission of additional necessary information or data according to Article 103 of the Act when the report of licensee is insufficient.

(2) The President of the KINS may perform field investigation when it is deemed necessary for comprehensive evaluation specified in Article 9.

Article 14 (Change of Report) Licensee shall report to the Minister of Science and Technology in writing in case there is any change in the previous reports already submitted according to Articles 5, 6 and 8.

Article 15 (Submission of Copy) Licensee shall submit a copy of the report to the President of the KINS in case he reports in writing to the Minister of Science and Technology according to Articles 5, 6, 8 and 14.

Addenda

Article 1 (Enforcement Date) This Notice shall enter into force on the date of its promulgation.

Article 2 (Repeal of Notice, etc.) Notice of the MOST No.2001-44 "Regulations on Report in case of Accident and Incident in Nuclear-related Facilities" and guideline of the MOST "Information Disclosure Guidelines on Nuclear Power Plant Accident and Incident" are repealed at the time this notice enters into force.

Article 3 (Measures to Follow) Nuclear reactor licensees shall amend relevant documents such as Technical Specifications for Operation in Articles 21 and 33 in the Act according to this notice and submit them to the Minister of Science and Technology within 6 months after its enforcement.

[Table]

Events to Report (related to Article 4)

1. Events to report commonly applied to all nuclear power utilization facilities

Events to Report	Report Deadline		Rating of Events
	Verbal report (within X hours)	Detailed report (within X days)	
1. Occurrence of release, fire or other accidents during transport or packaging of radioactive materials in Article 89 of the Act	1 hour	7 days	○
2. Occurrence of theft, loss, fire or other accidents of radiation generating devices or radioactive materials in Article 102 of the Act	1 hour	7 days	○
3. Declaration of emergency according to radiological emergency plan(*)	-	-	○
4. Surface contamination exceeds the followings at areas other than radiation area due to leakage of radioactive materials, etc. (for power reactor and research reactor, radiation controlled area and preservation area shall be regarded as radiation areas) a. 40 kBq/m^2($10^{-4}\mu Ci/cm^2$) for radioactive material emitting α ray b. 400 kBq/m^2($10^{-3}\mu Ci/cm^2$) for radioactive material not emitting α ray	1 hour	30 days	○
5. Occurrence of incidents which cause either of the followings or would cause due to leakage of radioactive materials or by radiation generating device: a. abnormal increase of radiation level lasting more than 1 hour exceeding 0.1 Sv of total effective dose rate when an individual stays for 24 hours b. when radiation density by which an individual staying for 24 hours intakes 5 times and more of annual intake limit lasts more than 1 hour (this rule is not applied to a place where personnel is not manned during normal operation such as hot cell or reactor containment building)	1 hour	30 days	○
6. Occurrences of the followings where radioactive material is released into environment a. release of liquid or air-borne radioactive material is released into environment at places other than authorized discharge b. unplanned and uncontrolled release of radioactive material into environment	4 hours	30 days	○

(*) : In case of radiological emergency, report shall be done in accordance with the Act on Physical Protection and Radiological Emergency at Nuclear Facilities, and Notice of the MOST No.2004-11(Standards for Establishment, etc. of the Radiological Emergency Plan for Nuclear Licensee).

Incidents to Report	Report Deadline		Rating of Events
	Verbal report (within X hours)	Detailed report (within X days)	
7. Occurrence of casualty of human as a result of the operation of the facilities. In the case of injury, it is limited only after severe injury after transport to hospital is confirmed	4 hours	30 days	-
8. In case that emergency measures were taken due to existing or possible threat to safe operation of facility caused by events including fire or toxic gas in facility	4 hours	30 days	-
9. Release of radioactive material as follows: a. one-hour average of air concentration at facility boundary (same as Exclusion Area Boundary when the EAB is established) exceeds discharge control limit during air discharge as set forth in the attached list No.3 the 5th column and No.4 the 4th column respectively the Ministerial Regulation "Standard on Radiation Protection" b. one-hour average of water concentration at facility boundary (same as Exclusion Area Boundary when the EAB is established) exceeds discharge control limit during air discharge as set forth in the attached list No.3 the 8th column and No.4 the 6th column respectively the Ministerial Regulation "Standard on Radiation Protection"(Exclude tritium and dissoluble rare gas.)	8 hours	30 days	○
10. Occurrence of loss of control for radioactive material which was a cause or possible cause to the following: a. abnormal increase of radiation level lasting for more than 1 hour when an individual staying for 24 hours can be exposed at radiation exceeding 50 mSv of total effective dose b. lasting for more than 1 hour of radiation level when an individual staying for 24 hours can intake radiation exceeding yearly intake limit due to radiation leakage inside or outside of radiation control area (This rule is not applied to a place where personnel is not manned during normal operation such as hot cell or reactor containment building.)	8 hours	30 days	○
11. In case emergency measures were taken due to existing or possible threat to safe operation of facility caused by natural disasters including wild fires, storm, tsunami and typhoons	8 hours	30 days	-
12. In case emergency measures were taken due to existing or possible threat to safe operation of facility caused by industrial disasters in the vicinity of facility	8 hours	30 days	-

2. Events to report commonly applied to power and research reactor facilities

Events to Report	Report Deadline		Rating of Events
	Verbal report (within X hours)	Detailed report (within X days)	
1. Violations of safety limits, and of action requirements of limits for safety systems settings of technical specifications for operation	1 hour	30 days	○
2. Violations of action requirements of limiting condition for operation of technical specifications for operation	4 hours	30 days	○
3. Disactuation of emergency core cooling system in spite that the system should have actuated during any transient following failure, malfunction or human error	4 hours	30 days	○
4. Occurrence of earthquake (apply only to actuation or possible actuation of earthquake monitor at plant, excludes the monitor malfunction)	4 hours	30 days	-
5. Incidents resulting in failures of primary parameter displaying group (safety parameter display system, etc.) or loss of emergency notification system in the main control room	8 hours	30 days	○
6. One of the following cases: a. Failure of safety related structures, systems or components of the spent fuel storage cask b. Severe degradation of spent fuel storage cask vessels	8 hours	30 days	○

3. Events to report applied to power reactor facilities

Events to Report	Report Deadline		Rating of Events
	Verbal report (within X hours)	Detailed report (within X days)	
1. Reactor trips(automatic or manual) due to failures of facilities or human error. Failures of facilities include propagated impact from off-site power system.	4 hours	30 days	○
2. Actuation of the following Engineered Safety Features (ESF) during the operational mode of Technical Specification applicable(includes malfunction). This excludes the planned actuation for test and routine surveillance. a. Emergency Core Cooling System (ECCS), containment isolation (excludes actuation of CVI system resulting from radiation level increase due to fuel replacement or planned airborne discharge from primary system of pressurized heavy water reactor), containment spray, auxiliary (emergency) feed water system b. Supply of power by automatic actuation of emergency diesel generator due to the low-low voltage signal of safety related bus	4 hours	30 days	○
3. Reduction of power in relation to the failures to meet the limiting conditions for operation of technical specification for operation	4 hours	-	-
4. Unanticipated release of radioactive materials including the following: a. Actual or possible actuation of high radiation signal of radiation monitor. This excludes airborne tritium concentration monitor at facilities using heavy water. b. Release of heavy water to outside the facility, used for cooling or moderation, the amount of which is more than 100 kg for 24 hours.	4 hours	30 days	○
5. Penetration defects or leakage detected at the pressure boundary of reactor coolant system	8 hours	30 days	○
6. The following component failures, a. Inoperability of one train of high or low pressure ECCS for 72 hours b. Inoperability of one train of recirculation system for 72 hours c. Inoperability of one train of containment spray system for 72 hours d. Inoperability of one train of auxiliary feed water system for 72 hours e. Inoperability of one train of safety related alternating or direct current power system for 72 hours	-	Component failure report in 30 days	-

f. Inoperability of containment isolation for more than 4 hours(8 hours for PHWR) due to loss of isolation function during test or operation g. Inoperability of more than 1 seismic monitor for more than 30 days h. Inoperability of more than 1 meteorological monitor channel(wind shift, speed, standard deviation of wind shift) for more than 7 days i. Inoperability or failures of more than 1 sensor or sensor train, spray or sprinkler system, fire hydrant, yard fire hydrant source for more than 14 days j. Inoperability of more than 1 pump, fire hydrant supply system, seismic category 1 fire hydrant system, fire resistant equipment(fire door, fire wall, penetration, etc.) for more than 7 days k. Failures of emergency diesel generator as follows: however, it does not include actuation failure of operator error, trip by the bypass trip signal on the emergency operation procedure, failures of components not used on the emergency operation procedure, and components failures not belonging to design category. 1) when start-up or speed-up was not accomplished automatically and manually 2) when prescribed voltage and frequency were not reached in a set time period 3) when the normal load was not undertaken			

4. Events to report applied to research reactor facilities

Events to Report	Report Deadline		Rating of Events
	Verbal report (within X hours)	Detailed report (within X days)	
1. Actuation (automatic or manual) of reactor protection system due to failures of facilities or human error. This excludes reactor trip by instant black-out of off-site power	4 hours	30 days	○
2. Actuation of Engineered Safety Features during the operational mode of Technical Specification applicable (includes malfunction). This excludes the planned actuation for test and routine surveillance.	4 hours	30 days	○
3. Unanticipated release of radioactive materials including the followings, a. Actual or possible actuation of high radiation signal of radiation monitor due to leakage. This excludes airborne tritium concentration monitor at facilities using heavy water. b. Release of heavy water to outside the facility, used for cooling or moderation, the amount of which is more than 50 kg for 24 hours.	4 hours	30 days	○
4. Component failure related items. a. Inoperability of more than 1 shutdown rod b. Inoperability of one train of safety related alternating or direct current power system for 1 hour c. Inoperability of more than 1 seismic monitor for more than 30 days d. Inoperability of all meteorological monitor channels for 1 check item or more (out of wind shift, speed, standard deviation of wind shift, etc.) for more than 7 days e. Inoperability of more than 1 sensor or sensor train, fire spray or sprinkler system, fire hydrant, yard fire hydrant water source for more than 14 days	-	Component failure report in 30 days	-

5. Events to report applied to other facilities excluding reactor

Events to Report	Report Deadline		Rating of Events
	Verbal report (within X hours)	Detailed report (within X days)	
1. That reached or may have reached criticality the nuclear raw or nuclear fuel material during refinement, conversion, fabrication, etc. This applies only to nuclear fuel cycle enterpriser.	1 hour	30 days	○
2. When incidents belonging to one of the followings to materials licensed under the AEA Article 65. a. Unplanned radiological contamination belonging to one of the followings. 1) Cases that radiation workers or civilians are limited to enter for 24 hours or that additional radiological control has to be taken. 2) Case that radiation concentration to exceed 5 times yearly intake limit lasts for more than 1 hour for an individual who stays there for 24 hours b. Failure or inoperability of radiation equipment belonging to one of the followings, 1) Radiation equipment installed to prevent release of radioactive material exceeding the regulatory limit or radiation exposure, and to mitigate effects of accident. 2) Equipment requested to be operable immediately upon failure or inoperability c. Fire or explosion belonging tp one of the following caused damage to licensed material, installations, storage vessel or equipment containing radioactive material. 1) Case that radiation concentration to exceed 5 times yearly intake limit lasts for more than 1 hour for an individual who stays there for 24 hours 2) Occurrence of damage affecting integrity of licensed material or storage vessel of it. d. Incidents belonging to one of the followings. 1) Sticking of source material outside of shielding cask 2) Damage of source material rack 3) Occurrence of safety problem of source material due to failure of source material cable or driving mechanism 4) Inoperability of exit-inlet control system 5) Detection of source material at outlet monitor of product 6) Structural damage of water tank or wall part, or abnormal reduction or leakage of storage water	8 hours	30 days	○

[Attached Form 1]

Initial Written Report on Event in Nuclear Power Utilization Facility

(related to Article 5 (1))

<table>
<tr><th colspan="14">Initial Written Report on Event in Nuclear Power Utilization Facility</th></tr>
<tr><td colspan="3">Report Series No.</td><td colspan="11"></td></tr>
<tr><td colspan="3">Name of facility</td><td colspan="11"></td></tr>
<tr><td colspan="3">Owner of facility</td><td colspan="11"></td></tr>
<tr><td colspan="3">Address of facility</td><td colspan="11"></td></tr>
<tr><td colspan="3">Related provisions</td><td colspan="11"></td></tr>
<tr><td colspan="3">Occurrence date</td><td colspan="11">yy mm dd hh mm</td></tr>
<tr><td colspan="3" rowspan="2">Report date</td><td colspan="4">Initial verbal report</td><td colspan="7">yy mm dd hh mm</td></tr>
<tr><td colspan="4">Initial written report</td><td colspan="7">yy mm dd</td></tr>
<tr><td colspan="3">Name, title, organization and telephone. no. of reporter</td><td colspan="11"></td></tr>
<tr><td colspan="3">Title of event</td><td colspan="11"></td></tr>
<tr><td rowspan="2">Provisional rating</td><td>none</td><td>0</td><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>6</td><td>7</td><td colspan="4">Evaluation criteria</td></tr>
<tr><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td colspan="4">Affects to defence in depth function ☐
Affects to on-site ☐
Affects to off-site ☐</td></tr>
<tr><td colspan="14"><Content of Event>
1. Outline of event

2. Facility status and progress of event

3. Basis of provisional rating

</td></tr>
<tr><td colspan="3">Safety affects to facility</td><td colspan="4">Y ☐ N ☐</td><td colspan="3">Radiological effect</td><td colspan="4">Y ☐ N ☐</td></tr>
</table>

[Attached Form 2]

Detailed Report on Event in Nuclear Power Utilization Facility (Cover)

(related to Article 5 (5))

Detailed Report on Event in Nuclear Power Utilization Facility (Cover)			
Report Series No.			
Name of facility			
Owner of facility			
Address of facility			
Related provisions			
Occurrence date	yy mm dd hh mm		
Report date	yy mm dd		
Name, title, organization and telephone no. of reporter			
Title of event			
<Content of event> I. Outline of event			
Safety affects to facility	Y ☐ N ☐	Radiological effect	Y ☐ N ☐

[Attached Form 2]

Detailed Report on Event in Nuclear Power Utilization Facility (cont'd)

Detailed Report on event in Nuclear Power Utilization Facility (cont'd)	
Report Series No.	

<Content of Event>

II. Detailed content of Event

III. Safety evaluation

IV. Root causes and corrective action

V. Lessons learned

[Attached Form 3]

Report on Equipment Failure in Nuclear Reactor and Related Facilities

(related to Article 6)

<table>
<tr><th colspan="4">Report on Equipment Failure in Nuclear Reactor and Related Facilities</th></tr>
<tr><td>Report Series No.</td><td colspan="3">Write serial no. of Report on Equipment Failures made in the relevant year and relevant Facility
(ex. : 05-Uljin Unit 7-3)</td></tr>
<tr><td>Name of facility</td><td></td><td>operation mode and power</td><td></td></tr>
<tr><td>Date of occurrence of failure (date of finding)</td><td>yy mm dd hh mm</td><td>report date</td><td>yy mm dd</td></tr>
<tr><td>Name, title, organization and telephone no. of reporter</td><td colspan="3"></td></tr>
<tr><td>Place of failure
(equipment and system)</td><td colspan="3"></td></tr>
<tr><td>Manufacturer of equipment</td><td colspan="3"></td></tr>
<tr><td>Title of failure</td><td colspan="3"></td></tr>
<tr><td colspan="4">1. content of failure

2. cause of failure

3. failure effects and safety evaluation

4. corrective action</td></tr>
</table>

[Attached Form 4] (related to Article 12 (1))

<table>
<tr><th colspan="14">THE INTERNATIONAL NUCLEAR EVENT SCALE (INES)</th></tr>
<tr><td colspan="14">EVENT RATING FORM (ERF)
to be sent to : IAEA, WAGRAMERSTRASSE 5, P.O.BOX 100, A-1400 VIENNA, AUSTRIA TELEX:1-12645, CABLE:INATOM VIENNA, FACSIMILE:43 1 2309723, TELEPHONE:43 1 2360 2685</td></tr>
<tr><td rowspan="2">EVENT TITLE</td><td colspan="12" rowspan="2"></td><td>EVENT DATE</td></tr>
<tr><td></td></tr>
<tr><td rowspan="3">RATING</td><td rowspan="2">RATING DATE</td><td rowspan="2">BELOW SCALE</td><td colspan="8">ON SCALE</td><td rowspan="3">SAFETY ATTRIBUTE</td><td colspan="2" rowspan="3">DEGR. DEFENCE ☐
IN-DEPTH
ON-SITE IMPACT ☐
OFF-SITE IMPACT☐</td></tr>
<tr><td>0</td><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>6</td><td>7</td></tr>
<tr><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>COUNTRY</td><td colspan="2"></td><td colspan="2">FACILITY NAME</td><td colspan="5"></td><td colspan="2">FACILITY TYPE</td><td colspan="2"></td></tr>
</table>

ASPECTS OF SIGNIFICANCE TO THE PUBLIC :

ACCIDENT ☐ INCIDENT ☐ DEVIATION ☐

	YES	NO
RADIOACTIVE RELEASES OFF-SITE	☐	☐
RADIOACTIVE RELEASES ON-SITE	☐	☐
WORKERS INJURED BY RADIATION	☐	☐
WORKERS INJURED PHYSICALLY	☐	☐
PLANT SAFETY IS UNDER CONTROL	☐	☐
THE EVENT REPORTED IS A DISCOVERY OF A DEFICIENCY BY ROUTINE SURVEILLANCE	☐	☐
A PRESS RELEASE WAS MADE (IF YES, PLEASE ATTACH IT)	☐	☐

SHORT DESCRIPTION OF THE EVENT : *

CONTACT PERSON FOR FURTHER INFORMATION	NAME : ADDRESS : Ministry of Science & Technology, 1 Joongang-Dong, Gwacheon City, KOREA PHONE : 82 - 2 - 2110 - 3669 FAX : 82 - 2 - 2110 - 3629

* PLEASE ATTACH ADDITIONAL INFORMATION ON JUSTIFICATION OF THE EVENT RATING AND DIFFICULTIES ENCOUNTERED, IF NEEDED

[Appendix 1]

Preparation Guideline of Event Report

1. Preparation guideline of initial event report of nuclear power utilization facility (related to Article 5 (1))

A. General

1) Initial written event report be reported via fax or e-mail using the attached form 1
2) 1 page enough as possible, additional pages available as needed
3) Content of event be described in essence in brief

B. Series No. of report

• Write series number of reported event by applicable year, facility and unit number, and -0

- example : in case of initial written event report of third report event occurred in 2005 at Uljin nuclear power plant unit 5, write as [05-ULJIN unit5-3-0]

C. Facility name, owner and address

• Write facility name, name of the organization owning the facility and address that the facility is located

D. Applicable provisions in this rule

• Identify all applicable provisions in this rule to event

- example : 1.4.a or 3.2.b of table

E. Occurrence (Finding) date, initial verbal report date, initial writing report date

Write date by minute and write present date of report writing for initial writing report date

F. Reporter : write head of department (director class) in charge of concerned facility

G. Title of event

1) Include system or component that affects most the incident
2) Write outcomes resulting from event such as reactor trips

H. Provisional rating of event scale

1) Write the rating results and mark applicable column out of 3 evaluation criteria according to Appendix 2 "Guideline for Rating of Event Scale"

2) Write "non-applicable" for cases that are not subject to rating

I. Content of event

1) Outline of event

a. Write operational status of facility prior to occurrence of event

b. Write estimated causes, etc. of event by 6 detail principle

c. Describe detailed occurrences by time sequence as determined at the point of report and write one by one when various events were

2) Facility status and progress of event

Describe the results from the event determined as of report time, safety status and progress of the event in brief

3) Basis of tentative classification

Describe applicable items used as basis of tentative rating in 2 of Appendix 2 "Guideline for Rating of Event Scale"

2. Preparation guideline of detailed event report of nuclear power utilization facility (related to Article 5 (5))

A. Report series number

- Write Series number of report events by applicable year, facility and unit number, and -1
  - Example : in case of detailed event report of third report event occurred in 2005 at Ulzin nuclear power plant unit 5, write [05-ULZIN5-3-1] and increase end number for subsequent report

B. Facility name, owner and address

- Write facility name, name of the organization owning the facility and address that the facility is located

C. Applicable provisions in this Regulation

- Identify all applicable provisions in this rule to events
  - example : 1.4.a or 3.2.b of table

D. Occurrence(Finding) date, initial verbal report date, initial writing report date

- Write date by minute and write present date of report writing for initial writing report date

E. Report date : Write the date of report

F. Reporter : Write head of department(director class) in charge of concerned facility

G. Title of event

1) Include system or component that affects most the event and root cause of the event

2) Write each title for multiple events. For the case when another event with same cause occurs within the period of report, include the event in one report

H. Content of event

1) Describe outline of event in prescriptive form in brief for the followings:

a. Operation status of facility prior to occurrence of event(include test and maintenance work in progress);

b. Inoperability of structures, system, component and human error, etc. at the initiating time of the event or contributed to the occurrence of the event;

c. Failure causes of systems or components, or human errors;

d. Failure mode and mechanism, and affects of failed equipment; and

e. Result of the event such as release of radioactive material(fact, form and amount of release).

2) Describe the followings throughly in detailed event report:

a. Operation status of facility prior to occurrence of event(operation mode, test and maintenance work, and operation stales of equipment);

b. Progress of initiation and development of the event(time and method of discovery of cause and problem, progress by time);

c. Enterpriser's reactions; and

d. Progress of recovery.

3) In safety evaluation, the followings shall be described except statements that were included in 2. and 4:

a. Investigation progress and result thereof to determine the causes;

b. Safety evaluation on event progress and result including measures taken;

c. Conclusion on evaluation; and

d. Severity evaluation in terms of safety.

4) Root cause and corrective measures includes the followings:

a. Root causes confirmed as a result of investigation; and

b. Short and long-term corrective measures to prevent recurrence of the event. The measures include measures already taken or under planning(also include measures for similar cases)

5) Lessons from the event include the followings:

a. Items needed to reflect to other units for recurrence of similar events; and

b. Existence of similar events experienced at similar or same facility at home and abroad.

I. Others

1) event report related to radiation hazard includes the followings:

a. Level of individual radiation exposure(anticipated or actual);

b. Causes of abnormal increase of concentration of radioactive material or radiation level;

c. Exposure record for each individual(including name, certificate number of resident, anticipated dose); and

d. Corrective measures taken or in progress to prevent recurrence including schedule to observe the safety requirements such as regulation limit or conditions, ALARA requirements and environmental standards.

2) Report related to theft or loss of radioactive material shall include the followings:

a. Kinds, amount and physical and chemical form of radioactive material;

b. Outline of situation of theft or loss;

c. Emergency measures related to licensed material;

d. Measures that were taken or will be taken for retrieval of radioactive material; and

e. Means or procedures adopted or under review for adoption to prevent recurrence.

[Appendix 2]

Guideline for Rating of Event Scale

(related to Articles 8 and 9)

1. Rating of Event Scale is as follows:

Type	Class	Nature	Typical Example
Accident	7	External release of a large fraction of the radioactive material giving radiological damage into a large area more than 1 country	Chernobyl NPP, USSR, 1986
	6	External release of the radioactive material giving radiological damage in such that would be likely to result in full implementation of radiological emergency plan	
	5	External release of the radioactive material giving radiological damage in such that would be likely to result in partial implementation of radiological emergency plan	Three Mile Island NPP, USA, 1979
	4	External release of radioactivity resulting in a dose at yearly limit to non radiation workers and needed off-site protective action such as food control	Saint-Laurent NPP, France, 1980
Incident	3	Severe loss of function of safety systems that could lead to accident conditions or deteriorate accidents	JCO, Japan, 1999
	2	Failures that require the reevaluation of safety systems, but could not possibly lead to or deteriorate accidents	Ciebo NPP, France, 1998
	1	Abnormally beyond the authorized regime due to equipment failures, worker error, defects of procedures	
Deviation	0	Failures regarded as part of normal operation and not affecting safety	

2. Method of event rating

A. Rating Criteria

1) Rating of event scale shall be done considering actual effects rendered facility, environment, public and workers due to accidents, failures or decline of safety function

2) Rating criteria by contributing factors is as follows:

a. Off-site and on-site effects are as follows

Class	Off-site effect	On-site effect
7	◦ External release of I-131 equivalent more than 10^{16} Bq	
6	◦ External release of I-131 of 10^{15} - 10^{16} Bq	
5	◦ External release of I-131 of 10^{14} - 10^{15} Bq	◦ Melting of core less than 10 % or release of less than 10 % core internals from fuel assembly ◦ When severe off-site effect might be possible due to radioactivity release equivalent to the above statement at other than nuclear power plant
4	◦ External release of radioactive material resulting in less than 10 mSv of exposure ◦ Loss of radiation source or transportation accident resulting in death of non-workers	◦ Release of more than 0.1 % core internals from fuel assembly ◦ Exposure of radiation workers more than 5 Gy ◦ Irretrievable radiation release of around 10^{15} Bq at the facility other than nuclear power plant
3	◦ External release by crystal group of radioactive material resulting in exposure more than hundreds μSv ◦ Loss of radiation source or transportation accident resulting in spontaneous radiation effect to public	◦ Contamination that resulted in or might result in exposure of 1 Gy for whole body or 10 Gy for skin to workers ◦ Retrievable radiation release of around 10^{15} Bq at the facility other than nuclear power plant
2		◦ Workers exposure exceeding yearly dose limit ◦ When medical action is needed anticipating that the exposure might reach yearly dose limit ◦ When sum of gamma and neutron irradiation measured at the distance of 1 m from source at the operation mode is beyond 50 mSv/h ◦ Existence of severe amount of radioactive material at the area not predicted on design - Contamination by liquid waste containing radiation of ~ 10^{11} Bq equivalent to Ru-106 Equivalent Dose - When surface or airborne contamination is more than 10 times the operation area limit due to spill of solid waste of ~ 10^{11} Bq equivalent to Ru-106 Equivalent Dose - Release of gaseous radioactive material inside the building of ~ 10^{10} Bq equivalent to I-131 Equivalent Dose

b. Rating for degradation of defence in-depth shall be done according to the following criterion and be such that the event is supposed to occur actually and thereafter the effect to safety function is evaluated.

(1) Event rating for impact on defense in depth with an initator

Initiator Frequency[1)] / Safety Function Operability[2)]	**Expected**	**Possible**	**Unlikely**
Full	0	1	2～3
Within OL&C	1～2[3)]	2～3[3)]	2～3[3)]
Adequate	2～3[3)]	2～3[3)]	2～3[3)]
Inadequate	3[4)]	3[4)]	3[4)]

(2) Event rating for impact on defense in depth without an actual initiator

Initiator Frequency[1)] / Safety Function Operability[2)]	**Expected**	**Possible**	**Unlikely**
Full	0	0	0
Within OL&C	0	0	0
Adequate	1～2[3)]	1	1
Inadequate	3	2	1

<Note 1> Initiator Frequency

○ Expected : initiators which are expected to occur once or several times during the life of the plant.

○ Possible : initiators which are not 'expected', but have an anticipated frequency during the plant lifetime of greater than about 1% (i.e. about 3×10^{-4}/a).

○ Unlikely : initiators considered in the design of the plant which are less likely than the above.

<Note 2> Safety Function Operability

○ Full : all safety systems and components provided by the design to cope with the particular initiator are fully operable.

○ Minimum required (by operational limits and conditions (OL&C)): minimum operability of safety systems specified in the OL&C for continued operation at power, even for a limited time.

○ Adequate : a level of operability of safety systems sufficient to achieve the particular safety function for the initiator being considered.

○ Inadequate : the degraded operability of the safety systems is such that the safety function cannot be fulfilled.

<Note 3> adopt the lower scale when the safety function availability is assured by redundancy and diversity of safety equipment for multiple rates

<Note 4> events that can be rated according to on and offsite effect evaluation guide

(3) When although safety function is degraded, but the degradation does not affect reactor at power

Remaining safety layers[2]	**Maximum potential consequences**[1]		
	Levels 5 ~ 7	Levels 3, 4	Levels 1, 2
More than 3	0	0	0
3	1	0	0
2	2	1	0
1 or 0	3	2	1

<Note 1> The potential highest scale means the highest scale that can occur due to the event the following general principles are applied in determining the scales.

- The applicable facility only shall be evaluated for the case with multiple facilities.
- The amount of potential radioactivity involved, physical and chemical composition and diffusion shape of radiation shall be considered.
- Situation that can happen physically due to defects of safety barriers affected by actual events shall be considered rather than assumptions used in safety evaluation

<Note 2> Safety barriers mean safety measures prepared to maintain safety function besides facility operation and include procedures, administration control, active and passive equipment.(redundancy concept is not applied)

B. Determination of Event Scale

Scale of an event shall be determined higher one after rating in accordance with 2 A.

C. Adjustment of Scale

1) In case there is a need of additional adjustment after rating according to A and B of 2 above, ascension or descension of one scale can be done.

2) Additional factors that can ascend 1 class are as follows.

a) Common cause failure

b) Improper procedures

c) Weak safety culture:

(1) Violations of actions or procedures of limiting condition for operation without evaluation of alternates;

(2) Inadequacy of quality assurance implementation;

(3) Accumulation of workers' errors;

(4) Failure of proper management for external release or improper dose management system; and

(5) Recurrence of failures due to inaction of corrective measures.

3) Additional factors that can descend 1 scale are as follows:

a) Completion of corrective action within shorter time compared to required time for reliable restart; and

b) When the duration of inoperability of the system is shorter than test interval of safety system.

D. Confirmation of Scale

Rating, determination and adjustment according to A~C of 2 will be followed by final confirmation about conformation to the following definition of scales:

1) Scale 0 : When the event does not result in important affect to safety;

2) Scale 1 : Deviation of operation range allowed;

3) Scale 2 : Important failure in terms of safety; and

4) Scale 3 : Failure close to accident (A state where function of defence in-depth is not left).

E. Examples of Rating by Frequency of Occurrence affecting Reactor in Power Operation

1) Pressurized water reactor

Expected	**Possible**	**Unlikely**
1. Reactor trip 2. Inadvertent chemical shim dilution 3. Loss of main feedwater flow 4. Reactor coolant system depressurization by inadvertent operation of an active component (e.g. a safety or relief valve) 5. Inadvertent reactor coolant system by normal or auxiliary pressurizer spray cooldown 6. Steam generator tube leakage in excess plant Technical Specifications, but less than the equivalent of a full tube rupture 7. Power conversion system leakage that would not prevent a controlled reactor shutdown and/or cooldown 8. Reactor coolant system leakage that would not prevent a controlled reactor shutdown and/or cooldown 9. Loss of off-site AC power, including of voltage and frequency disturbances 10. Operation with a fuel assembly in any misoriented or misplaced position 11. Inadvertent withdrawal of any single control assembly during refuelling 12. Minor fuel handling incident 13. Complete loss or interruption of forced reactor coolant flow, excluding reactor coolant pump locked rotor	1. Small LOCA 2. Full rupture of one steam generator tube 3. Dropping of a spent fuel assembly involving only the dropped assembly 4. Leakage from spent fuel pool in excess of normal make-up capability 5. Blowdown of reactor coolant through multiple safety or relief valves	1. Major LOCA, up to and including the largest justified pipe rupture in the reactor 2. Single control rod ejection 3. Major power conversion system pipe rupture, up to and including the largest justified pipe rupture 4. Dropping of a spent fuel assembly onto other spent fuel assemblies

2) Pressurized heavy water reactor

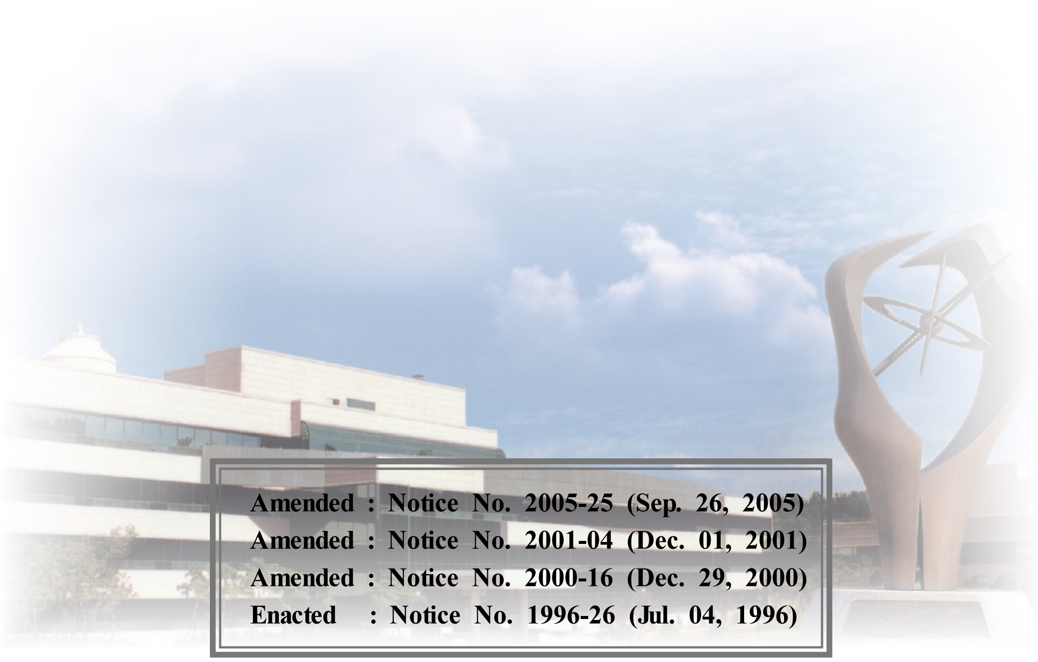
Expected	Possible	Unlikely
1. Reactor trip 2. Inadvertent chemical shim dilution 3. Loss of main feedwater flow 4. Loss of reactor coolant system pressure control (high or low) owing to failure or inadvertent operation of an active component (e.g. feed, bleed or relief valve) 5. Steam generator tube leakage in excess of plant operating specification but less than the equivalent of a full tube rupture 6. Reactor coolant system leakage that would not prevent a controlled reactor shutdown and cooldown 7. Power conversion system leakage that would not prevent a controlled reactor shutdown and cooldown 8. Loss of off-site power AC, including consideration of voltage and frequency disturbances; 9. Operation with fuel bundle(s) in any misplaced position 10. Minor fuel handling incident 11. Reactor coolant pump(s) trip 12. Loss of main feedwater flow to one or more steam generators 13. Flow blockage in an individual channel (less than 70%) 14. Loss of moderator cooling 15. Loss of computer control 16. Unplanned regional increase in reactivity	1. Small LOCA (including pressure tube rupture) 2. Full rupture of one steam generator tube 3. Blowdown of reactor coolant through multiple safety or relief valves 4. Damage to irradiated fuel or loss of cooling to fuelling machine containing irradiated fuel 5. Leakage from irradiated fuel bay in excess of normal make-up capability 6. Feedwater line break 7. Flow blockage in an individual channel (more than 70%) 8. Moderator failure 9. Loss of end shield cooling 10. Shutdown cooling failure 11. Unplanned bulk increase in reactivity 12. Loss of service water (low pressure, high pressure service water or recirculated cooling water) 13. Loss of instrument air 14. Loss of on-site electrical power (Class IV, III, II or I)	1. Major LOCA, up to and including the largest justified pipe rupture in the reactor coolant pressure boundary 2. Major power conversion system pipe rupture, up to and including the largest justified pipe rupture

3) Research Reactor

Expected	Possible	Unlikely
1. Loss of primary system cooling 2. Loss of off-site power 3. Loss of bypass flow control 4. Loss of secondary system cooling 5. Loss of cooling for reflector cooling system 6. Improper reactivity insertion at start-up 7. Improper withdrawal of control rod 8. Improper withdrawal of experiment sample	1. Leakage of primary coolant 2. Sticking of primary cooling system pump shaft 3. Leakage of reflector heavy water 4. Piping break of reflector heavy water 5. Falling down during fuel handling at reactor cavity 6. Accident during fuel handling at spent fuel pit	1. Occurrence of design basis earthquake 2. Rupture of beam tube 3. Fuel damage by flow obstruction

【 11 】

Pressure Test Criteria for Major Components of Nuclear Reactor Facilities



Amended : Notice No. 2005-25 (Sep. 26, 2005)
Amended : Notice No. 2001-04 (Dec. 01, 2001)
Amended : Notice No. 2000-16 (Dec. 29, 2000)
Enacted : Notice No. 1996-26 (Jul. 04, 1996)

◎ **Notice of the Minister of Science and Technology No.2005-25 (MOST.reactor.020)**

The Pressure Test Criteria for Major Components of Nuclear Reactor Facilities as provided for in Article 41 (3) of the Regulation on Technical Standards for Nuclear Reactor Facilities, etc. is hereby notified publicly as follows:

September 26, 2005
Minister of Science and Technology

Pressure Test Criteria for Major Components of Nuclear Reactor Facilities

Article 1 (Purpose) The purpose of this notice is to prescribe the pressure test requirements of vessels (excluding auxiliary boilers), piping, major pumps and valves of nuclear reactor facilities as provided for in Article 41 (3) of the Regulation on Technical Standards for Nuclear Reactor Facilities, etc.

Article 2 (Scope of Application) This notice is applied to all pressure-retaining components, appurtenances, and completed systems classified as safety-class by the "Regulation on Safety Classification and Applicable Codes and Standards for Nuclear Reactor Facilities" as provided for in Article 12 of the Regulation on Technical Standards for Nuclear Reactor Facilities, etc. Provided, that bolts, studs, nuts, washers, and gaskets are exempted from the pressure test.

Article 3 (Definitions) The definitions of terms used in this notice shall be as follows:

1. The term "pressure test" means a pressurizing test performed at a pressure higher than designed to confirm the integrity of systems or facilities, including hydrostatic tests or pneumatic tests; and
2. The term "leakage test" means the examination for the leakage of components, appurtenances, or systems following hydrostatic tests or pneumatic tests.

Article 4 (Time of Pressure Test) (1) The installed system shall be pressure tested prior to initial operation. Provided, that items which, when assembled, form a completed system may be tested in the form of subassemblies at the stage of manufacture or installation if the system is inaccessible after installation or its system pressure test is difficult to be performed.

(2) Components and appurtenances shall be pressure tested prior to installation in a system. Provided, that the system pressure test may substitute a component or appurtenance pressure test, only in case the component can be repaired by welding in accordance with the rules of construction codes and re-subjected to the required system pressure test following the completion of repair.

Article 5 (Pressure Test) The requirements for pressure tests shall be as follows:

1. Water or an alternative liquid, as permitted by the design specification, shall be used for the hydrostatic test and the component or system in which the test is to be conducted shall be vented during the filling operation to minimize air pocketing;
2. The gas used as the test medium shall be non-flammable and have no detrimental effects on materials;
3. It is recommended that the pressure test be made at a temperature that will minimize the possibility of brittle fracture. The test pressure shall not be applied until the component, appurtenance, or system and the pressurizing fluid are at the approximately same temperature;
4. Components and appurtenances shall be pressure tested in accordance with codes and standards applied for their design and manufacturing; and
5. The installed system shall be hydrostatically tested at not less than 1.25 times the lowest design pressure of any component (and pneumatically tested at 1.2 times the lowest design pressure for Class 1 systems) within the boundary protected by the overpressure protection devices which satisfy the requirements of construction codes. The hydrostatic test pressure shall be maintained for at least 10 min.

Article 6 (Leakage Test) The requirements for leakage tests shall be as follows:

1. Following the application of the hydrostatic test pressure for the required time, all joints, connections, and regions of high stress, such as regions around openings and thickness transition sections, shall be examined for leakage. Except in the case of pumps and valves which shall be examined at test pressure, this examination shall be made at a pressure equal to the greater of the design pressure or three fourths of the test pressure;

2. Leakage of temporary gaskets and seals, installed for the purpose of conducting the hydrostatic test and which will be replaced later, may be permitted unless the leakage exceeds the capacity to maintain system test pressure for the required amount of time. Other leaks, such as those from permanent seals, seats, and gasketed joints in components, may be permitted when specifically allowed by the design specification. Leakage from temporary seals or leakage permitted by the design specification shall be directed away from the surface of the component to avoid masking leaks from other joints; and
3. When Class 2 and Class 3 piping welds are inaccessible for leakage tests, no pressure drop shall be allowed when the pressure test is conducted for more than 1 hr (in case wall thickness exceeds 1 inch, 1 hour per 1 inch is added) at test pressure under the conditions of radiographic examination for butt welds and magnetic particle examination or liquid penetrant examination for socket weld or partial penetration welds.

Article 7 (Special Test Pressure Situation) In special situation where the requirements of Articles 5 and 6 are not applicable, the requirements for pressure tests shall be as follows:

1. Components designed for external pressure only shall be subjected to an internal test pressure at 1.25 times the design external pressure;
2. Pressure chambers of combination units that have been designed to operate independently shall be hydrostatically tested as separate vessels; that is, each chamber shall be tested without pressure in the adjacent chamber. When pressure chambers of combination units have their common elements designed for the maximum differential pressure that can occur during startup, operation, and shutdown, and the differential pressure is less than the higher of the design pressures of the adjacent chambers, the common elements shall be subjected to a hydrostatic test pressure of at least 1.25 times the maximum differential pressure. Following the hydrostatic test of the common elements and their inspection, the adjacent chambers shall be hydro-statically tested in accordance with Article 5;
3. When the damage of pumps is anticipated, the piping between the discharge line of a centrifugal pump and the first shutoff valve may be pressure and leakage tested at the maximum operating pressure; and
4. Vessels and their piping connection with a design pressure of less than 15 psig shall be pressure tested in accordance with codes and standards applied to their design and manufacturing.

Article 8 (Preparation for Testing) The followings shall be prepared before pressure tests.

1. All joints including welded joints shall be left uninsulated and exposed for examination during the test and foreign materials such as paint particles, etc. shall be removed for pre-service pressure tests. Provided, that the followings shall be applied in case of primary painting before pressure test for protection from corrosion due to long period storage:
   a. For the piping or equipment whose calculated pressure is below 10 kg/cm^2 (hereinafter the pressure means instrument pressure), pressure test shall be performed at not lower than 10 kg/cm^2;
   b. The pressure test shall be kept for not less than 30 minutes; and
   c. For the piping or equipment whose test pressure is below 50 kg/cm^2, magnetic test (MT) or penetration test (PT) shall be performed in addition to non destructive test for all the welded joints in accordance with the relevant codes and standards.
2. Components designed to contain vapor or gas may be provided with additional temporary supports, if necessary, to support the weight of the test liquid.
3. Pressure test gages used in pressure testing shall be connected directly to the component concerned. If the indicating gage is not readily visible to the operator controlling the pressure applied, an additional indicating gage shall be provided where it will be visible to the operator for the duration of the test. For systems with a large volumetric content, it is recommended that a recording gage be used in addition to the indicating gages. Analog type indicating pressure gages used in testing shall be graduated over a range not less than 1.5 times nor more than 4 times the test pressure. Digital type pressure gages may be used without range restriction provided the combined error due to calibration and readability does not exceed 1% of the test pressure.

Article 9 (Machining After Pressure Test) An additional amount of material, not to exceed 10% of the wall thickness or 3/8 in. (10 mm), whichever is less, is permitted on the completed component during pressure testing where machining to critical dimensions and tolerances is required.

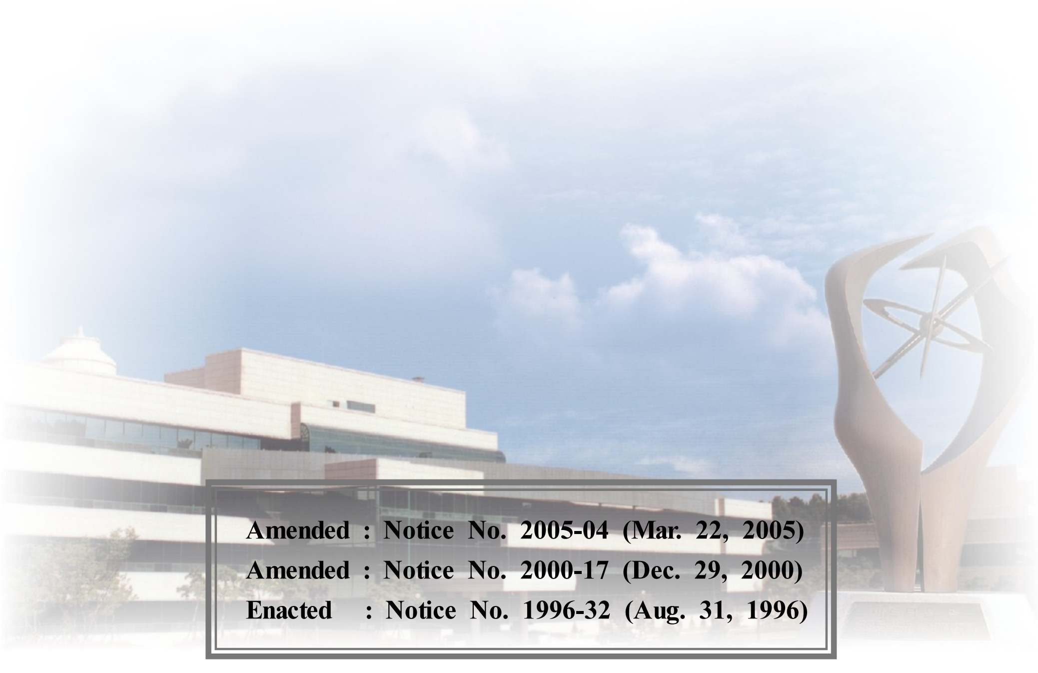
Addenda

Article 1 (Enforcement Date) This notice shall enter into force on the date of its promulgation.

Article 2 (Repeal of Notice) Notice of the MOST No.2001-40 "Pressure Test Criteria for Major Components of Nuclear Reactor Facilities" is repealed at the time this Notice enters into force.

【 12 】

Guidelines for Application of Korea Electric Power Industry Code (KEPIC) as Technical Standards of Nuclear Reactor Facilities



Amended : Notice No. 2005-04 (Mar. 22, 2005)
Amended : Notice No. 2000-17 (Dec. 29, 2000)
Enacted : Notice No. 1996-32 (Aug. 31, 1996)

◎ **Notice of the Minister of Science and Technology No.2005-04 (MOST.reactor.021)**

The Guidelines for Application of the Korea Electric Power Industry Code issued by the Korea Electric Association as the technical standards related to the construction and operation of nuclear power reactor and related facilities, defined in Articles 12 and 22 of the Atomic Energy Act, are hereby notified publicly as follows:

March 22, 2005
Minister of Science and Technology

Guidelines for Application of Korea Electric Power Industry Code (KEPIC) as Technical Standards of Nuclear Reactor Facilities

Article 1 (Purpose) The purpose of this notice is to prescribe the necessary requirements related to scope, method, etc. in applying Korea Electric Power Industry Code as the technical standard for the nuclear power reactor and its related facilities (hereinafter referred to as "nuclear reactor facilities") defined in Articles 12 and 22 of the Atomic Energy Act.

Article 2 (Definitions of Terms) Definitions of the terms used in this notice shall be as follows:

1. The term "technical standards for nuclear reactor facilities" means the technical standards used to confirm the safety of nuclear reactor facilities by the Minister of Science and Technology, which are based on the standards of construction permit and operating license provided for in Articles 12 and 22 of the Atomic Energy Act and the technical standards provided for in Articles 3 through 85 of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc;
2. The term "Korea Electric Power Industry Code (KEPIC)" means the industrial technical standards developed and maintained by the Korea Electric Association for applying to the electrical industry; and
3. The term "reference standards" means the technical standards used as technical basis in developing KEPIC as defined in the column of "reference standards" of Table 1.

Article 3 (Scope of Application) (1) This notice may be applied to the safety-related facilities classified in accordance with "Regulation on Safety Classification and Applicable Codes and Standards for Nuclear Reactor Facilities".

(2) The whole or partial portion of the KEPIC defined in this notice may be applied optionally to the specific nuclear reactor facilities. In this case, the details such as applicable date, scope of application, etc. of the KEPIC shall follow the licensing conditions or the related regulations for the facilities.

(3) KEPIC applied as the technical standards of nuclear reactor facilities in accordance with this notice shall be limited to those corresponding standards in KEPIC 2000 edition, 2001 addenda, 2002 addenda and 2003 addenda to the reference standards of Table 1. Provided, that Paragraph 2 may apply to the technical standards which are not included in Table 1.

Article 4 (Application Method) The method of application of the KEPIC to the technical standards for nuclear reactor facilities shall follow Articles 5 through 8.

Article 5 (Interpretation of Standard) If there is an argument among the related parties on the interpretation of KEPIC as technical standard for nuclear reactor facilities, the interpretation of the Minister of Science and Technology shall prevail.

Article 6 (Application of Technical Contents Verified) The KEPIC contents which have been verified as suitable to assure the safety of domestic or foreign nuclear reactor facilities shall be applied as the technical standards of nuclear reactor facilities as follows:

1. The reference standards shall be applied in case there are any differences in technical contents between KEPIC and reference standards. Provided, that KEPIC shall be applied when the priority of KEPIC is stipulated in Remarks of Table 1 or when the Minister of Science and Technology deems KEPIC as suitable; and
2. The technical contents of reference standards which are not included in KEPIC may be applied as the technical standards of the nuclear reactor facilities when the Minister of Science and Technology deems it necessary for the safety of nuclear reactor facilities.

Article 7 (Application of Technical Contents Unverified) (1) The contents of KEPIC which have not been used in domestic or foreign nuclear reactor facilities, nor verified as suitable to assure the safety may be applied only in case the Minister of

Science and Technology approves its suitability.

(2) The limitations defined in Table 2 shall be followed for the application of KEPIC.

Article 8 (Report) (1) The Korea Electric Association shall report the status of development and maintenance of KEPIC, operation status of certification system, etc. semi-annually to the Minister of Science and Technology. The Minister may request the corrective action for the reported contents.

(2) The Korea Electric Association shall take a corrective action for the request of Paragraph 1 unless there is any special reason.

Addenda

Article 1 (Enforcement Date) This notice shall enter into force on the date of its promulgation.

Article 2 (Relationship with Other Existing Notices) The relationship between this notice and the other existing notices shall be as follows, in spite of the enforcement of this notice.

1. The KEPIC issued before this notice may be applied only if such KEPIC meets requirements of this notice.
2. The detailed requirements for "Safety Classification and Applicable Codes and Standards", "In-Service Inspection", and "In-Service Test" shall be defined separately from this notice.

Article 3 (Repeal of Notice) Notice of the MOST No.2000-17 "Guidelines for Application of KEPIC as the Technical Standards of Nuclear Reactor Facilities" is repealed at the time this notice enters into force.

[Table 1]

KEPIC to be applied as Technical Standard of Nuclear Reactor Facilities

Area	Category	Title	Reference Standards	Remarks
QA (Quality Assurance)	QAP	Nuclear Quality Assurance	ASME NQA-1 (‘94 edition, ‘95 addenda)	
	QAI	Authorized Inspection	ASME QAI-1 ('95 edition, ‘96 addenda)	KEPIC shall be prior.
	QAR	Certificate of Registered Professional Engineer	ASME Appendix XXIII (‘96 addenda)	KEPIC shall be prior.
MN (Nuclear Mechanical Components)	MNA	General Requirements	ASME III NCA ('95 edition ~ ’00 addenda)	KEPIC shall be prior for MNA. Table 2 shall be applied.
	MNB	Class 1 Components	ASME III Div.1 NB ('95 edition ~ ’00 addenda)	
	MNC	Class 2 Components	ASME III Div.1 NC ('95 edition ~ ’00 addenda)	
	MND	Class 3 Components	ASME III Div.1 ND ('95 edition ~ ’00 addenda)	
	MNE	Class MC Components	ASME III Div.1 NE ('95 edition ~ ’00 addenda)	
	MNF	Component Supports	ASME III Div.1 NF ('95 edition ~ ’00 addenda)	
	MNG	Core Support Structures	ASME III Div.1 NG ('95 edition ~ ’00 addenda)	
	MNZ	Appendices	ASME III Div.1 NZ ('95 edition ~ ’00 addenda)	
MI (In-Service Inspection of Nuclear Power Plant Components)	MIA	General Requirements	ASME XI Div.1 IWA ('95 edition ~ ’00 addenda)	Table 2 shall be applied.
	MIB	Class 1 Components	ASME XI Div.1 IWB ('95 edition ~ ’00 addenda)	
	MIC	Class 2 Components	ASME XI Div.1 IWC ('95 edition ~ ’00 addenda)	
	MID	Class 3 Components	ASME XI Div.1 IWD ('95 edition ~ ’00 addenda)	
	MIE	Class MC and CC Components	ASME XI Div.1 IWE ('95 edition ~ ’00 addenda)	
	MIF	Class 1, 2, 3, and MC Component Supports	ASME XI Div.1 IWF ('95 edition ~ ‘00 addenda)	
	MIL	Requirements for Class CC Concrete Components	ASME XI Div.1 IWL ('95 edition ~ ‘00 addenda)	
	MIZ	Appendices	ASME XI Div.1 Appedix ('95 edition ~ ’00 addenda)	

Area	Category	Title	Reference Standards	Remarks
MO (In-Service Testing of Nuclear Power Plant Components)	MOA	General Requirements	ASME OM-ISTA ('95 edition ~ '99 addenda)	Table 2 shall be applied.
	MOB	Inservice Testing of Pumps	ASME OM-ISTB ('95 edition ~ '00 addenda)	
	MOC	Inservice Testing of Valves	ASME OM-ISTC ('95 edition ~ '00 addenda)	
	MOD	Inservice Testing of Pressure Relief Devices	ASME OM-App. I ('95 edition ~ '00 addenda)	
	MOE	Inservice Testing of Snubbers	ASME OM-lSTD ('95 edition ~ '00 addenda)	
	MOF	Performance Testing of Closed Cooling Water Systems	ASME OM-Part 2 ('94 edition, '99 addenda)	
	MOG	Vibration Testing of Piping Systems	ASME OM-Part 3 ('94 edition ~ '00 addenda)	
	MOH	Performance Testing and Monitering of Power-Operated Relief Valve(PORV) Assemblies	ASME OM-Part 13 ('94 edition ~ '00 addenda)	
	MOI	Inservice Testing and Maintenance of Diesel Devices	ASME OM-Part 16 ('94 edition, '99 addenda)	
	MON	Code Cases	ASME OMN Code Case ('98 edition ~ '00 addenda)	
MF (Functional Qualification of Mechanical Equipment used in Nuclear Power Plants)	MFA	General Requirements	ASME QME-1 Sec.QR ('97 edition, 98 addenda)	
	MFB	Functional Qualification of Active Pump Assemblies	ASME QME-1 Sec.QP ('97 edition)	
	MFC	Functional Qualification of Active Valve Assemblies	ASME QME-1 Sec.QV ('97 edition, '00 addenda)	

Area	Category	Title	Reference Standards	Remarks
EN (Nuclear Electrical Components)	ENA	General Requirements	ANSI/ANS 51.1-1983 (R1988)	
	ENB	Design	IEEE 279('71, R78), 308('91), 352('87, R93), 379('94, '00), 384('92, R97), 420('82), 494('74, R9), 497('81), 577('76, R92), 603('98), 7-4.3.2('93), 1023('88, R95),ANSI/ISA S67.04('94, '00)	
	END	Qualification	IEEE 323('83, R96), 344('87), 420('82), 627('80, R96)	
	ENF	Periodic Surveillance Testing	IEEE 338('87, R93)	
SN (Nuclear Structures)	SNA	General Requirements	ASME III, NCA ('95 edition ~ '00 addenda)	KEPIC shall be prior.
	SNB	Concrete Containment	ASME III, Div.2 CC ('95 edition ~ '00 addenda)	
	SNC	Concrete Structures	ACI 349('97)	
	SND	Steel Structures	AISC N690('94)	
ST (Structure General)	STA	Design Loads	ASCE 7('98) Load criteria and Interpretation for Architecture Issued by the Architectural Society of Korea('00)	Table 2 shall be applied.
	STB	Seismic Analysis	ASCE 4('86), IEEE 344('87), ASME QME-1('97), ANSI/ANS-2.2('88)	KEPIC shall be prior.

[Table 2]

Limitations for Application of KEPIC 2000 Edition and 2001/2002/2003 Addenda

Area	Limitations
Common	(1) The requirements of KEPIC which are not consistent with the quality assurance requirements of the Atomic Energy Law and its subordinate law shall not be applied.
MN (Nuclear Mechanical Components)	(1) Weld leg dimensions Licensees shall not apply Paragraph MNB-3683.4(3)(A), the equation of footnote (7) in Figure MNC-3673.2-1 and Figure MND-3673.2-1. (2) Seismic design. Licensees shall use Articles MNB-3200, MNB-3600, MNC-3600, and MND-3600 up to and including the 1995 Edition. Licensees shall not use these Articles in the 1996 Edition through the latest edition and its addenda. (3) Independence of inspection Licensees shall not apply MNA 4200.10(1).
MI (In-Service Inspection of Nuclear Power Plant Components)	(1) Licensees shall not apply the reference standards MIA-1600-1 in Table. (2) Licensees shall apply re-certification period of 3 years only for level I, II instead of 5 years for level I, II, III defined in MIA 2314. (3) The 'authorized inspector' in MIA 4410 which allows the other procedure of welding material control in case of acceptance by the authorized inspector, shall be changed to 'regulatory agency'. (4) The sentence of exemption of the periodic system pressure test for the penetrating piping of containment vessel defined in MIA 5110 (3) shall be deleted. (5) The in-service inspection program for steam generator tubing is governed by the relevant requirements in the technical specifications for operation, etc. (6) Examination of Concrete Containment Structure For the applications of MIL of KEPIC 2001/2002/2003 addenda, the licensee shall apply the followings additionally. (A) For Class CC applications, the licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such

Area	Limitations
MI (In-Service Inspection of Nuclear Power Plant Components)	inaccessible areas. For each inaccessible area identified, the licensee shall provide the followings in the ISI Summary Report required by MIA-6000 1) a description of the type and estimated extent of degradation, and the conditions that led to the degradation; 2) an evaluation of each area, and the result of the evaluation ; and 3) a description of necessary corrective actions. (B) Personnel who examine containment concrete surfaces and tendon hardware, wires, or strands shall meet the qualification provisions in MIA 2300. The "owner-defined" personnel qualification provisions in MIL 2310(4) are not approved for use. (7) Examination of metal containments and the liners of concrete containment Licensees applying Subsection IWE, KEPIC 2001/2002/2003 Addenda, shall satisfy the followings additionally. (A) For Class MC applications, the licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. For each inaccessible area identified, the licensee shall provide the followings in the ISI Summary Report as required by MIA-6000: 1) a description of the type and estimated extent of degradation, and the conditions that led to the degradation; 2) an evaluation of each area, and the result of the evaluation; and 3) a description of necessary corrective actions. (B) The following Requirements may be used as an alternative to the requirements of IWE-2430. If the examinations reveal flaws or areas of degradation exceeding the acceptance standards of Table MIE-3410-1, an evaluation must be performed to determine whether additional component examinations are required. For each flaw or area of degradation identified which exceeds acceptance standards, the licensee shall provide the followings in the ISI Summary Report required by MIA-6000: 1) a description of each flaw or area, including the extent of degradation, and the conditions that led to the degradation; 2) the acceptability of each flaw or area, and the need for additional examinations to verify that similar degradation does not exist in similar components; 3) a description of necessary corrective actions; and 4) the type and number of additional examination to verify the similar degradation.

Area	Limitations
MI (In-Service Inspection of Nuclear Power Plant Components)	(C) A general visual examination must be performed once each period. When performing remotely the visual examinations, the maximum direct examination distance specified in Table MIA- 2210-1 may be extended and the minimum illumination requirements specified in Table MIA- 2210-1 may be decreased provided that the conditions or indications for which the visual examination is performed can be detected at the chosen distance and illumination. (D) VT-1 and VT-3 examinations must be conducted in accordance with MIA-2200. Personnel conducting examinations in accordance with the VT-1 or VT-3 examination method shall be qualified in accordance with MIA-2300. The "owner-defined" personnel qualification provisions in MIE-2330(1) for personnel who conduct VT-1 and VT-3 examinations are not approved for use. (E) The VT-3 examination method shall be used for the examinations in Items E1.12 and E1.30 of Table MIE-2500-1, and the VT-1 examination method shall be used for the examination in Item E4.11 of Table MIE-2500-1. An examination of the pressure-retaining bolted connections in Item E1.11 of Table MIE-2500-1 shall be conducted once each period using the VT-3 examination method. The "owner-defined" visual examination provisions in MIE-2310(1) are not approved for use for VT-1 and VT-3 examinations. (F) Containment bolted connections that are disassembled during the scheduled examinations in Item E1.11 of Table MIE-2500-1 shall be examined using the VT-3 examination method. Flaws or degradation identified during the VT-3 examination must be examined using the VT-1 examination method. The criteria in the material specification or MIB-3517.1 shall be used to evaluate containment bolting flaws or degradation. As an alternative to performing VT-3 examinations of containment bolted connections that are disassembled during the scheduled performance of Item E1.11, VT-3 examinations of containment bolted connections may be conducted whenever containment bolted connections are disassembled for any reason. (G) The ultrasonic examination acceptance standard specified in MIE-3511.3 for Class MC pressure-retaining components shall also be applied to metallic liners of Class CC pressure-retaining components. (H) Following items shall be examined additionally: 1) Circumferential welds of flued head and bellows seal penetration shall be examined in addition to the item E1.10 of Table MIE 2500-1. 2) Sealants, gaskets, dissimilar metal welds, and bolt connections shall be examined in accordance with items E5.10, E5.20, E7.10, and E8.20 in Table MIE 2500-1 of KEPIC 2000 Edition, respectively. (8) Class 1 piping Licensees may not apply MIB-1220, "Components Exempt from Examination," of KEPIC and shall apply IWB-1220, 1989 Edition of ASME Code Section XI.

Area	Limitations
MI (In-Service Inspection of Nuclear Power Plant Components)	(9) Underwater Welding The provisions in MIA-4660 “underwater welding” of KEPIC MI are not approved for use on irradiated material. (10) Flaws of Class 3 piping ASME Code Case N-513(Rev. 0), "Evaluation Criteria for Temporary Acceptance of Flaws in Class 3 Piping," and N-523-1, "Mechanical Clamping Devices for Class 2 and 3 Piping" may be applied. For the applications of Code Case N-523-1, the licensee shall apply all the requirements of this Code Case. For the applications of Code Case N-513, the licensee shall apply all the requirements of this Code Case on the following conditions. (A) For the applications of Code Case N-513, specific safety factors of Article 4.0 shall be met. (B) Code Case N-513 may not be applied in the following cases: 1) components other than pipe and tube, such as pumps, valves, expansion joints, and heat exchangers; 2) leakage thru flange gasket; 3) non-structural seal-welded threaded connections to prevent leakage (Integrity of thread shall be maintained even thought leak path of seal weld is not a structural flaw.); and 4) failed socket weld. (11) MIZ, Appendix VIII personnel qualification All personnel qualified for performing ultrasonic examinations in accordance with Appendix VIII shall receive 8 hours of annual hands-on training on specimens that contain cracks. Training must be completed no earlier than 6 months prior to performing ultrasonic examinations at a licensee's facility. Training time requirement may not be applied in case that the personnel continuously performs ultrasonic examination continually. (12) MIZ, Appendix VIII specimen set requirements (A) When applying Supplements 2, 3, and 10 to Appendix VIII, the following examination coverage criteria requirements must be used. 1) Piping must be examined in two axial directions, and when examination in the circumferential direction is required, the circumferential examination must be performed in two directions, provided access is available. Dissimilar metal welds must be examined axially and circumferentially.

Area	Limitations
MI (In-Service Inspection of Nuclear Power Plant Components)	2) Where examination from both sides is not possible, full coverage credit may be claimed from a single side for ferritic welds. Where examination from both sides is not possible on austenitic welds or dissimilar metal welds, full coverage credit from a single side may be claimed only after completing a successful single-sided Appendix VIII demonstration using flaws on the opposite side of the weld. Dissimilar metal weld qualifications must be demonstrated from the austenitic side of the weld and may be used to perform examinations from either side of the weld. (B) The following provisions must be used in addition to the requirements of Supplement 4 to Appendix VIII: 1) Related to Paragraph 3.1[Detection acceptance criteria] of Supplement 4, Personnel are qualified for detection if the results of the performance demonstration satisfy the detection requirements of Appendix VIII, Table VIII-S4-1 and no flaw greater than 0.25 inch through wall dimension is missed ; and 2) Related to Paragraph 1.1(5) [Detection test matrix] of Supplement 4, Flaws smaller than the 50 percent of allowable flaw size, as defined in MIB-3500, need not be included as detection flaws. For procedures applied from the inside surface, use the minimum thickness specified in the scope of the procedure to calculate a/t. For procedures applied from the outside surface, the actual thickness of the test specimen is to be used to calculate a/t. (C) When applying Supplement 4 to Appendix VIII, the following provisions must be used: 1) A depth sizing requirement of 0.15 inch RMS must be used in lieu of the requirements in Subparagraphs 3.2(1) of Supplement 4, and a length sizing requirement of 0.75 inch RMS must be used in lieu of the requirement in Subparagraph 3.2(2). 2) In lieu of the location acceptance criteria requirements of Subparagraph 2.1(2) of Supplement 4, a flaw will be considered detected when reported within 1.0 inch or 10 percent of the metal path to the flaw, whichever is greater, of its true location in the X and Y directions. 3) In lieu of the flaw type requirements of Subparagraph 1.1(5)(a) of Supplement 4, a minimum of 70 percent of the flaws in the detection and sizing tests shall be cracks. Notches, if used, must be limited by the followings: a) Notches must be limited to the case where examinations are performed from the clad surface; b) Notches must be semi-elliptical with a tip width of less than or equal to 0.010 inches; and c) Notches must be perpendicular to the surface within ± 2 degrees. 4) In lieu of the detection test matrix requirements in paragraphs 1.1(5)(b) and 1.1(5)(c) of Supplement 4, personnel demonstration test sets must contain a representative distribution of flaw orientations, sizes, and locations.

Area	Limitations
MI (In-Service Inspection of Nuclear Power Plant Components)	(D) The following provisions must be used in addition to the requirements of Supplement 6 to Appendix VIII: 1) With regard to Paragraph 3.1[Detection Acceptance Criteria] of Supplement 6, the following provisions must be met for the detection qualification of personnel: a) No surface connected flaw greater than 0.25 inch through wall has been missed; and b) No embedded flaw greater than 0.50 inch through wall has been missed. 2) With regard to Paragraph 3.1[Detection Acceptance Criteria] of Supplement 6, all flaws within the scope of the procedure are detected for procedure qualification. 3) With regard to Paragraph 1.1(2) of Supplement 6, flaws smaller than the 50 percent of allowable flaw size, as defined in MIB-3500, need not be included as detection flaws. Flaws which are less than the allowable flaw size, as defined in MIB-3500, may be used as detection and sizing flaws. 4) Notches are not permitted. (E) When applying Supplement 6 to Appendix VIII, the following provisions must be used: 1) A depth sizing requirement of 0.25 inch RMS must be used in lieu of the requirements of Subparagraphs 3.2(1), 3.2(3)(b), and 3.2(3)(c) of Supplement 6; 2) With regard to the location acceptance criteria requirements in Subparagraph 2.1(2) of Supplement 6, a flaw will be considered detected when reported within 1.0 inch or 10 percent of the metal path to the flaw, whichever is greater, of its true location in the X and Y directions; 3) In lieu of the length sizing criteria requirements of Subparagraph 3.2(2) of Supplement 6, a length sizing acceptance criteria of 0.75 inch RMS must be used; 4) In lieu of the detection specimen requirements in Subparagraph 1.1(5)(a) of Supplement 6, a minimum of 55 percent of the flaws must be cracks. The remaining flaws may be cracks or fabrication type flaws, such as slag and lack of fusion. The use of notches is not allowed; and 5) With regard to paragraphs 1.1(5)(b) and 1.1(5)(c) detection test matrix of Supplement 6, personnel demonstration test sets must contain a representative distribution of flaw orientations, sizes, and locations.

Area	Limitations
MI (In-Service Inspection of Nuclear Power Plant Components)	(F) The following provisions may be used for personnel qualification for combined Supplements 4 and 6 to Appendix VIII. Licensees choosing to apply this combined qualification shall apply all of the provisions of Supplements 4 and 6 including the following provisions: 1) For detection and sizing, the total number of flaws shall be at least 10. A minimum of 5 flaws shall be those from Supplement 4, and a minimum of 50 percent of the flaws shall be those from Supplement 6. At least 50 percent of the flaws in any sizing must be cracks. Notches are not acceptable for Supplement 6; 2) Examination personnel are qualified for detection and length sizing when the results of any combined performance demonstration satisfy the acceptance criteria of Supplement 4 to Appendix VIII; and 3) Examination personnel are qualified for depth sizing when Supplement 4 to Appendix VIII and Supplement 6 to Appendix VIII flaws are sized within the respective acceptance criteria of those supplements. (G) When applying Supplement 4 or 6 to Appendix VIII, or combined Supplement 4 and 6 qualification, the following additional provisions must be used, and examination coverage must include the followings: 1) The clad to base metal interface, including a minimum of 15 percent T (Thickness measured from the clad to base metal interface), shall be examined from four orthogonal directions using procedures and personnel qualified in accordance with Supplement 4 to Appendix VIII; 2) If the clad-to-base-metal-interface procedure demonstrates detectabality of flaws with a tilt angle relative to the weld centerline of at least 45 degrees, the remainder of the examination volume is considered fully examined if coverage is obtained in one parallel and one perpendicular direction. This must be accomplished using a procedure and personnel qualified for single-side examination in accordance with Supplement 6. Subsequent examinations of this volume may be performed using examination techniques qualified for a tilt angle of at least 10 degrees; and 3) The examination volume not addressed by (G) 1) above of this limitation shall be considered fully examined if coverage is obtained in one parallel and one perpendicular direction, using a procedure and personnel qualified for single sided examination when the provisions of (G)2) of this limitation are met. (H) When applying Supplement 5 to Appendix VIII, at least 50 percent of the flaws in the demonstration test set must be cracks and the maximum misorientation shall be demonstrated with cracks. Flaws in nozzles with bore diameters equal to or less than 4 inches may be notches. (I) When applying Supplement 5, Paragraph (1), to Appendix VIII, the following provision must be used in calculating the number of permissible false calls: 1) The number of false calls allowed must be D/10, with a maximum of 3, where D is the diameter of the nozzle.

Area	Limitations
MI (In-Service Inspection of Nuclear Power Plant Components)	(J) When applying Supplement 5 to Appendix VIII, qualification for nozzle inside radius may be performed in accordance with Code Case N-552, "Qualification for nozzle inside radius section from the outside surface", if the requirements in (11)(I)1) of this limitations are satisfied. (K) When performing nozzle-to-vessel weld examinations, the following provisions must be used when the requirements contained in Supplement 7 to Appendix VIII are applied for nozzle-to-vessel welds in conjunction with Supplement 4 to Appendix VIII, Supplement 6 to Appendix VIII, or combined Supplement 4 and Supplement 6 qualification. 1) For examination of nozzle-to-vessel welds conducted from the bore, the following provisions are required to qualify the procedures, equipment, and personnel: a) For detection qualification, a minimum of four flaws in one or more full-scale nozzle mock-ups must be added to the test set. The specimens must comply with Supplement 6, paragraph 1.1, to Appendix VIII, except for flaw locations specified in Table VIII S6-1. Flaws may be either notches, fabrication flaws or cracks. Seventy-five(75) percent of the flaws must be cracks or fabrication flaws. Flaw locations and orientations must be selected from the choices shown in paragraph (K) 4) below of this limitaion, Table VIII-S7-1(Modified), with the exception that flaws in the outer eighty-five(85) percent of the weld need not be perpendicular to the weld. There may be no more than two flaws from each category, and at least one subsurface flaw must be included. b) For length sizing of flaws, a minimum of four flaws as in paragraph (K)1)a) of this Notice must be included in the test set. The length sizing results must be added to the results of combined Supplement 4 and 6 to Appendix VIII. The combined results must meet the acceptance standards contained in (12)(E)3) of this limitation. c) For depth sizing of flaws, a minimum of four flaws as in (K) 1) a) above of this limitation must be included in the test set. Their depths must be distributed over the ranges of Supplement 4, Paragraph 1.1, to Appendix VIII, for the inner 15 percent of the wall thickness and Supplement 6, Paragraph 1.1, to Appendix VIII, for the remainder of the wall thickness. The depth sizing results must be combined with the sizing results from Supplement 4 to Appendix VIII for the inner 15 percent and to Supplement 6 to Appendix VIII for the remainder of the wall thickness. The combined results must meet the depth sizing acceptance criteria contained in paragraph (12)(C)1), (12)(E)1) and (12)(F)3) in this limitation.

Area	Limitations
MI (In-Service Inspection of Nuclear Power Plant Components)	2) For examination of reactor pressure vessel nozzle-to-vessel welds conducted from the inside of the vessel, a) The clad to base metal interface and the adjacent examination volume to a minimum depth of 15 percent T (Thickness measured from the clad to base metal interface) must be examined from four orthogonal directions using a procedure and personnel qualified in accordance with Supplement 4 to Appendix VIII as modified by (B) and (C) of (12) in this limitations. b) When the examination volume defined in (K)2)a) of this paragraph cannot be effectively examined in all four directions, the examination must be augmented by examination from the nozzle bore using a procedure and personnel qualified in accordance with (K) 1) of this paragraph. c) The remainder of the examination volume not covered by (K)2)b) or a combination of (K)2)a) and (K)2)b) of this paragraph, must be examined from the nozzle bore using a procedure and personnel qualified in accordance with (K)1) of this paragraph, or from the vessel shell using a procedure and personnel qualified for single sided examination in accordance with Supplement 6 to Appendix VIII, as modified by (D), (E), (F) and (G) of (12) in this limitations. 3) For examination of reactor pressure vessel nozzle-to-shell welds conducted from the outside of the vessel, a) The clad to base metal interface and the adjacent metal to a depth of 15 percent T(Thickness measured from the clad to base metal interface) must be examined from one radial and two opposing circumferential directions using a procedure and personnel qualified in accordance with Supplement 4 to Appendix VIII, as modified by (B) and (C) of (12) in this limitations, for examinations performed in the radial direction, and Supplement 5 to Appendix VIII, as modified by (J) of (12) in this limitations, for examinations performed in the circumferential direction. b) The examination volume not addressed by (K)3)a) in this paragraph must be examined in a minimum of one radial direction using a procedure and personnel qualified for single sided examination in accordance with Supplement 6 to Appendix VIII, as modified by (D), (E), (F), and (G) of (12) in this limitations. 4) Table VIII-S7-1, "Flaw Locations and Orientations," of Supplement 7 to Appendix VIII, shall be modified as follows:

Area	Limitations
MI (In-Service Inspection of Nuclear Power Plant Components)	Table VIII S7-1 Flaw Locations and Orientation Group / Parallel to weld / Perpendicular to weld Inner 15 percent / O / O OD Surface / O / Subsurface / O / (L) As a modification to the requirements of Supplement 8, Subparagraph 1.1(3), to Appendix VIII, notches may be located within one diameter of each end of the bolt or stud. (M) When implementing Supplement 12 to Appendix VIII, only the provisions related to the coordinated implementation of Supplement 3 to Supplement 2 performance demonstrations are to be applied. (13) MIZ, Appendix VIII single side ferritic vessel and piping and stainless steel piping examination. (A) Examinations performed from one side of a ferritic vessel weld must be conducted with equipment, procedures, and personnel that have demonstrated proficiency with single side examinations. To demonstrate equivalency to two sided examinations, the demonstration must be performed to the requirements of Appendix VIII as modified by this paragraph and (B) through (G) of (12) in this limitations, on specimens containing flaws with non-optimum sound energy reflecting characteristics or flaws similar to those in the vessel being examined. (B) Examinations performed from one side of a ferritic or stainless steel pipe weld must be conducted with equipment, procedures, and personnel that have demonstrated proficiency with single side examinations. To demonstrate equivalency to two sided examinations, the demonstration must be performed to the requirements of Appendix VIII as modified by this paragraph and (A) of (12) in this limitations. (14) Certification of NDE personnel. (A) Level I and II nondestructive examination personnel shall be recertified on a 3-year interval in lieu of the 5-year interval specified in MIA-2314 (1) and (2). (B) Paragraph MIA-2316 may only be used to qualify personnel that observe for leakage during system leakage and hydrostatic tests conducted in accordance with MIA-5211(1) and (2). (C) When qualifying visual examination personnel for VT-3 visual examinations under paragraph MIA-2317, the proficiency of the training must be demonstrated by administering an initial qualification examination and administering subsequent examinations on a 3-year interval.

Area	Limitations
MI (In-Service Inspection of Nuclear Power Plant Components)	(15) Alternative non-destructive examination methods. The provisions for the substitution of alternative examination methods, a combination of methods, or newly developed techniques defined in MIA-2240 in KEPIC 2000 must be applied. The provisions in MIA-2240 of KEPIC 2001 addenda(including its later addenda), are not approved for use. The provisions in MIA-4520(3), allowing the substitution of alternative examination methods, a combination of methods, or newly developed techniques for the methods specified in the Construction Code, are not approved for use. (16) System leakage tests When performing system leakage tests in accordance MIA-5213(1), a 10-minute hold time after attaining test pressure is required for Class 2 and Class 3 components that are not in use during normal operating conditions. No hold time is required for the remaining Class 2 and Class 3 components provided that the system has been in operation for at least 4 hours for insulated components or 10 minutes for un-insulated components. (17) Table MIB-2500-1 examination requirements (A) The provisions of Table MIB-2500-1, Examination Category B-D, Full Penetration Welded Nozzles in Vessels, Items B3.120 and B3.140 (examination plan B) in the KEPIC 2000 Edition must be applied when using the KEPIC 2002 addenda(including its later edition and addenda). A visual examination with enhanced magnification that has a resolution sensitivity to detect a 1-mil width wire or crack, utilizing the allowable flaw length criteria in Table IWB-3512-1, may be performed in lieu of an ultrasonic examination. (B) The provisions of Table IWB-2500-1, Examination Category B-G-2, Item B7.80 in the ASME 1995 Edition are applicable only to reused bolting. (C) The provisions of Table IWB-2500-1, Examination Category B-K, Item B10.10 in the ASME 1995 addenda must be applied.

Area	Limitations
MO (In-Service Testing of Nuclear Power Plant Components)	(1) Motor-operated valve testing shall be performed in accordance with the requirements of inservice operating test for category A, B defined in MOC 4200, KEPIC 2000 or the requirements of valve testing defined in MOC 3500, KEPIC 2001/2002/2003 addenda. And a program to ensure that motor-operated valves continue to be capable of performing their design basis safety functions shall be established. (2) Code Cases except MON-1 "Alternative Requirements for preservice and inservice testing of motor operated valve assemblies used in nuclear power plants" may be applied through such procedures as approval of relief request. (3) When applying Appendix I, "Check Valve Condition Monitoring Program" of the MOC, following requirements shall be satisfied. (A) Valve opening and closing functions must be demonstrated when flow testing or examination methods (nonintrusive, or disassembly and inspection) are used. (B) The initial test interval may not exceed two fuel cycles or 3 years, whichever is longer. Any extension of this interval may not exceed one fuel cycle per extension with the maximum interval not to exceed 10 years. Trending and evaluation of existing data must be used to reduce or extend the time interval between tests. (C) If the condition monitoring program is discontinued, then the test requirements of MOC 4510 through 4540, KEPIC 2000 edition, or MOC 3510/3520/3540/5221, KEPIC 2001/2002/2003 addenda must be applied. (4) MOE "Inservice testing for Snubbers" instead of the requirements of Snubbers defined in MIF 5200(1), (2) and MIF 5300(1), (2) may be applied by making appropriate changes to their technical specifications or licensee-controlled documents. (5) Manual valves shall be tested at 2 year interval instead of 5 year interval defined in MOC 3540, KEPIC 2002 addenda.
ST (Structure General)	(1) STA 2000, STA 3270, STA 4000s for design load, which are standards for commercial facilities, are not approved for use in nuclear reactor facilities.

【 13 】

Repeal of "Regulation on Processes subject to Manufacturing Inspection of Nuclear Reactors, etc." and so on

Enacted : Notice No. 2000-09 (Jun. 23, 2000)

◎ **Notice of the Minister of Science and Technology No.2000-09 (MOST.reactor.022)**

The notice to repeal "Regulation on Processes subject to Manufacturing Inspection of Nuclear Reactors, etc." and so on is hereby notified publicly as follows:

June 23, 2000
Minister of Science and Technology

Repeal of "Regulation on Processes subject to Manufacturing Inspection of Nuclear Reactors, etc." and so on

The following notices of the MOST shall be repealed at the time this notice enters into force:

1. Regulation on Processes subject to Manufacturing Inspection of Nuclear Reactors, etc. (No.85-07, July 20, 1985);

2. Standards for Technical Capability and Quality Assurance Program related with the Permit of Nuclear Production Business (No.88-17, December 9, 1988);

3. Standards for Quality Assurance for the Equipment Qualification (No.96-20, June 4, 1996);

4. Standards for Permit for Equipment Qualification Business and the Items Subject to the Qualification (No.96-22, June 11, 1996);

5. Regulation on Applications for Approval of Explanatory Statement of Design and Work Method of Nuclear Fuel Cycle Facilities (No.86-15, February 5, 1986);

6. Technical Standards on the Inspection of Nuclear Fuel (No.96-18, May 20, 1996);

7. Standards on Submission of Design Documents of Nuclear Reactor and Related Facilities (No. 92-10, May 25, 1992);

8. Standards on Quality Assurance of Nuclear Reactor and Related Facilities (No. 90-03, April 6, 1990); and

9. Regulation on Periodic Inspection of Nuclear Reactor Facilities (No.98-06, June 27, 1998).

【 14 】

Standards for Safety Valves and Relief Valves of Nuclear Reactor Facilities

Amended : Notice No. 2001-38 (Dec. 01, 2001)
Enacted : Notice No. 2000-18 (Dec. 29, 2000)

◎ **Notice of the Minister of Science and Technology No.2001-38 (MOST.reactor.023)**

The Standards for Safety Valves and Relief Valves of Nuclear Reactor Facilities as provided for in Article 37 of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc. are hereby notified publicly as follows:

December 1, 2001

Minister for Science and Technology

Standards for Safety Valves and Relief Valves of Nuclear Reactor Facilities

Article 1 (Purpose) The purpose of this notice is to prescribe criteria on safety valves and relief valves of nuclear reactor facilities, as provided for in Article 37 of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc.

Article 2 (Scope of Application) This notice shall be applied to manufacturing and installing pressure relief devices to protect the system against pressure and temperatures beyond design pressure and operation limit, by a holder of construction permit under Article 11 (1) of the Atomic Energy Act.

Article 3 (Definition) Definitions of the terms used in this notice shall be as follows:

1. The term "overpressure" as used in this notice refers to a system pressure increase exceeding design pressure due to thermal imbalances, excess pump flow rate, and other reasons;
2. The term "pressure relief devices" refers to the devices which operate automatically upon reaching a preset pressure to keep the system pressure from exceeding the safety limit and include safety valves, relief valves, safety relief valves, motor-operated pressure relief valves, pilot operated pressure relief valve, and non-reclosing pressure relief device;

3. The term "safety limit" refers to the highest pressure allowed in the system, which is set as the 110% of design pressure; and
4. The definition of safety class used in this notice refers to the one prescribed in the Notice of the Minister of Science and Technology "Regulations on Safety Classification and Applicable Codes and Standards for Nuclear Reactor Facilities."

Article 4 (Installation) Pressure relief devices shall meet the following criteria:

1. Pressure relief devices shall be installed at a location as close as practicable to the possible major sources of overpressure;
2. The connection between a system and a pressure relief device shall be installed in such a manner that flows are not obstructed;
3. The connection between a system and a safety valve shall be such that the pressure loss of the line must not exceed 2% of the relieving pressure;
4. Connections between a system and a safety relief valve or a relief valve shall be such that the pressure loss of the line must not exceed 3% of relieving pressure;
5. Safety, safety relief, and relief valves shall be installed in an upright position;
6. The inside areas of the discharging piping connected with a pressure relief device shall not be less than that of the valve outlet. Any back pressure shall not reduce the relieving capacity of the relieving devices below the level required to protect the system. The potential for flashing shall also be considered in determining the back pressure; and
7. A drain shall be installed at the lowest point such that liquid or residue can be collected and not interfere with proper relieving operation.

Article 5 (Material) Material of the pressure retaining parts of pressure relief devices shall be compatible with the fluid characteristics and environmental conditions, and meet safety classification requirements and the following criteria:

1. The pressure retaining parts of pressure relief devices shall meet material requirements (MD) and nuclear mechanical requirements (MN) of Korea Electric Power Industry Code (hereinafter referred to as "KEPIC") of the Korea Electric Association or equivalent technical standards;
2. Material of pressure relief devices' parts shall withstand fluid characteristics such

as temperature, pressure and flow rate, and environmental conditions such as radiation. Parts containing reactor coolant shall be made of stainless steel or corrosion-resistant material not to suffer corrosion; and

3. Parts of pressure relief devices shall be manufactured with smallest defects possible per size or use.

Article 6 (Relieving Capacity) Relieving capacity and number thereof, etc. of pressure relief devices shall meet the following criteria:

1. Relieving capacity shall be sufficient to keep pressures of any component in a pressure boundary of the system from rising above the safety limits;
2. Relieving capacity shall be sufficient even when used with pressure reducing devices;
3. When more than 1 pressure relief device is used for overpressure protection, the capacity of the smallest one shall not be less than 50% of the largest one; and
4. In case of a component isolable from the system, the relieving capacity required for the overpressure protection of the component shall be provided using one or more pressure relief devices meeting requirements set forth in Article 8.

Article 7 (Set Pressure) The set pressure of at least one of the pressure relief devices connected to a system shall not be greater than the Design Pressure of any component within the pressure retaining boundary of the protected system. Additional pressure relief devices may have higher pressure settings, but in no case shall these settings be such that the total accumulated pressure exceeds the safety limit of the system.

Article 8 (Design Requirements) Pressure relief devices shall meet the following criteria:

1. Spring-loaded valves shall open automatically by the fluid forces directly acting against the spring;
2. Balanced valves, whose operation is independent of back pressure, may be used only when means are provided to verify the operability of the balancing device. To the balanced valves of safety class 1, multi-back pressure balancing device shall be attached additionally;

3. Pressure relief devices shall be designed in such a way that potential impairment of the overpressure protection function by the service exposure to fluids can be determined by test or examination;
4. Pressure relief devices of Safety Class 1 shall be equipped with the means for direct or indirect remote monitoring of their valve position (open or close);
5. Pressure relief devices and their associated pressure sensing elements shall be designed in such a way that their correct operability can be demonstrated under service or test conditions; and
6. Other design requirements, non-reclosing pressure relief devices, and matters related to certification shall follow MNB 7500, 7600, 7700 of the KEPIC or equivalents thereto.

Article 9 (Marking) Manufacturer of pressure relief devices or certifier related to their manufacture shall clearly mark specifications, etc. of pressure relief devices in such a way that the marking will not be obliterated in service.

Addenda

Article 1 (Enforcement Date) This notice shall enter into force on the date of its promulgation.

Article 2 (Revision of Other Notice) The Table 2 "Technical Standards for Structures and Equipment of Nuclear Reactor Facilities" of Notice of the MOST No.2000-08 "Technical Standards for the Locations, Structures and Equipment of Nuclear Reactor Facilities" shall be deleted at the enforcement of this notice.

Article 3 (Repeal of Notice) Notice of the MOST No.2000-18, "Standards for Safety Valves and Relief Valves of Nuclear Reactor Facilities" shall be repealed at the time this notice enters into force.

【 15 】

Standards for Performance of Emergency Core Cooling System of Pressurized Light Water Reactor

Amended : Notice No. 2001-39 (Dec. 01, 2001)
Enacted : Notice No. 2000-19 (Dec. 29, 2000)

◎ **Notice of the Minister of Science and Technology No.2001-39 (MOST.reactor.024)**

The Standards for Performance of Emergency Core Cooling System of the Pressurized Light Water Reactor in accordance with Article 30 (1) of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc. are hereby notified publicly as follows:

December 1, 2001
Minister of Science and Technology

Standards for Performance of Emergency Core Cooling System of Pressurized Light Water Reactor

Article 1 (Purpose) The purpose of this notice is to prescribe the Standards for Performance of Emergency Core Cooling System (ECCS) of the Pressurized Light Water Reactor in accordance with Article 30 (1) of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc.

Article 2 (Definitions) Definitions of terms used in this notice are as follows:

1. The term "loss of coolant accident (LOCA)" means the postulated accident that results from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system; and
2. The term "evaluation model" means the calculational framework for evaluating the behavior of the reactor system during a postulated LOCA. It includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in the programs, procedure for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure.

Article 3 (Acceptance Criteria) Each pressurized water reactor fueled with uranium oxide pellets within cylindrical zirconium alloy cladding shall be provided with an ECCS that must be designed so that its calculated cooling performance following postulated LOCAs conforms to the criteria set forth in the followings:

1. Peak cladding temperature: The calculated maximum fuel element cladding temperature shall not exceed 1204ºC.
2. Maximum cladding oxidation: The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
3. Maximum hydrogen generation: The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel were to react.
4. Coolable geometry: Calculated changes in core geometry shall be such that the core remains amenable to cooling.
5. Long-term cooling: After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Article 4 (Evaluation Model) ECCS cooling performance shall be calculated in accordance with one of the following acceptable evaluation models and shall be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated.

1. Conservative model
   The conservative model must provide the model and calculation method whose conservatism is justified and approved for use.
2. Best estimate model
   The best estimate model must provide sufficient justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in Article 3, there is a high level of probability that the criteria would not be exceeded.

Article 5 (Items related to Change or Error) (1) Each installer or operator of nuclear power reactor shall estimate the effect of any change to or error in an acceptable evaluation model or in the application of such a model, and shall report the estimated results to the Minister of Science and Technology annually.

(2) If the estimated results in accordance with Paragraph 1 fall under any of the followings, they shall be reported to the Minister of Science and Technology within 30 days:

1. When the results in a calculated peak fuel cladding temperature are different by more than 30ºC from the peak cladding temperature calculated for the limiting transient using the last acceptable model; or
2. When the sum of the absolute magnitudes of the respective temperature changes due to accumulated changes and errors is greater than 30ºC.

Addenda

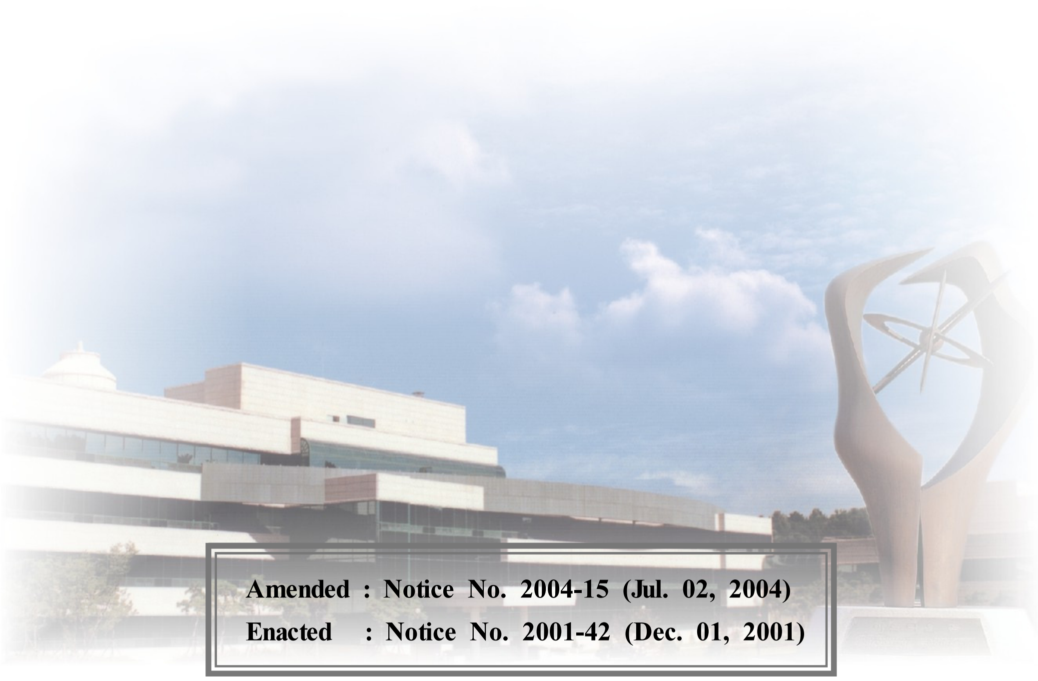
Article 1 (Enforcement Date) This notice shall enter into force on the date of its promulgation.

Article 2 (Repeal of Notice) Notice of the MOST No.2000-19 "Standards for Performance of Emergency Core Cooling System of the Pressurized Light Water Reactor" is repealed at the time this notice enters into force.

Notice of the Minister of Science and Technology No.2004-15 (MOST.reactor.025)

【 16 】

Standards for Leakage Rate Tests of Reactor Containment



Amended : Notice No. 2004-15 (Jul. 02, 2004)
Enacted : Notice No. 2001-42 (Dec. 01, 2001)

◎ **Notice of the Minister of Science and Technology No.2004-15 (MOST.reactor.025)**

The Standards for Leakage Rate Tests of Reactor Containment in accordance with Article 23 (1) 4 and (2) of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc. are hereby notified publicly as follows:

July 2, 2004
Minister of Science and Technology

Standards for Leakage Rate Tests of Reactor Containment

Chapter 1 General Provisions

Article 1 (Purpose) The purpose of this notice is to prescribe the technical standards and related matters for leakage rate tests of reactor containment in accordance with Article 23 (1) 4 and (2) of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc.

Article 2 (Definitions) Definitions of terms used in this notice are as follows:

1. The term "leakage rate tests of reactor containment" means tests to measure the leakage rate across the containment penetrations in order to confirm the leak-tight integrity of the reactor containment;
2. The term "integrated leakage rate tests" means tests to measure the total leakage rate through all potential leakage paths of the reactor containment assuming design basis accidents;
3. The term "local leakage rate tests" means tests intended to detect local leaks and to measure leakage across each pressure-containing or leakage-limiting boundary for the reactor containment penetrations. It includes 'containment isolation valve leakage rate tests' which measures leakage across containment isolation valves and 'containment penetrations leakage rate tests' which measures leakage across electrical penetrations, air lock door seals, and nuclear fuel transfer channels, etc;
4. The term "Pa" means the calculated peak containment internal pressure related to the design basis accident;
5. The term "La" means the maximum allowable leakage rate of reactor containment

at pressure Pa as specified in the technical specifications;

6. The term "leakage rate before maintenance" means the leakage rate which was measured before maintenance or repair;
7. The term "leakage rate after maintenance" means the leakage rate which was measured after maintenance or repair; and
8. The term "maximum pathway leakage rate" means the larger leakage rate between two isolation barriers located in the leakage path of penetration of the reactor containment.

Chapter 2 Integrated Leakage Rate Tests

Article 3 (Test Interval) (1) The operator of a nuclear power reactor (hereinafter referred to as "operator") shall perform the preoperational integrated leakage rate tests before the initial fuel loading.

(2) The operator shall perform the first periodic integrated leakage rate tests within 3 years after the preoperational integrated leakage rate tests, and every 5 years for subsequent tests.

(3) The operator may change the test interval to 10 years when each of the following documents is submitted to, and demonstrated to have its validity by, the Minister of Science and Technology:

1. Safety evaluation on the interval extension of integrated leakage rate tests;
2. Results of more than 2 times of recent integrated leakage rate tests; and
3. Results of the local leakage rate tests during the period of the Subparagraph 2 and operation history related to the leakage of the reactor containment.

(4) The operator shall perform at least 2 times of integrated leakage rate test within 5 year interval when the integrated leakage rate before maintenance does not meet the acceptance criteria of Article 6, even though 10 year interval in accordance with Paragraph 3 was approved.

(5) The operator shall perform the integrated leakage rate test every overhaul period until consecutive 2 times of test are satisfied, when integrated leakage rate before maintenance does not meet the acceptance criteria of Article 6 in consecutive 2 times of periodic test.

Article 4 (Test Condition, etc.) (1) The operator shall perform the integrated leakage rate tests under the conditions of reactor shutdown and control for safety.

(2) The operator shall perform a visual inspection of the accessible interior and

exterior surfaces of the containment structures and components prior to any test and confirm if there is any evidence of structural deterioration which may affect either the containment structural integrity or leak-tightness.

(3) The operator, when the system arrangement does not agree with the preset test condition, shall evaluate the effect to leakage rate and reflect it to the test results.

(4) The operator, when the maintenance was made for the leakage area, shall measure the local leakage rate before and after maintenance and reflect it to the test results.

Article 5 (Test Method) (1) The test pressure shall be less than design pressure of the containment and shall be maintained not less than 0.96 Pa during testing period.

(2) The integrated leakage rate shall be calculated with the measured data which are taken from the beginning after stabilization of containment to the ending time of the stable trend of leakage rate after lasting at least 24 hours.

(3) The verification test shall be performed in order to confirm the correctiveness of the leakage rate tests results, and the composite leakage rate shall be calculated with the measured data, which are taken up to the time of the stable trend of leakage rate lasting at least 4 hours.

Article 6 (Acceptance Criteria) (1) Integrated leakage rate shall not exceed 0.75 La.

(2) Error of the measured leakage rate shall not exceed 0.25 La.

Article 7 (Recording and Reporting) (1) The operator shall provide the report for integrated leakage rate tests results and keep it available upon request.

(2) The report for preoperational integrated leakage rate tests in accordance with Paragraph 1 shall contain schedule, measuring facilities, methods, conditions of integrated leakage rate tests and show that the test data and analysis results meet the acceptance criteria.

(3) The report for periodic integrated leakage rate tests in accordance with Paragraph 1 shall contain the results of local leakage rate tests performed after previous integrated leakage rate tests, maintenance history of the reactor containment penetrations and operation history related to leakage, other than the matters prescribed in Paragraph 2. The report also shall contain analysis results of test data, reasons and relevant corrective actions for the tests which did not meet the acceptance criteria.

(4) When the criteria of Article 6 are not met, the acceptance criteria shall be met before operation mode by the appropriate maintenance and tests.

Chapter 3 Local Leakage Rate Tests

Article 8 (Test Interval) (1) The operator shall perform the preoperational local leakage rate tests before initial fuel loading.

(2) The operator shall perform the local leakage rate tests at the prescribed test intervals for each specific test object.

Article 9 (Test Method) (1) The operator shall perform the local leakage rate tests using one of the following methods:

1. Method of measuring gas flow rate required for maintaining the test pressure in the pressurized condition;
2. Method of measuring change rate of pressure and temperature of test in the pressurized condition; and
3. Other methods demonstrated to be valid as local leakage rate tests.

(2) The test shall be conducted at a pressure not less than Pa, and also not more than 1.10 Pa.

(3) The pressure shall be applied in the same direction as that when the isolation valve would be required to perform its isolation function, unless it can be determined that the results from the test for a pressure applied in a different direction will provide equivalent or more conservative results.

(4) Isolation valve leakage rate tests shall be performed in a condition that the isolation valve are closed by normal operation.

Article 10 (Acceptance Criteria) (1) The combined local leakage rate on maximum pathway leakage rate basis shall not exceed 0.60 La.

(2) Local leakage rate across each penetration shall not exceed each prescribed criteria.

Article 11 (Record and Action) (1) The operator shall provide the record for leakage rate before and after maintenance for each penetration, and keep it available upon request.

(2) The operator shall retest after maintenance, when the acceptance criteria of Article 10 are not met.

Article 12 (Facility Modification) When the facility modifications that can affect to the leak-tight integrity of the reactor containment are made, the operator shall confirm the leak-tightness by measuring the leakage rate for the region.

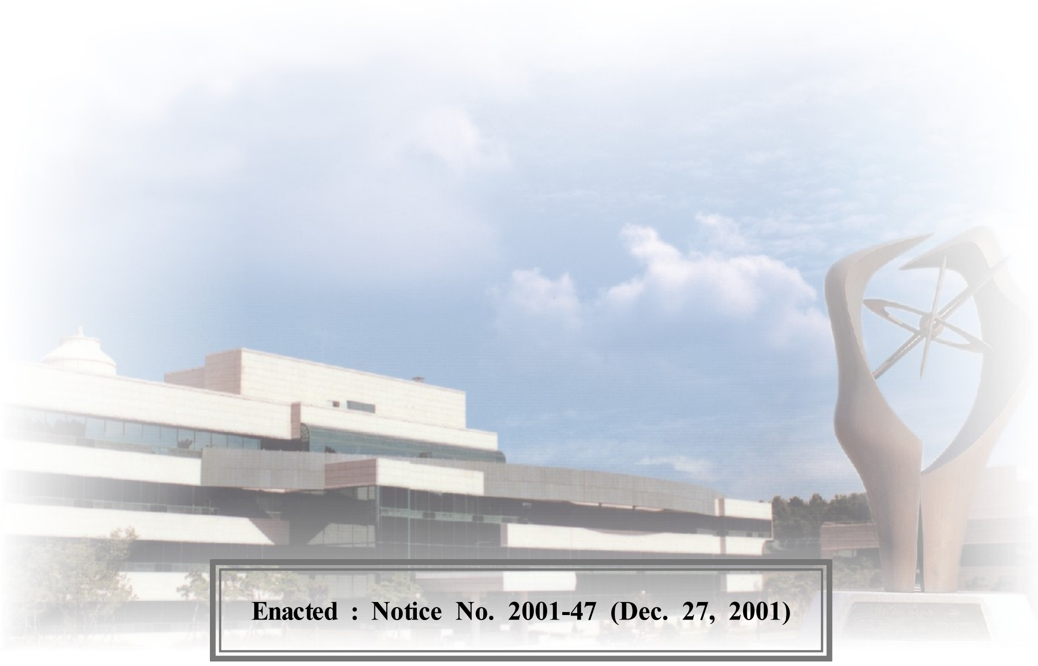
Addenda

Article 1 (Enforcement Date) The notice shall enter into force on the date of its promulgation.

Article 2 (Repeal of Notice) Notice of the MOST No.2001-42 "Standards for Leakage Rate Tests of Reactor Containment" is repealed at the time this notice enters into force.

【 17 】

Detailed Requirements for Quality Assurance of Nuclear Reactor Facilities



Enacted : Notice No. 2001-47 (Dec. 27, 2001)

◎ **Notice of the Minister of Science and Technology No.2001-47 (MOST.reactor.026)**

The detailed requirements for quality assurance regarding the construction and operations of nuclear reactor facilities as provided for in Article 67 (2) of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc. are hereby notified publicly as follows:

December 27, 2001
Minister of Science and Technology

Detailed Requirements for Quality Assurance of Nuclear Reactor Facilities

Article 1 (Purpose) The purpose of this notice is to prescribe the detailed requirements for quality assurance regarding the construction and operations of nuclear reactor facilities as provided for in Article 67 (2) of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc.

Article 2 (Scope of Application) This notice shall apply to quality assurance program prepared and submitted by nuclear power-related enterprisers such as an installer and an operator of nuclear power reactor, an installer/operator of nuclear research reactor, and a nuclear fuel cycle enterpriser as provided for in Articles 11 (2), 21 (2), 33 (2) and 43 (3) of the Atomic Energy Act.

Article 3 (Detailed Requirements) Detailed requirements as provided for in Articles 68 through 85 of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc. shall apply by each of the followings:

1. Construction of nuclear power reactor : Nuclear quality assurance requirements (QAP) of Korea Electric Power Industry Code (KEPIC) applicable under the notice of the MOST No.2005-04 or equivalent technical standards thereto;
2. Operation of nuclear power reactor : Nuclear quality assurance requirements (QAP) of Korea Electric Power Industry Code (KEPIC) applicable under the notice of the MOST No.2005-04 or equivalent technical standards thereto, and ANSI/ANS 3.2 Administrative Control and Quality Assurance for the Operational Phase of Nuclear Power Plants, 1994 Edition; and

3. Construction and operation of nuclear research reactor : ANSI/ANS 15.8 Quality Assurance Program Requirements for Research Reactors, 1995 Edition.

Article 4 (Method of Application) The application method of detailed requirements in this notice is the same as that of Article 4 of notice of the MOST No.2005-04 Guideline for Application of Korea Electric Power Industrial Code as the Technical Standards of Nuclear Reactor Facilities (March 22, 2005).

Addendum

This notice shall enter into force on the date of its promulgation.

【 18 】

Regulation on Pre-operational Inspection of Nuclear Reactor Facilities

Amended : Notice No. 2005-09 (May 18, 2005)
Enacted : Notice No. 2001-48 (Dec. 27, 2001)

◎ **Notice of the Minister of Science and Technology No.2005-09 (MOST.reactor.027)**

The Regulation on Pre-operational Inspection of Nuclear Reactor Facilities (Notice of the MOST No.2001-48, December 27, 2001) is hereby revised and notified publicly as follows:

May 18, 2005
Minister of Science and Technology

Regulation on Pre-operational Inspection of Nuclear Reactor Facilities

Article 1 (Purpose) The purpose of this notice is to prescribe matters on pre-operational inspection of nuclear reactor facilities as provided for in Article 29 of the Enforcement Decree of the Atomic Energy Act (hereinafter referred to as the "Decree").

Article 2 (Facilities subject to Pre-operational Inspection) Nuclear reactor facilities for which installer of nuclear power reactor has to undergo pre-operational inspection in accordance with Article 29 of the Decree are as follows:

1. Reactor pressure vessel;
2. Reactor coolant system facility;
3. Instrumentation and control system facility;
4. Fuel material handling and storage facility;
5. Radioactive waste disposal system facility;
6. Radiation control facility;
7. Reactor containment facility;
8. Reactor safety system facility;
9. Electric power system facility;
10. Power conversion system facility; and
11. Other facilities related to safety of nuclear reactor.

Article 3 (Time, etc. of Inspection) (1) Times for pre-operational inspections per process as provided for in Article 29 (1) of the Decree are as follows:

1. When inspections of main stages such as foundation excavation, rebar installation,

liner plate installation and fabrication, concrete placement, for structuring of reactor containment, etc. can be conducted, and when integrated construction test such as structural integrity test and integrated leakage rate test of containment can be conducted: The time of inspections per main unit process is shown in Table 1.

2. When cold functional test, per each system, of various equipments and components such as pumps, motors, heat exchangers, valves, etc. can be conducted, after completion of installation, to verify functions: The detailed time of inspections per items is shown in Table 2.
3. When cold hydrostatic tests such as reactor coolant system hydrostatic test and steam generator secondary side hydrostatic test can be conducted, and when integrated performance verification by hot functional tests, under conditions similar to actual operation, of systems such as reactor coolant system, instrumentation and control system and reactor safety system can be conducted: The detailed time of inspections per items is shown in Table 3.
4. When inspections of initial fuel loading test, hot functional test after the fuel loading, initial criticality test, low power reactor physics test and power ascension test for each power level can be conducted: The detailed time of inspections per items is shown in Table 4.

(2) The process and time for installer of nuclear power reactor to undergo pre-operational inspection in accordance with Article 29 (2) of the Decree are where and when inspections of installation, welding, non-destructive test, and pressure test of components and piping of systems can be conducted. The detailed time of inspections per items is shown in Table 5.

(3) Inspection items as provided for in Article 3 (1) and (2) may differ by type of reactor and design characteristics, and equipment, components, and structures could be selected for inspection considering their importance.

Article 4 (Inspection Period) The period of pre-operational inspection shall be duration of, structural works including foundation excavation, equipment installation and functional test, and all commissioning tests including power ascending operation for each power level.

Article 5 (Application for Inspection) (1) The Minister of Science and Technology may require the installer of nuclear power reactor to submit the pre-operational inspection application document when he intends to conduct pre-operational inspection as provided for in Article 29 (2) of the Decree.

(2) When an installer of nuclear power reactor intends to change the content of the pre-operational inspection application submitted in accordance with the Paragraph (1), the installer shall submit details and reasons of changes.

Article 6 (Submission of Inspection Plan) The president of Korea Institute of Nuclear Safety shall review the pre-operational inspection application submitted by the installer of nuclear power reactor in accordance with Articles 3 and 5, and shall report inspection plan to the Minister of Science and Technology by 15 days prior to inspection.

Article 7 (Submission of Inspection Report) The president of the Korea Institute of Nuclear Safety shall report to the Minister the pre-operational inspection results within 30 days after completion of inspection when he conducted pre-operational inspection applied for in accordance with Article 3.

Addendum

This notice shall enter into force on the date of its promulgation.

[Table 1]

Time of Inspection of Structures, etc.

(related to Subparagraph 1 of Article 3 (1))

Inspection of structures of nuclear reactor facility shall be conducted at commencement of the following items and when verification of main processes is possible. Inspection of integrated leakage rate tests of containment shall be conducted when review of test procedure such as leakage rate calculation methodology and test are possible.

1. Structures a. Foundation excavation and treatment works b. Permanent dewatering system works c. Structure backfill works d. Facility water proof treatment works e. Rebar installation works f. Mechanical rebar splice works g. Concrete works h. Equipment foundation grout works i. Containment post tension system works j. Containment liner plate installation works k. Steel structure installation works l. Stainless liner plate installation works m. Concrete anchor bolt installation works n. Concrete masonry works o. Seismic qualification inspection p. Radiation resistant coating works q. Sealing works of safety related openings and penetrations
2. Integrated construction test a. Structural integrity test b. Integrated leakage rate test

[Table 2]

Time of Cold Functional Test of Systems

(related to Subparagraph 2 of Article 3 (1))

Inspection of cold functional test of each facility shall be conducted when tests of the following items are possible.

1. Reactor pressure vessel a. Core operating limit supervisory system test b. Core protection calculator test c. Comprehensive vibration assessment program
2. Reactor coolant system facility a. Reactor coolant gas vent system test b. Reactor coolant pump vibration monitoring system test c. Loose part monitoring system test d. Acoustic leak monitoring system test e. Reactor internal vibration monitoring system test
3. Instrumentation and control system facility a. Engineered safety features actuation system test b. Reactor regulating and reactor power cutback system test c. Instrument correlation test d. Reactor regulating system test e. Diverse protection system test f. Plant protection system response time measuring test g. Plant protection system test h. Inadequate core cooling monitoring system test i. Pressurizer pressure and level control test j. Ex-core nuclear instrumentation system test k. Turbine control and protection system test l. Feed water control system test m. Steam bypass control system test
4. Fuel material handling and storage facility a. Spent fuel pool cooling and cleanup system test b. Fuel transfer system and transfer tube test c. Control rod exchange system test d. Spent fuel handling crane test e. Refueling machine test f. Spent fuel pool leak-tight gate
5. Radioactive waste disposal system facility a. Liquid radioactive waste system test b. Gaseous radioactive waste system test c. Solid radioactive waste system test

6. Radiation control facility
   a. Radiation monitoring system test (process, criticality, effluent, area radiation monitoring system)
   b. Personal monitoring and radiation survey instrument test
   c. Radiation/radioactivity counting device test
   d. Meteorological monitoring system test
   e. Environment monitoring system test

7. Reactor containment facility
   a. Containment local leakage rate test
   b. Containment ventilation system test
   c. Containment spray system test
   d. Containment fan cooler test
   e. Combustible gas control system test

8. Reactor safety system facility
   a. Safety injection tank test
   b. High pressure safety injection system test
   c. Low pressure safety injection system test
   d. Shutdown cooling system test

9. Electric Power system facility
   a. Mechanical and electrical system test of the emergency (standby and alternative) diesel generator
   b. AC power system test
   c. DC power system test
   d. Uninterrupted power supply system test
   e. Generator facility system test
   f. Transformer facility system test
   g. Switch yard facility system test
   h. Reactor trip switch gear test

10. Power conversion system facility
   a. Main steam system test
   b. Steam generator blowdown system test
   c. Main feed water and condensate water system test
   d. Auxiliary feed water system test
   e. Turbine and turbine auxiliary system test
   f. Turbine control fluid system test
   g. Generator auxiliary system test
   h. Auxiliary steam system test

11. Other facilities related to safety of nuclear reactor
   a. Service water system
      1) Essential service water system test
      2) Component cooling water system test
      3) Essential chilled water system

b. Heating, venting and air conditioning (HVAC) system facility
1) Control room HVAC system test
2) Auxiliary building HVAC system test
3) Fuel building HVAC system test
4) Engineered safety features HVAC system test
5) Containment HVAC system test
6) Radioactive waste building HVAC system test

c. Auxiliary system facility
1) Compressed air system test
2) Sampling system test
3) Chemical and volume control system test
4) Fire protection system test
5) Fire detection and alarm system test
6) Diesel generator fuel system test
7) Containment polar crane test
8) Seismic monitoring system test

[Table 3]

Time of Hydrostatic Test and Hot Functional Test

(related to Subparagraph 3 of Article 3 (1))

Inspection of hydrostatic test and hot functional test shall be conducted when tests of the following items are possible.

Hydrostatic test
1. Reactor coolant system hydrostatic test
2. Steam generator secondary side hydrostatic test
Hot functional test
1. Reactor pressure vessel a. Reactor internal vibration assessment test
2. Reactor coolant system facility a. Normal and transient condition vibration test of piping systems b. Expansion and restriction of piping and systems c. reactor coolant pump sealing and cooling function test d. Primary pressure relief system test
3. Instrumentation and control system facility a. Instrument correlation test b. Control element driving mechanism (CEDM) function test c Integrated test of engineered safety feature system d. Inadequate core cooling monitoring system test e. Remote shutdown control panel test
4. Fuel material handling and storage facility (not applicable)
5. Radioactive waste disposal system facility (not applicable)
6. Radiation control facility (not applicable)
7. Reactor containment facility (not applicable)
8. Reactor safety system facility a Safety injection system test b. Shutdown cooling system test
9. Electric power system facility (not applicable)
10. Power conversion system facility a. Main steam system test b. Steam generator blowdown system test c. Main feed water and condensate water system test d. Auxiliary feed water system test e. Turbine and turbine auxiliary system test
11. Other facilities related to safety of nuclear reactor a. Service water system facility b. HVAC system facility (not applicable) c. Auxiliary system facility 1) Sampling system test 2) Chemical and volume control system test

[Table 4]

Time of Initial Fuel Loading and Commissioning Test Inspection

(related to Subparagraph 4 of Article 3 (1))

Inspection of initial fuel loading and commissioning test shall be conducted when tests of the following items are possible.

1. Initial fuel loading
2. Initial criticality test
3. Core performance assessment test
4. Axial xenon oscillation test
5. Moderator temperature reactivity coefficient
6. Rod worth
7. Boron reactivity worth measurement
8. Initial critical boron concentration
9. Power reactivity coefficient assessment and power defect measurement
10. Reactor coolant system flow measurement test
11. Unit load transient test
12. Reactor internal vibration monitoring test
13. Loose part monitoring system test
14. Acoustic leak monitoring system test
15. Reactor coolant pump vibration monitoring system test
16. Reactor coolant system hydrostatic test
17. Pressurizer function test
18. Natural circulation test
19. Post core loading CEDM function test
20. Power ascension test and instrument correlation test
21. Core function test in case of control rod drop and ejection
22. Core protection system test
23. Chemical and radiochemistry tests
24. Neutron and gamma radiation level measuring and shielding capability test
25. Turbine trip test
26. Reactor power cutback system test
27. Plant shutdown from outside control room
28. Loss of off-site power test
29. Load rejection test for each power level
30. Control system checkout test
31. Atmospheric dump valve and steam bypass valve capacity test
32. Main feed water control valve transfer test
33. Main turbine protective function test

[Table 5]

Time of Installation Inspection

(related to Article 3 (2))

Installation inspection of each facility is conducted when installation, welding, non-destructive test and pressure test of the following items are possible.

Facility	System to be inspected	Major component and equipment
1. Reactor pressure vessel	a. Reactor vessel	○ Reactor vessel shell ○ Upper head and its apparatus ○ Fasteners ○ Vessel support structure
	b. Reactor vessel internal structure	○ Upper structure ○ Lower structure ○ Core supporting structure
	c. CEDM	○ CEDM
	d. In-core neutron flux instrumentation and its support	○ In-core neutron flux detector ○ Lower instrumentation tubing and support
2. Reactor coolant system facility	a. Pressurizer	○ Pressurizer ○ Pressurizer relief tank ○ Related piping and valve
	b. Reactor coolant pump	○ Reactor coolant pumps
	c. Steam generator	○ Steam generator
	d. Coolant piping	○ Reactor coolant piping
3. Instrumentation and control facility	a. Instrumentation and control facility	○ Signal detector, processor and signal line ○ Signal processing logic and operating facility ○ Indication and monitoring facility ○ Instrumentation and control cable and cable pat
	b. Control board and cabinet	○ Main control board facility ○ Emergency shutdown control panel facility ○ Cabinet facility
	c. Man-machine interface facility (human engineering)	○ Main control room facility ○ Emergency shutdown facility ○ Safety performance display system
4. Fuel material handling and storage facility	a. Fuel transfer system	○ Fuel transfer car ○ Fuel transfer tube

	b. Fuel handling system	○ Crane, hoist and winches ○ Refueling machine ○ Spent fuel handling machine
	c. Fuel storage system	○ New fuel storage rack ○ Spent fuel storage rack and leak-tight gate
	d. Spent fuel pool cooling and cleanup system	○ Pool cooling and cleanup pump ○ Pool heat exchanger ○ Pool demineralizer, filter, and related piping and valve
5. Radioactive waste disposal facility	a. Liquid radioactive waste processing system	○ Tank ○ Pump ○ Liquid radioactive waste processing equipment /facility ○ Piping/valves ○ Instruments ○ Radioactive drain facility
	b. Gaseous radioactive waste processing system	○ Tank ○ Pump ○ Gaseous radioactive waste processing equipment/facility ○ Piping/valves ○ Instrument
	c. Solid radioactive waste processing system	○ Tank ○ Pump ○ Solid radioactive waste processing equipment/facility ○ Piping/valve ○ Instruments ○ Waste storage facility
6. Radiation control facility	a. Site radiation monitoring system	○ Area radiation monitoring device ○ Process radiation monitoring device ○ Effluent radiation monitoring device
	b. Radiation control facility	○ Access control facility ○ Radioactive sample measuring and analysis laboratory ○ Radioactive contamination protection facility and decontamination equipment ○ Radiation measuring device and health physics equipment ○ Shielding facility

	c. Meteorology monitoring facility	○ Meteorology monitoring sensor ○ Meteorology monitoring recorder ○ Data communication system ○ Meteorological measurement control system ○ Emergency backup power
	d. Environment monitoring facility	○ Environmental radiation survey instrument ○ Environmental radioactivity counting device ○ Data communication system ○ Environment monitoring control system ○ Emergency backup power ○ Laboratory
7. Reactor containment facility	a. Containment combustible gas control system	○ Hydrogen recombiner ○ Hydrogen monitoring system (analyzer) ○ Combustible gas mixing facility ○ Containment ventilation system ○ Multiple penetration
	b. Containment isolation system	○ Containment isolation valves ○ Equipment hatch, and personnel and emergency airlocks ○ Electrical penetrations ○ Other penetration
	c. Containment spray system	○ Containment spray pumps ○ Containment spray chemical additive tanks ○ Spray eductor ○ Spray nozzle and header ○ Related piping and valves
	d. Containment resident heat removal system	○ Containment cooling fan ○ Containment heat exchanger
8. Reactor safety system facility	a. Residual heat removal system	○ Residual heat removal pump ○ Residual heat removal heat exchanger ○ Related piping and valves
	b. Safety injection system	○ Safety injection tanks ○ High pressure and low pressure safety injection pumps ○ Related piping and valves
9. Electric power system facility	a. Off-site power system	○ Switchyard switch gear facility ○ Switchyard protection facility ○ Switchyard power supply facility

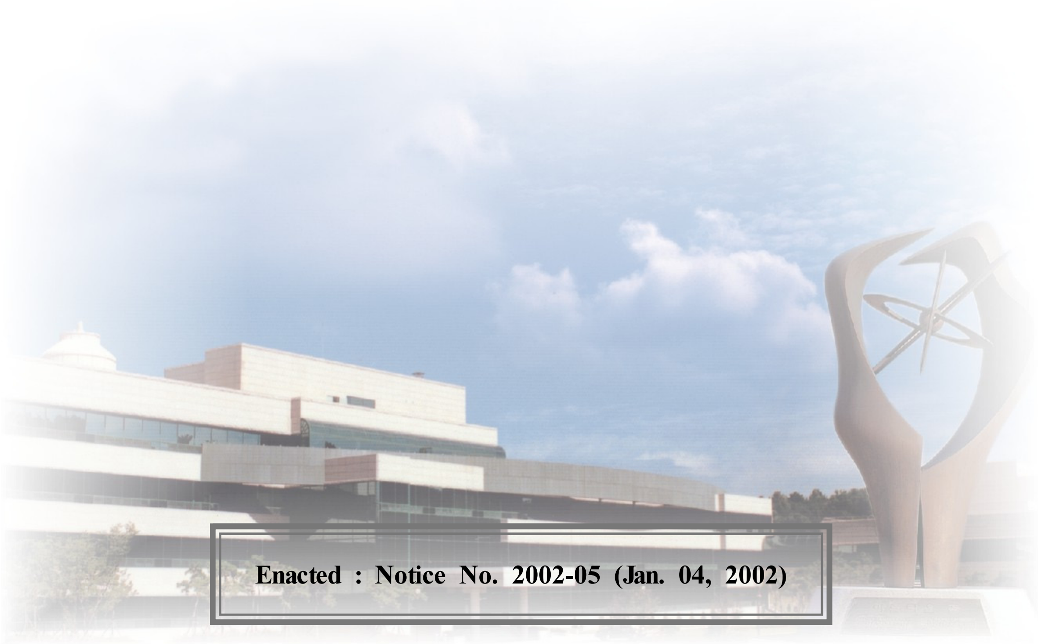
	b. Site AC power system	○ Emergency (standby, alternative) power supply facility ○ Generator facility ○ Transformer facility ○ Switchgear facility ○ Uninterrupted power supply facility ○ Cable and cable path facility
	c Site DC power system	○ Battery and charger facility ○ Distribution panel facility ○ Cable and cable path facility
10. Power conversion system facility	a. Main steam system	○ Main steam line ○ Main steam isolation and bypass valve ○ Flow restrictors ○ Main steam safety valve ○ Main steam atmospheric dump valve ○ Turbine bypass valve ○ Related piping and valves
	b. Steam generator blowdown system	○ Regenerative heat exchanger ○ Non-regenerative heat exchangers ○ Flash tanks ○ High capacity blowdown transfer pumps ○ Filters ○ Ion exchanger ○ Related piping and valves
	c. Feedwater and condensate system	○ Condensate storage tank ○ Condensate water pump ○ Main feed water pump ○ Main condenser ○ Circulating water pump ○ Steam jet air ejector ○ Condensate water demineralizer ○ Main feed water heater ○ Main feed water control and isolation valve ○ Related piping and valves
	d. Auxiliary feed water system	○ Motor driven pumps ○ Turbine driven pumps ○ Related piping and valves

		e. Turbine and turbine auxiliary system	◯ Turbine ◯ Turbine stop and control valves ◯ Turbine control oil pump and tank ◯ Moisture separator reheater ◯ Turbine control and safety facility ◯ Turbine integrity ◯ Turbine lubrication oil pumps and tanks ◯ Moisture separator and exhaust line ◯ Related piping and valves
		f. Generator and related system	◯ Generators (mechanical parts) ◯ Generator cooling system ◯ Related piping and valve
		g. Auxiliary steam system	◯ Boiler ◯ Related piping and valve
11. Other facilities related to safety of nuclear reactor	a.Service water system facility	1) Essential service water system	◯ Essential service water pump ◯ Traveling screen ◯ Related piping and valve
		2) Component cooling water system	◯ Component cooling water pump ◯ Heat exchanger ◯ Surge tank ◯ Chemical addition tank ◯ Related piping and valves
		3) Essential chilled water system	◯ Pump ◯ Chiller ◯ Air Separator ◯ Compression tank ◯ Related piping and valve
	b.HVAC and ventil-ation system facility	1) Main control room HVAC and ventilation system facility 2) Auxiliary building HVAC and ventilation system facility 3) Fuel building HVAC and ventilation system facility 4) Engineered safety facility HVAC and ventilation system facility 5) Containment HVAC and ventilation system facility 6) Radioactive waste building HVAC and ventilation system facility	◯ Blower ◯ HVAC and chiller ◯ Filter ◯ Duct and damper

	c.Auxiliary system facility	1) Compressed air system	○ Air compressor ○ Dryer ○ Service and instrument air piping ○ Storage tank ○ Related piping and valve
		2) Chemical and volume control system	○ Regenerative heat exchanger ○ Letdown heat exchanger ○ Seal water injection heat exchanger ○ Ion exchanger ○ Volume control tank ○ Chemical additive package ○ Boric acid tank ○ Charging pumps and boric acid make-up pump ○ Reactor makeup water pump and storage tank ○ Refueling water storage tank ○ Related piping and valves
		3) Fire protection system	○ Fire barrier ○ Fire suppression facility ○ Fire detection and alarm facility ○ Related piping and valves
		4) Diesel generator fuel storage and transfer system	○ Diesel engine and auxiliary system ○ Diesel generators ○ Fuel oil storage tanks ○ Fuel oil transfer pumps ○ Related piping and valve
		5) Seismic monitoring system	○ Seismic monitoring sensor ○ Seismic monitoring recorder ○ Seismic monitoring control system

【 19 】

Regulation on Schedule for First Periodic Safety Review of Nuclear Power Reactor Facilities



◎ **Notice of the Minister of Science and Technology No.2002-05 (MOST.reactor.028)**

The schedule for the first periodic safety review of nuclear power reactor facilities as provided for in Paragraph 3 of Addenda (July 16, 2001) of the Enforcement Decree of the Atomic Energy Act is hereby publicly notified as follows:

January 4, 2002
Minister of Science and Technology

Regulation on Schedule for First Periodic Safety Review of Nuclear Power Reactor Facilities

Article 1 (Purpose) The purpose of this notice is to set forth the schedule for the first periodic safety review of nuclear power reactor facilities which have been operating for more than 10 years, as of July 17, 2001, from the date on which operating license was issued for such facilities.

Article 2 (Scope of Application) The nuclear power reactor facilities for which the first periodic safety review has to be completed until December 31, 2006 in accordance with Paragraph 3 of Addenda of the Enforcement Decree of the Atomic Energy Act shall be as follows:

Nuclear Facilities (unit)	Date of Operating License or the First Criticality	Remark
Kori 1	1977.06.19	Date of the first criticality
Wolsong 1	1982.11.21	〃
Kori 2	1983.08.10	Date of operating license
Kori 3	1984.09.29	〃
Kori 4	1985.08.07	〃

Nuclear Facilities (unit)	Date of Operating License or the First Criticality	Remarks
Yonggwang 1	1985.12.23	Date of operating license
Yonggwang 2	1986.09.12	〃
Uljin 1	1987.12.23	〃
Uljin 2	1988.12.29	〃

Article 3 (Time Limit of Submission of Review Report) The operator of nuclear power plant shall, after completing periodic safety review in accordance with Article 2, submit the review report to the Minister of Science and Technology on or before the time limit set forth as follows:

Nuclear Facilities (unit)	Time limit of Submission of Review Report	Nuclear Facilities (unit)	Time limit of Submission of Review Report
Kori 1	2002.11.30	Yonggwang 1	2005.06.30
Wolsong 1	2003.06.30	Yonggwang 2	2005.06.30
Kori 2	2003.12.31	Uljin 1	2006.12.31
Kori 3	2004.06.30	Uljin 2	2006.12.31
Kori 4	2004.06.30		

Addenda

1. **(Enforcement Date)** This notice shall enter into force on the date of its promulgation.

2. **(Repeal)** This notice shall be repealed at the time of completion of all the first periodic review under the provision of this notice.

【 20 】

Technical Standards for Investigation and Evaluation of Meteorological Conditions of Nuclear Reactor Facility Sites

Enacted : Notice No. 2003-11 (Jul. 24, 2003)

◎ **Notice of the Minister of Science and Technology No.2003-11 (MOST.reactor.029)**

The Technical Standards for Investigation and Evaluation of Meteorological Conditions of Nuclear Reactor Facility Sites as provided for in Article 6 of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc. are hereby notified publicly as follows:

July 24, 2003
Minister of Science and Technology

Technical Standards for Investigation and Evaluation of Meteorological Conditions of Nuclear Reactor Facility Sites

Chapter 1 General Provisions

Article 1 (Purpose) The purpose of this notice is, in evaluating acceptability of nuclear reactor facility sites, to prescribe technical standards for investigation and evaluation on meteorological phenomena of nuclear reactor facility sites and atmospheric transport and dilution of gaseous effluents in case of radioactive release, as provided for in Article 6 (Meteorological Conditions) of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc.

Article 2 (Definitions) Definitions of the terms used in this notice shall be as follows:

1. The term "maximum snow cover" means the maximum depth of snow cover that is accumulated on the ground during or after snowfall;
2. The term "tornado" means a rotating whirling of air ranging in width from a few meters to hundreds of meters and meteorological phenomena is accompanied by a funnel-shaped downward extension of a cumulonimbus cloud;
3. The term "Fujita-Pearson Scale" means the rates a tornado's intensity based on the damaged area (moving distance and width) and damaged conditions it inflicts. The category is listed in the Table 1;
4. The term "maximum wind speed" means the maximum speed of wind among the 10 minute averaged wind speed;
5. The term "instantaneous maximum wind speed" means the maximum speed of wind among the measured wind speed value; and

6. The term "equivalent rainfall" means the rainfall equivalent to the snowfall when it is melted.

Article 3 (Scope of Application) This notice shall be applied to the evaluation of the acceptability of nuclear reactor facility sites and to the safety assessment of application for construction permit and operating license as provided for in Subparagraph 2 of Article 12 and Subparagraph 2 of Article 22 of the Atomic Energy Act.

Chapter 2 Typhoon

Article 4 (Investigation of Data, etc.) (1) Data investigation of storm caused by typhoon shall be as follows:

1. Data on storm observed onsite and nearby meteorological observation stations (hereinafter referred to as "meteorological observation station") in the latest not less than one year shall be investigated. The data shall be evaluated whether it represents the site's meteorological condition;
2. If the data is proven to be representative, long-term data on annual maximum wind speed and annual instantaneous maximum wind speed measured at 10 meter above the ground nearby meteorological observation station shall be investigated; and
3. If the data of Subparagraph 1 is not proven to be representative, long-term data of nearby meteorological observation station which is most-relevant to the site may be used with conservative correction.

(2) Data investigation of heavy rainfall caused by typhoon shall be as follows:

1. The data on heavy rainfall observed onsite and nearby meteorological observation stations in the latest not less than one year shall be investigated. The data shall be evaluated whether the data represent the site's meteorological condition;
2. If the data is proven to be representative, long-term data on maximum hourly precipitation and maximum precipitation in 24 hours measured nearby meteorological observation station shall be investigated; and
3. If the data of Subparagraph 1 is not proven to be representative, long-term data of nearby meteorological observation station which is most-relevant to the site may be used with conservative correction.

(3) Data investigation of storm caused by other than typhoon (cyclonic storm) shall be subject to the foregoing Paragraph 1.

Article 5 (Method of Analysis) (1) Analysis of storm caused by typhoon shall be as follows:

1. In analyzing the frequency of storm's occurrence, 100-year return period maximum wind speed and instantaneous maximum wind speed shall be analyzed using the data measured for more than 30 years. In inevitable case, the data measured less than 30 years may be used. In case of using the data less than 30 years, the evident reason of the data choice shall be presented, and the data shall be compared with nationwide data in order to minimize the uncertainties; and
2. Analysis of the data according to Subparagraph 1 shall be implemented using appropriate statistical method or numerical model considering the site's climatological and topographical characteristics.

(2) Analysis of heavy rainfall caused by typhoon shall be as follows:

1. In analyzing the frequency of heavy rainfall's occurrence, maximum hourly precipitation and maximum precipitation in 24 hours, which is anticipated for a relevant time period, shall be analyzed using the data measured for more than 30 years. In inevitable case, the data measured less than 30 years may be used. In case of using the data less than 30 years, the evident reason of the data choice shall be presented, and the data shall be compared with nationwide data in order to minimize the uncertainties; and
2. Analysis of the data according to Subparagraph 1 shall be implemented using appropriate statistical method or numerical model considering the site's climatological and topographical characteristics.

(3) The data analysis of storm caused by other than typhoon (cyclonic storm) shall be subject to the foregoing Paragraph 1.

Chapter 3 Heavy Snow

Article 6 (Investigation of Data, etc.) (1) The data on heavy snow observed onsite and nearby meteorological observation stations during the same period in the latest not less than one year shall be investigated. The data shall be evaluated whether it represents the site's meteorological condition.

(2) If the data is proven to be representative, long-term data on yearly maximum snow cover, density and 48-hour maximum winter precipitation measured nearby meteorological observation station shall be investigated and converted into equivalent rainfall.

(3) If the data of Subparagraph 1 is not proven to be representative, long-term data

of nearby meteorological observation station which is most-relevant to the site may be used with conservative correction.

Article 7 (Method of Analysis) (1) In analyzing heavy snow, equivalent rainfall relevant to 100-year return period maximum snow cover and the 48-hour probable maximum winter precipitation shall be analyzed using the data measured for more than 30 years. In inevitable case, the data measured less than 30 years may be used. In case of using the data less than 30 years, the evident reason of the data choice shall be presented, and the data shall be compared with nationwide data in order to minimize the uncertainties.

(2) Analysis of data according to Paragraph 1 shall be implemented using appropriate statistical method or numerical model considering the site's climatological and topographical characteristics.

Chapter 4 Heavy Rainfall

Article 8 (Investigation of Data, etc) Data investigation of heavy rainfall caused by other than typhoon shall be subject to Article 4 (2).

Article 9 (Method of Analysis) Data analysis of heavy rainfall caused by other than typhoon shall be subject to Article 5 (2).

Chapter 5 Tornado

Article 10 (Investigation of Data, etc.) (1) The following data shall be investigated to evaluate effects of tornado on the proposed site and its nearby area:

1. Record on the past occurrence of the tornado and its similar phenomena;
2. Record on the damaged area (moving distance and width) and damaged results; and
3. Record on the tornado's intensity (maximum wind speed, maximum rotational wind speed, maximum translational speed, maximum rotational radius and maximum pressure differential, etc.).

(2) If the data obtained from Paragraph 1 is insufficient, it shall be classified as Fujita-Pearson Scale according to the Table 1 using the data in Subparagraphs 1 and 2 of Paragraph 1.

Article 11 (Method of Analysis) (1) Based on the data which was evaluated in accordance with Article 10, potential maximum wind speed and pressure differential of tornado, and the influence of dispersed material thereby shall be analyzed using the appropriate analysis model.

(2) If the data on the tornado is insufficient, tornado's intensity shall be analyzed by comparing the damage caused by a tornado which was known as the greatest with the Fujita-Pearson Scale.

Chapter 6 Onsite Meteorological Measurement Program

Article 12 (Meteorological Measurement) (1) The following parameters shall be measured at the place that represents the site:

1. Wind direction and wind speed;
2. Temperature;
3. Atmospheric stability;
4. Precipitation;
5. Humidity; and
6. Mixing height.

(2) Accuracy, installation location and height of instruments which are required to measure the parameters in Paragraph 1 shall follow the Table 2.

Article 13 (Meteorological Measurement Period) (1) Meteorological measurement period as provided for in Article 12 (1) shall be the latest not less than one year for construction permit, and the latest not less than two years for operating license.

(2) Data recovery rate for the measurement period in Paragraph 1 shall be not less than 90%.

Article 14 (Accuracy Maintenance) To maintain accuracy of the measurement sensors as provided for in Article 12 (1), meteorological measurement systems shall be calibrated every six months.

Article 15 (Data Processing) (1) Data processing of wind speed, wind direction and data processing of temperature differential and standard deviation of wind direction for the calculation of atmospheric stability shall be averaged for 10-minute using more than 180 times instantaneous measured values (3-second interval).

(2) Atmospheric stability classification shall be based on the temperature differential calculated from temperature differences between 10 m above ground and release

height (ΔT method). Provided, that if the atmospheric condition is either unstable or neutral, or if wind speed exceeds 1.5 m/sec, standard deviation of wind direction method ($\sigma\Theta$ method) may be used to classify the atmospheric stability. Also, other method (ex, Bulk Richardson Number, etc.) may be used if the application suitability is proven considering site characteristics.

(3) Classification of atmospheric stability as provided for in Paragraph 2 is listed in Table 3.

(4) Hourly mixing height shall be calculated based on upper meteorological measurement more than one day a season and more than 4 times a day.

(5) Data processing of temperature, precipitation and humidity shall follow the Korean Meteorological Administration's surface meteorological measurement method.

Article 16 (Special Meteorological Measurement and Analysis) (1) The following additional parameters shall be measured dividing into surface meteorological measurement and upper meteorological measurement, if the topography and the difference of surface's physical conditions, due to site location in complex region or sea-shore, may effect the diffusion and the dilution of the released gaseous radioactive materials. Effects of characteristics of sea-land breeze or land breeze shall also be analyzed:

1. Wind direction and wind speed;
2. Temperature;
3. Humidity; and
4. Sea surface temperature.

(2) The accuracy, installation location and height of the instruments for measurement of items as provided for in the Paragraph 1 are listed in Table 4.

(3) Special meteorological measurement shall be implemented more than 2 days a season in 3 hour interval.

Chapter 7 Diffusion Characteristics of Radioactive Materials

Article 17 (Evaluation of Diffusion and Dilution) Diffusion and dilution characteristics of gaseous radioactive materials shall be evaluated as follows, using the appropriate diffusion and dilution model considering climatological and topographical characteristics of the site.

1. Meteorological data

   a. Atmospheric diffusion in potential accident shall be evaluated using onsite meteorological data which was measured at 10 m above the ground in case of a ground release, and at height representing the actual release point in case of elevated release.

b. Atmospheric diffusion in routine release shall be evaluated using onsite meteorological data which was measured at height representing the actual release point.

2. Atmospheric diffusion coefficient

a. The applicability of lateral and vertical diffusion coefficient (σ_{y}, σ_{z}) shall be reviewed considering characteristics of topographical and meteorological conditions of the site.

b. Atmospheric diffusion coefficient shall be corrected considering building wake effects and meandering effects due to neighborhood buildings.

3. Type of release

a. When release height of radioactive materials is lower than twice the height of adjacent structures, as in case of vent or penetration, it shall be assumed ground release.

b. When release height of radioactive materials is higher than twice the height of adjacent structures, as in case of stack, it shall be assumed elevated release.

c. In case height of the actual release is to be considered, in spite of the fact that the release height of radioactive materials is lower than twice the height of adjacent solid structures, its suitability shall be proved through wind tunnel experiment or field diffusion experiment considering natural conditions of the site.

4. Atmospheric diffusion factor (χ/Q) in potential accident

a. Atmospheric diffusion factor for potential accident shall be calculated by appropriate distances (including exclusion area boundary, nearest residence zone boundary, emergency planing zone boundary and outer boundary of low population zone) and by appropriate time periods (including 0-2 hour time period, causing the largest exposure after an accident).

b. Atmospheric diffusion factor for potential accident shall be determined by whichever is larger between the following two values; maximum 99.5 percentile χ/Q value from each sector and 95 percentile χ/Q value for the entire site regardless of direction.

5. Atmospheric diffusion factor (χ/Q) for routine release

a. Atmospheric diffusion factor for routine release shall be calculated by appropriate distances (for each sector, to a distance of 80 km including exclusion area boundary, nearest residence zone boundary, emergency planing zone boundary and outer boundary of low population zone).

Addenda

Article 1 (Enforcement Date) This notice shall enter into force on the date of its promulgation.

Article 2 (Transitional Measure) Investigation and evaluation of the meteorological conditions of existing nuclear reactor facility sites conducted in accordance with Notice of the MOST No.2000-08 "Technical Standards for Locations, Structures and Equipment of Nuclear Reactor Facilities" shall be regarded as conducted under this notice.

Article 3 (Revisions of Other Notices) At the time this notice enters into force, Order 3 "Guidelines for atmospheric conditions of the site and surrounding area" and 2), 3) of Order 6 "Guidelines for investigating and evaluating nautral phenomena of the site and the surrounding area" in the Table 1 "Technical Standards for Locations of Nuclear Reactor Facilities" of Notice of the MOST No.2000-08 "Technical Standards for Locations, Structures and Equipment of Nuclear Reactor Facilities" shall be deleted.

[Table 1]

Fujita-Pearson Scale

F-Category	Maxim Wind Speed (m/sec)	Moving Distance (km)	Moving Width (m)
F0	〈33	〈1.6	〈16
F1	33 ~ 49	1.6 ~ 5.0	16 ~ 50
F2	50 ~ 69	5.1 ~ 16.0	51 ~ 110
F3	70 ~ 92	16.1 ~ 50.9	161 ~ 509
F4	93 ~ 116	51.0 ~ 160.0	510 ~ 1600
F5	117 ~ 140	160.1 ~ 507.0	1601 ~ 5070
〉F5	〉140	〉507.0	〉5070

[Table 2]

Instruments' Accuracy and Installation Location

Parameter	Starting Threshold	Accuracy	Installation Height
Wind direction	$U^{(1)} \leq 0.5$ m/s	±5˚	10m above the ground and release height(2)
Wind speed	$U^{(1)} \leq 0.5$ m/s	0.2 m/s ($U^{(1)}$ 〈 2m/s) 10% ($U^{(1)} \geq$ 2m/s)	10m above the ground and release height(2)
Temperature	-	±0.5℃, ΔT: ±0.1℃	1.5m above the ground, in ΔT above 10m and release height(2)
Precipitation	0.5mm	±10%	ground
Humidity	-	±5%	1.5m above the ground

(1) U represents wind speed.

(2) If the meteorological tower is like a transmission tower, wind speed·wind direction·temperature instruments shall be installed at the end of mounting object, which should have enough length not to be affected by the transmission tower, with a dominant prevailing wind direction.

[Table 3]

Classification of Atmospheric Stability

Atmospheric State	**Pasquill Categories**	**Temperature Change with Height (ΔT/Δz, ℃/100m)**	**σΘ(1)(deg)**	**Rb(2)**
Extremely unstable	A	ΔT/Δz ≤ -1.9	22.5 ≤ σΘ	Rb ≤ -0.35
Unstable	B	-1.9 〈 ΔT/Δz ≤ -1.7	17.5 ≤ σΘ 〈 22.5	-0.35 〈 Rb ≤ -0.18
Slightly unstable	C	-1.7 〈 ΔT/Δz ≤ -1.5	12.5 ≤ σΘ 〈 17.5	-0.18 〈 Rb ≤ -0.04
neutral	D	-1.5 〈 ΔT/Δz ≤ -0.5	7.5 ≤ σΘ 〈 12.5	-0.04 〈 Rb ≤ 0.01
Slightly stable	E	-0.5 〈 ΔT/Δz ≤ 1.5	3.8 ≤ σΘ 〈 7.5	0.01 〈 Rb ≤ 0.07
Stable	F	1.5 〈 ΔT/Δz ≤ 4.0	2.1 ≤ σΘ 〈 3.8	0.07 〈 Rb ≤ 0.13
Extremely stable	G	4.0 〈 ΔT/Δz	σΘ 〈 2.1	0.13 〈 Rb

(1) Standard deviation of horizontal wind direction fluctuation over a period of 10 minutes.

(2) Bulk Richardson Number : $Rb = gz(\Theta_z - \Theta_s)/(T_0 \overline{U}^2)$

where, g : gravitational acceleration ($9.8 m/sec^2$),

z : measurement height (10m),

Θ_z : Potential temperature above 10m (or Temperature),

Θ_s : Potential temperature above 1.5m (or Temperature),

T_0 : Average temperature between 2 layers,

$\overline{U}$: Average wind speed above 10m.

[Table 4]

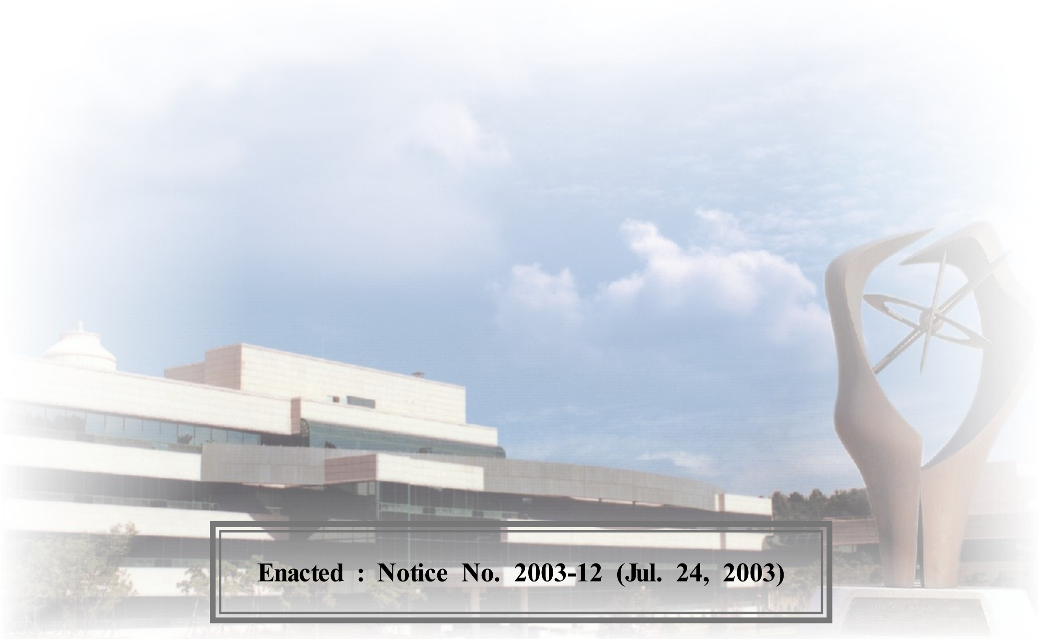
Special Measurement Instruments' Accuracy and Installation Location

Parameter	Starting Threshold	Accuracy	Installation Location and Height
Surface wind			
Wind direction	$U^{(1)} \leq 0.5$ m/s	±5°	representative place of the off-site region's characteristics, 10m above the ground
Wind speed	$U^{(1)} \leq 0.5$ m/s	0.2 m/s ($U^{(1)} < 2$m/s) 10 % ($U^{(1)} \geq 2$m/s)	
Upper wind			
Wind direction	$U^{(1)} \leq 0.5$ m/s	±5°	
Wind speed	$U^{(1)} \leq 0.5$ m/s	0.2 m/s ($U^{(1)} < 2$m/s) 10 % ($U^{(1)} \geq 2$m/s)	above the site (below 1km)
Temperature	-	±0.5℃	above the site (below 1km)
Humidity	-	±5%	above the site (below 1km)

(1) 'U' represents wind speed.

【 21 】

Technical Standards for Investigation and Evaluation of Hydrological and Oceanographic Characteristics of Nuclear Reactor Facility Sites



Enacted : Notice No. 2003-12 (Jul. 24, 2003)

◎ **Notice of the Minister of Science and Technology No.2003-12 (MOST.reactor.030)**

The Technical Standards for Investigation and Evaluation of Hydrological and Oceanographic Characteristics of Nuclear Reactor Facility Sites in accordance with Article 7 of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc. are hereby notified publicly as follows:

July 24, 2003
Minister of Science and Technology

Technical Standards for Investigation and Evaluation of Hydrological and Oceanographic Characteristics of Nuclear Reactor Facility Sites

Chapter 1 General Provisions

Article 1 (Purpose) The purpose of this notice is to prescribe details as regards investigation and evaluation of hydrological and oceanographic conditions of nuclear reactor facility sites as provided for in Article 7 (Hydrological and Oceanographic Conditions) of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc.

Article 2 (Definitions) The terms used herein shall be defined as follows:

1. The term "probable maximum precipitation (PMP)" means the maximum possible precipitation in a certain area, due to a precipitation under most severe meteorological conditions;
2. The term "probable maximum flood (PMF)" means a flood due to a probable maximum precipitation (PMP);
3. The term "storm surge" means an extraordinary piling up of water in shallow depths due to wind stress and bottom friction, together with the atmospheric pressure reduction which occurs in conjunction with severe storms; and
4. The term "tsunami" means a wave train produced by impulsive disturbances of a body of water caused by displacements associated with submarine earthquakes, volcanic eruptions, submarine slumps or shoreline landslides, of which the duration is from a few minutes to tens of minutes.

Article 3 (Scope of Application) This notice shall apply to evaluating the safety of construction permit or operating license of nuclear reactor and related facilities (hereinafter referred to as "reactor facilities") as provided for in Subparagraph 2 of Article 12 and Subparagraph 2 of Article 22 of the Atomic Energy Act.

Chapter 2 Inland Flooding

Article 4 (Direct Rainfall on Site) (1) Data survey for evaluation of effects by the direct rainfall on the site shall comply with the followings:

1. Area and topography, layout planning and present status of the site shall be investigated; and
2. Related parameters including discharge characteristics or surface flow patterns of rainfall shall be investigated.

(2) Information on probable maximum precipitation including assessment criteria and analysis technique shall be presented.

(3) Evaluation of inundation effects on safety related structures shall be executed.

Article 5 (River Flood) (1) Data survey for evaluation of effects by river flooding shall comply with the followings:

1. Principal historical flood data at the site and its vicinity shall be investigated;
2. Related parameters including cross sections of upstream and downstream rivers, channel gradient, river basin or discharge characteristics shall be investigated; and
3. Related parameters including maximum storage of dams or cross sections of upstream rivers shall be investigated.

(2) In case that the site may be affected by river flooding, evaluation of river flooding effects shall be executed in pursuance of the followings:

1. Evaluation of PMP including the analysis of rainfall process and runoff process at the river system of the site and its vicinity shall be executed; and
2. Evaluation of PMF at the site and its vicinity, and of probable maximum flooding level at the site caused by PMP shall be executed.

(3) Evaluation of flooding effects caused by failure of upstream reservoirs and dams shall be executed using the proper model in consideration of the hydrological and topographical feature of the site.

(4) In case of the evaluation in Paragraph 3, evaluation of considerations including the failure modes of dams at the river system of the site and its vicinity, and the

hydraulic structures shall be executed. And evaluation of probable maximum flooding level at the site including flood wave effects caused by dam failures, failures of downstream dams, and multiple dam failures shall be executed.

(5) If it is proven that the failures of reservoirs or dams which are built or to be built upstream of the river in the vicinity of the site are not probable, the evaluation in Paragraph 3 may be omitted.

(6) The evaluation of flooding effects at the site and its vicinity shall be executed including the cases where more than one causes for flood are combined.

(7) Ice flooding and blockage data at the site and its vicinity including ice jams, wind-driven ice ridges, floes and so forth shall be investigated. Evaluation of ice flooding and blockage effects on safety-related facilities and water supplies shall be properly executed.

(8) Potential channel diversions at the site and its vicinity shall be investigated. Evaluation of channel diversion effects on safety-related facilities and water supplies shall be executed.

Chapter 3 Coastal Flooding

Article 6 (Data Investigation) Data survey for coastal flooding on the coastal site and its vicinity shall comply with the followings:

1. Historical coastal flooding data at the site and its vicinity shall be investigated;
2. Detailed bathymetry and coastline configuration at the site and its vicinity shall be investigated; and
3. All phenomena related with sea water levels including sea level datum, shoaling effect, refraction, wave breaking at the site and its vicinity shall be investigated.

Article 7 (Storm Surges) (1) In the coastal sites, data survey for evaluation of storm surges by the stochastic methods shall comply with the followings:

1. Data on storm surges observed at the site and its vicinal observation stations in the latest more than one year shall be investigated. And the data shall be proven representative as the site characteristics;
2. If the data is proven to be representative, long-term data on storm surges at the vicinal observation stations shall be investigated; and
3. If the data from Subparagraph 1 is not proven to be representative, most-relevant long-term data on storm surges at vicinal observation stations shall be revised for the conservative evaluation of storm surges.

(2) Long-term data of more than 30 years shall be used for the evaluation of storm surges by the stochastic methods. The evaluation of maximum storm surge runup shall be worth due consideration of breaking or non-breaking in the near-shore of site. In case of using the data less than 30 years, the evident reason of data choice shall be presented, and the data shall be compared with nationwide data in order to minimize the uncertainties.

(3) For the evaluation in Paragraph 2, evaluation of storm surges shall be executed through the appropriate stochastic methods in consideration of the climatological and topographical features of the region.

(4) Data survey for evaluation of storm surges using the numerical models shall comply with the needed data of attached Table 1.

(5) For the evaluation using the numerical models, suitability of models and pertinency of model to the region shall be proven.

(6) The evaluated data of surges and seiches by each return period at the site shall be included.

Article 8 (Tsunamis) (1) Data survey for evaluation of tsunamis using the numerical models shall include analyses of relevant data prescribed in the attached Table 2, according to the features of models.

(2) For the evaluation using the numerical models, evaluation of tsunami runup and drawdown shall be executed considering characteristics of tsunami sources, propagation, wave deformation by shoaling, refraction, reflection, diffraction and so forth.

(3) For the evaluation using the numerical models, suitability of models and pertinency of model to the region shall be proven.

(4) The evaluated data of tsunamis and seiches by each return period at the site shall be included.

Article 9 (Tidal Levels) (1) In the coastal sites, data survey for evaluation of sea levels by the stochastic methods shall comply with each of the followings:

1. Data on sea levels at the site and its vicinal observation stations shall be investigated. And the data shall be proven representative as the site characteristics;
2. If the data is proven to be representative, long-term data on sea levels at the observation stations shall be investigated; and

3. If the data from Subparagraph 1 is not proven to be representative, long-term data on seawater levels at the vicinal observation stations shall be revised for the conservative evaluation of sea levels.

(2) Principal harmonic component related with tide and sea level data including datum level shall be evaluated.

Article 10 (Wave Activities) (1) In the coastal sites, data survey for evaluation of wave activities by the stochastic methods shall comply with the followings:

1. Data on wave activities observed at the site and its vicinal observation stations in the latest more than one year shall be investigated. And the data shall be proven representative as the site characteristics;
2. If the data is proven to be representative, long-term data on seawater levels at the observation stations shall be investigated; and
3. If the data from Subparagraph 1 is not proven to be representative, long-term data on wave activities at the vicinal observation stations shall be revised for the conservative evaluation of wave activities.

(2) For the evaluation of rising by the wave runup, evaluation of wave activities including wave spectrum analysis data, deep water design wave, shallow wave and runup shall be executed, and evaluation of resonance by artificial structures including breakwaters shall be executed.

Article 11 (Combinations of Flood-causing Events) (1) The combinations of more than one flood-causing events shall be considered for the evaluation of flooding effects at the site and its vicinity.

(2) For the combinations of flood-causing events in Paragraph 1, conservative evaluation of flooding effects at the site shall be executed in order to ensure the safety against coastal flooding.

(3) Sea water level at the site shall be observed constantly and evaluation of flooding effects at the site shall be executed periodically.

Chapter 4 Features of Radioactive Material Release

Article 12 (Assessment of Release into Surface Water) (1) Data on streams including flow velocity, flow rate, inflow, outflow, evaporation, infiltration and so forth at the site and its vicinity shall be investigated monthly in the latest more than one year.

And the data shall be proven representative as the site characteristics.

(2) For assessment of radioactive material release into the streams and evaluation of effects to the surface water users, possible route shall be estimated considering the hydrological and topographical feature of the site. The representative site data as provided for in Paragraph 1 and the proper model shall be used for radioactive material transport analysis.

(3) For evaluation in Paragraph 2, conservative site-specific hydrological parameters including dispersion, dilution factor, adsorption and sedimentation coefficient, travel time per unit distance and so forth shall be used.

(4) In case it is proven that engineering preparedness is ready for prevention of release into the streams of radioactive material or reducing the risk concerned, the evaluation in Paragraph 2 may be omitted.

Article 13 (Assessment of Release into Groundwater) (1) Data survey for the release assessment into the groundwater of radioactive material shall comply with the followings:

1. Data on groundwater including flow velocity, flow rate, hydraulic gradient, porosity, groundwater table, hydraulic conductivity and so forth at the site and its vicinity shall be investigated by the boring test, the piezometric measurement or the pumping test monthly in the latest more than one year. And the data shall be proven representative as the site characteristics;
2. Adsorption coefficients by radionuclides of principal rocks shall be investigated. And the data shall be proven representative as the site characteristics;
3. Regional and local aquifers, sources and sinks shall be investigated including the information on groundwater distribution, and the origin and evidence of data; and
4. Groundwater use data such as wells, pumping methods and storage structures at the site and its vicinity, and water requirement of the plant shall be investigated including the information on the origin and evidence of the data.

(2) For assessment of radioactive material release into the groundwater and evaluation of effects to the groundwater users, possible route shall be estimated considering the hydrological and geological feature of the site. The representative site data as provided for in Paragraph 1 and the proper model shall be used for radioactive material transport analysis.

(3) For evaluation in Paragraph 2, conservative site-specific hydrological parameters including dispersion, dilution factor, adsorption and sedimentation coefficient, travel time per unit distance and so forth shall be used.

(4) Groundwater monitoring program including procedures, methods and schedules of monitoring for the protection of the present and future groundwater users shall be established and executed considering the site specific characteristics.

(5) In case it is proven that engineering preparedness is ready for prevention of release into the groundwater of radioactive material or reducing the risk concerned, the evaluation in Paragraph 2 may be omitted.

Article 14 (Assessment of Release into Seawater) (1) Data on seawater flow characteristics including tide, currents, tidal currents, wind waves and so forth, and physical parameters including temperature, salinity and so forth at the site and its vicinity shall be investigated per layer seasonally in the latest more than one year. And the data shall be proven representative as the site characteristics.

(2) For assessment of radioactive material release into the seawater and evaluation of effects to the seawater users, possible route shall be estimated considering the ocean environmental and topographical feature of the site. The representative site data as provided for in Paragraph 1 and the proper model shall be used for radioactive material transport analysis.

(3) For evaluation in Paragraph 2, conservative site-specific hydrological parameters including dispersion, dilution factor, adsorption and sedimentation coefficient, travel time per unit distance and so forth shall be used.

(4) In case it is proven that engineering preparedness is ready for prevention of release into the seawater of radioactive material or reducing the risk concerned, the evaluation in Paragraph 2 may be omitted.

Article 15 (Assessment of Release into Lakes, Reservoirs, Estuaries, etc.) (1) Data on lakes, reservoirs, estuaries and so forth at the site and its vicinity shall be investigated monthly in the latest more than one year. And the data shall be proven representative as the site characteristics.

(2) For assessment of radioactive material release into the lakes, reservoirs, estuaries, etc. and evaluation of effects to the water users, possible route shall be estimated considering the hydrological and topographical feature of the site. The representative site data as provided for in Paragraph 1 and the proper model shall be used for radioactive material transport analysis.

(3) For evaluation in Paragraph 2, conservative site-specific hydrological parameters including dispersion, dilution factor, adsorption and sedimentation coefficient, travel

time per unit distance and so forth shall be used.

(4) In case it is proven that engineering preparedness is ready for prevention of release into the lakes, reservoirs, estuaries, etc. of radioactive material or reducing the risk concerned, the evaluation in Paragraph 2 may be omitted.

Chapter 5 Water Supply

Article 16 (Industrial Water) (1) Data survey for industrial water sources shall comply with the followings:

1. Data on low water levels of vicinal rivers as water sources shall be investigated; and
2. Related parameters including dam storage, inflow and so forth shall be investigated monthly in the latest more than one year. And the data shall be proven representative as the site characteristics.

(2) Water supply capacity for nuclear facilities shall be evaluated considering the hydrological and topographical feature of the site and its vicinity.

(3) For evaluation in Paragraph 2, necessary cooling water, water supply condition and so forth shall be considered conservatively in order to ensure the safety of nuclear facilities.

(4) The effects by the projected water control structures shall be described and evaluated.

(5) Historical and potential ice accumulations at the site and its vicinity shall be investigated, and the appropriate evaluation of ice effects shall be executed.

(6) Historical and projected channel diversions at the vicinal rivers shall be investigated, and the appropriate evaluation of channel diversions shall be executed.

Article 17 (Ultimate Heat Sink) (1) Related parameters including sea level, flow characteristics of sea such as tide, ocean currents, tidal currents, waves, and so forth at the site and its vicinity shall be investigated seasonally in the latest more than one year. And the data shall be proven representative as the site characteristics.

(2) Data survey for water temperatures by the stochastic methods shall comply with the followings:

1. Water temperatures at the site and its vicinal observation stations shall be investigated in the latest more than one year. And the data shall be proven representative as the site characteristics;

2. If the representative of data is proven, long-term water temperature data at the observation stations shall be investigated; and
3. If the representative of data is not proven, long-term water temperature data at the observation stations shall be revised for the conservative evaluation of water temperature data.

(3) Water supply capacity for nuclear facilities shall be evaluated considering the coastal and topographical feature of the site and its vicinity.

(4) For evaluation in Paragraph 3, necessary cooling water, water supply condition and so forth shall be considered conservatively in order to ensure the safety of nuclear facilities.

(5) Minimum water levels by storm surges, seiches, tsunamis and so forth at the site and its vicinity shall be evaluated.

(6) Historical low water conditions at the site and its vicinity shall be investigated.

(7) The effects by the projected water control structures shall be described and evaluated.

(8) Estimated minimum water level shall be compared with the elevation of safety-related cooling water supply facilities at the site. If necessary, the appropriate countermeasures shall be established in order to ensure the performance of the facilities.

(9) Capability of the ultimate heat sink to supply sufficient cooling water under normal and emergency conditions shall be evaluated.

(10) Historical and potential ice accumulations at the site and its vicinity shall be investigated, and the appropriate evaluation of ice effects shall be executed.

(11) Hydraulic design basis and safety of channels and reservoirs for cooling water shall be evaluated.

Chapter 6 Protection Measures

Article 18 (Engineering Preparedness) (1) Estimated maximum flood level of inland and coastal flooding shall be compared with the site ground level or elevation of safety-related facilities. If necessary, the appropriate flood protection plan shall be established, and the protection barriers shall be installed for the protection of safety-related structures, systems, and components.

(2) Design basis on the groundwater pressure of the buried safety-related structures, systems, and components as well as the safety of the permanent dewatering system shall be evaluated.

Article 19 (Emergency Procedures) In order to ensure the safety of the site against flooding and the safety functions of the safety-related structures, if necessary, relevant technical specifications for operation and emergency procedures shall be established and implemented.

Addenda

Article 1 (Enforcement Date) This notice shall enter into force on the date of its promulgation.

Article 2 (Revisions of Other Notice) At the same time of enforcement of this notice, Order 4 "Guidelines for investigating and evaluating hydrologic characteristics of the site and the surrounding area" and Order 6 1), 4) "Guidelines for investigating and evaluating natural phenomena of the site and the surrounding area" of Table 1 "Technical Standards for Location of Nuclear Reactor Facilities" of Notice of the Minister of Science and Technology No.2000-08 "Technical Standards for Locations, Structures and Equipment of Nuclear Reactor Facilities" shall be abrogated.

[Table 1]

Investigation Factors of Storm Surges for Numerical Model

Classification of data	**Investigation factors**
1. Data on storms of the region	○ Minimum central pressure and associated peripheral pressure ○ Storm track ○ Maximum sustained wind speed and its radius ○ Duration of storm and associated winds ○ Wind fetch ○ Direction and speed of movement of storm ○ The closest point to the coast
2. Local data in the vicinity of the site	○ Sea level datum ○ Seabed friction coefficients ○ Wind stress coefficients ○ Bathymetry of offshore area ○ Coastline configuration

[Table 2]

Investigation Factors of Tsunamis for Numerical Model

Classification of data	**Investigation factors**
1. Seismic and geological data for seabed movement	○ Earthquake magnitude ○ Maximum vertical and horizontal ground displacement ○ Source length and width ○ Source orientation, shape, focal depth and epicenter location ○ Decay of displacement with distance from fault ○ Time-dependant seabed movement (source, time history, fault rupture pattern, etc.)
2. Data on tsunami propagation	○ Initial form of water surface displacement ○ Bathymetry and boundary condition of the ocean
3. Local data in the vicinity of the site	○ Sea level datum ○ Shoaling ○ Refraction ○ Coastline configuration ○ Wave breaking

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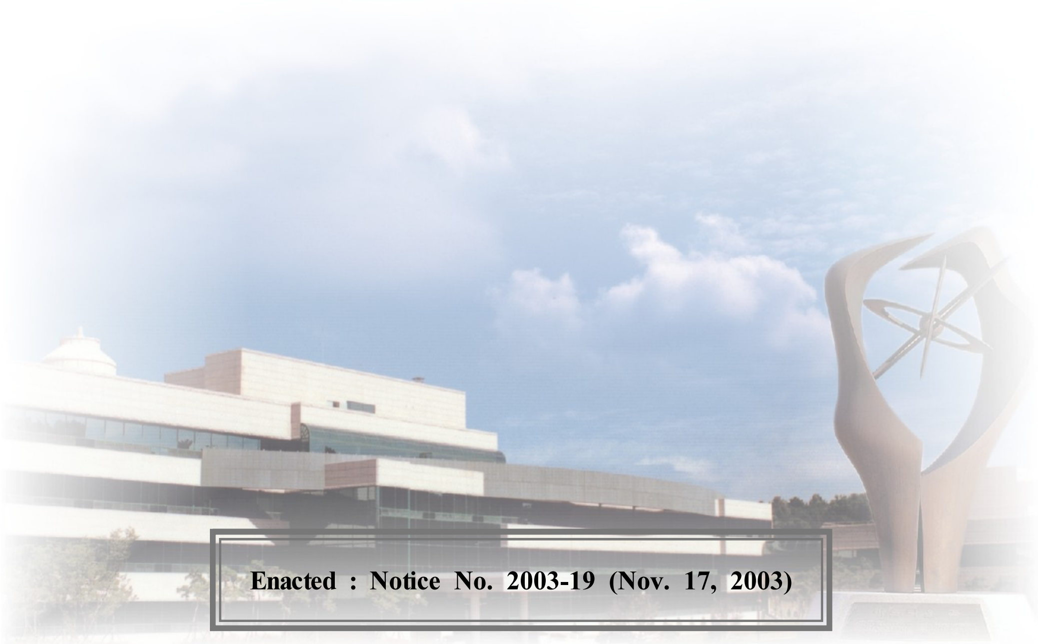
Regulation on Establishment and Implementation of Fire Protection Program



Enacted : Notice No. 2003-19 (Nov. 17, 2003)

◎ **Notice of the Minister of Science and Technology No.2003-19 (MOST.reactor.031)**

The Regulation on Establishment and Implementation of Fire Protection Program as provided for in Article 59 of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc. is hereby notified publicly as follows:

November 17, 2003
Minister of Science and Technology

Regulation on Establishment and Implementation of Fire Protection Program

Article 1 (Purpose) The purpose of this notice is to specify the regulation on establishment and implementation of fire protection program in accordance with Articles 29 (1) and 36 of the Atomic Energy Act, Article 102 of the Enforcement Decree of the Act and Article 59 of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc.

Article 2 (Scope of Application) This notice shall apply to an operator of a nuclear power reactor (hereinafter referred to as "reactor operator") and its related facilities (hereinafter referred to as "reactor facilities") for which necessary measures shall be taken in order to maintain the ability to perform reactor safe shutdown functions, and minimize radioactive releases to the environment.

Article 3 (Definitions) Definitions of the terms used in this notice shall be as follows:

1. The term “noncombustible material” means a material which, in the form in which it is used and under the conditions anticipated, will not ignite, burn, support combustion, or release flammable vapors when subjected to fire or heat;
2. The term “combustible material” means a material that does not meet the definition of noncombustible;
3. The term “fire barrier” means those components of construction (walls, floors, and their supports), including beams, joists, columns, penetration seals or closures, fire doors, and fire dampers that are rated by accredited laboratories in hours of resistance to fire and are used to prevent the spread of fire;

4. The term “fire area” means a building or part of a building comprising one or more rooms or spaces, that is separated from other areas by fire barriers, constructed to prevent the spread of fire to or from the remainder of the building for a given period of time;
5. The term “fire zone” means a subdivision of a fire area in which fire separation between items important to safety is provided by limitation combustible materials, spatial separation, or fixed fire extinguishing systems;
6. The term “design basis fire” means those fires that may develop in local areas, assuming that no manual, automatic, or other fire-fighting action has been initiated, and that have passed flashover and have reached its peak burning rate. It is used as a measure to anticipate the maximum damage of the fire area;
7. The term “fire hazard analysis” means qualitative and/or quantitative hazard analysis to investigate potential in situ and transient fire hazards in each fire area and to assess measures for fire prevention, fire detection, fire suppression and fire containment in oder to demonstrate that the plant will maintain the ability to perform safe shutdown functions and minimize radioactive releases to the environment in the event of a fire;
8. The term “fire protection program” means the integrated plan and actions involving components, procedures, and personnel utilized in carrying out all activities of fire protection. It includes system and facility design, fire prevention, fire detection, annunciation, confinement, suppression, administrative controls, fire brigade organization, inspection and maintenance, training, quality assurance, and testing;
9. The term “fire protection implementation program” means a document prepared by reactor operator to implement fire protection program required for operating the reactor facilities; and
10. The term “plant fire brigade” means the team of shift personnel assigned to fire-fighting at the initial stage of a fire in the reactor facilities.

Article 4 (Establishing of Fire Protection Program) A reactor operator shall establish a fire protection program of reactor facilities concerned (for example, on 2-unit basis) in accordance with the followings:

1. Reactor operator shall appoint a fire protection officer who manages the fire protection program of the reactor facilities.
2. Fire protection officer shall be responsible for establishing, implementing and managing of a fire protection program and shall be entitled to perform his task effectively.

3. Fire protection officer shall constitute a fire protection implementation system to establish, implement and manage a proper level of fire protection program for reactor safety.
4. The fire protection implementation program shall include the followings based on the defense-in-depth concept:
   a. Authority and responsibility of the organizations involved in fire protection activities;
   b. Administrative procedures for fire prevention;
   c. Administrative procedures for fire fighting;
   d. Administrative procedures for post-fire reactor safe shutdown; and
   e. Plant fire brigade training program.

Article 5 (Guidelines for Fire Protection Implementation Program) Fire protection implementation program shall include the followings:

1. Authority and responsibility of the organizations involved in fire protection activities
   a. The authority and responsibility for fire protection officer shall be established.
   b. The organizational responsibilities, authority and lines of communication pertaining to fire protection shall be defined among the various positions. The positions/organizations shall be designated for carrying out each of the followings:
      1) Preparation and update of the fire protection implementation program and management of implementing results;
      2) Administrative procedures for fire prevention, fighting and post-fire reactor safe shutdown;
      3) Organization, operation and training of plant fire brigade and management of implementing results;
      4) Emergency plans including instructions for fire brigade members during a fire and fire-fighting strategy on each fire area;
      5) Periodic review and updating of the fire hazard analysis;
      6) A records management system including means for documentation and analysis of records of fire events;
      7) The functional performance of all fire protection features such as fire detection and alarm system, fire suppression system, fire escape apparatus and fire barrier; and
      8) Quality assurance specifically addressing fire protection.

c. The plant fire brigade shall have at least five members on each shift. The authority and responsibility of the fire brigade leader and members shall be clearly defined in accordance with the fire suppression measures of each fire area or fire zone. Provided, that the personnel required for reactor safe shutdown shall not be a member of the plant fire brigade.

2. Administrative procedures for fire prevention

a. The types, quantities and positions of ignition sources and combustible materials used for operation of reactor facilities shall be identified and the status of transient combustible materials shall be inspected periodically. The administrative procedure for handling hazardous combustible materials and limiting their use shall be established.

b. An administrative procedure shall be established and implemented to control maintenance and modification activities (welding, flame cutting, brazing and soldering operations) that necessitate the use of a ignition source or that may themselves create an ignition source. A work permit system or, if necessary, a work monitor system shall be introduced in the administrative procedure.

c. A fire prevention plan shall be established for the areas where the potential fire damage may jeopardize the capabilities to shut down the reactor and maintain it in a safe shutdown condition or may increase the possibilities of radioactive releases during the maintenance period.

d. The appropriateness of the fire hazard analysis shall be reviewed every ten years. In case of any modification of design in reactor facilities or any change in operation method that may affect each of the followings, the relevant part of the fire hazard analysis shall be revised:

1) Separation of fire protection area;
2) Types and quantities of combustible materials;
3) Design basis fire;
4) Fire detection and suppression system;
5) Fire hazards; and
6) Capabilities of reactor safe shutdown, residual heat removal and prevention of radioactive releases.

3. Administrative procedures for fire-fighting

a. Emergency plans shall be established including instructions for plant fire brigade members and plant personnel (including operators) in case of fire.

b. All fire protection features such as fire detection and suppression system and protective equipment for fire brigade members shall be summarized and described for each fire area.

c. Inspection items, intervals, procedures and inspection department including technical inspection standard shall be described in the self-inspection program for fire detection and suppression system.

d. Fire-fighting strategy shall be established for the areas where the potential fire damage may jeopardize the capabilities to shut down the reactor and maintain it in a safe shutdown condition, or may increase the possibilities of radioactive releases during the maintenance period.

4. Administrative procedures for post-fire reactor safe shutdown

a. An emergency plan shall be established to ensure the capability for safe shutdown of the reactor in consideration of the effect on the capability to safely shutdown the reactor and maintain it in a safe shutdown condition and of the possibility of radioactive releases to the environment in case of a fire.

b. Fire escape apparatus and fire barrier for each fire areas shall be summarized and described.

c. Inspection items, intervals, procedures and inspection department including technical inspection standard shall be described in the self-inspection program for fire escape apparatus and fire barrier.

5. Education and training program for the plant fire brigade

a. An education and training program for the plant fire brigade shall be established.

b. The instruction of the plant fire brigade shall be conducted by qualified individuals who are knowledgeable, experienced, and suitably trained in a accredited professional agency, using the types of equipment available in the reactor facilities.

c. The instruction program shall include the followings to ensure that the capability to fight potential fires is established and maintained, and periodic refresher training sessions shall be provided to repeat the instruction program for all brigade members over a 2-year period:

1) Plant fire fighting strategy with specific identification of each individual's responsibilities;

2) Identification of the type and location of fire hazards and associated types of fires that could occur in the plant;

3) The toxic and corrosive characteristics of expected products of combustion;

4) Identification of the location of fire fighting equipment for each fire area and familiarization with the layout of the plant, including access and egress routes of each area;

5) The proper use of available fire fighting equipment and the corrective method of fighting each type of fire;
6) The proper use of communication, lighting, ventilation, and emergency breathing equipment;
7) The proper fire-fighting method inside the buildings and confined spaces;
8) The direction and coordination for fire-fighting activities (for fire brigade leaders only);
9) Details of fire-fighting strategies and procedures;
10) Review of the latest plant modifications and corresponding changes in fire fighting plans; and
11) Coordination with the other fire brigades.

d. Fire brigade drills shall be conducted on a shift basis and drills shall be performed at regular intervals not to exceed 3 months for each shift fire brigade. Each fire brigade member shall participate in each drill, but must participate in at least two drills per year.

e. A sufficient number of these drills shall be unannounced, but not less than one for each shift fire brigade per year.

f. The drills shall be preplanned to establish the training objectives of the drill and shall be critiqued by the fire protection officer or drill evaluator to determine how well the training objectives have been met. Performance deficiencies of a fire brigade or of individual fire brigade members shall be remedied by scheduling additional training for the brigade or members. Unsatisfactory drill performance shall be followed by a repeat drill within 30 days.

g. Items to be appraised in fire brigade drills shall be as follows:
1) Effectiveness of fire alarm, time to notify and to assemble fire brigade members, and selection of equipment and fire-fighting strategies; and
2) Knowledge of each brigade member with his or her role in fire fighting strategy, conformance with fire fighting procedures and use of the equipment, fire brigade leader's direction.

h. Individual records of education and training provided to each fire brigade member, including drill critiques, shall be maintained for at least 3 years.

Article 6 (Submission of Fire Protection Implementation Program) A person who intends to operate reactor facilities shall submit a fire protection implementation program to the Minister of Science and Technology 3 months prior to the initial fuel loading.

Addenda

Article 1 (Enforcement Date) This notice shall enter into force on the date of its promulgation.

Article 2 (Transitional Measures) A person who is operating reactor facilities or who has obtained the construction permit at the enforcement date of this notice shall submit a fire protection implementation program to the Minister of Science and Technology within 12 months from the date of this notice.

Article 3 (Exceptional Measures) The provisions related to the plant fire brigade shall not apply to the fire protection program required for operation of research reactor.

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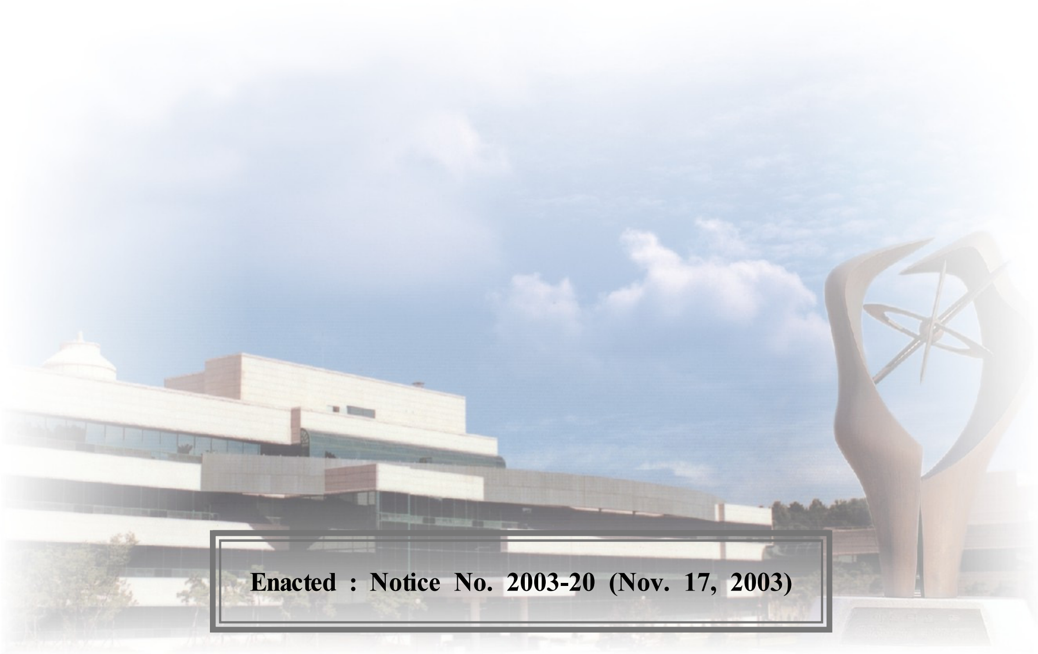
Technical Standards for Fire Hazard Analysis



Enacted : Notice No. 2003-20 (Nov. 17, 2003)

◎ **Notice of the Minister of Science and Technology No.2003-20 (MOST.reactor.032)**

The Technical Standards for Fire Hazard Analysis as provided for in Article 14 of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc. is hereby notified publicly as follows:

November 17, 2003
Minister of Science and Technology

Technical Standards for Fire Hazard Analysis

Article 1 (Purpose) The purpose of this notice is to specify the technical standards for fire hazard analysis in accordance with Subparagraph 2 of Article 12, Subparagraph 2 of Article 22 and Article 33 (3) of the Atomic Energy Act and Article 14 of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc.

Article 2 (Scope of Application) This notice shall apply to the nuclear power reactor and related facilities so that they do not present any impediment to the protection against radiation hazards to human bodies, materials and the general public caused by the radioactive materials.

Article 3 (Definitions) Definitions of the terms used in this notice shall be as follows:

1. The term “noncombustible material” means a material that, in the form in which it is used and under the conditions anticipated, will not ignite, burn, support combustion, or release flammable vapors when subjected to fire or heat;
2. The term “combustible material” means a material that does not meet the definition of noncombustible;
3. The term “fire barrier” means those components of construction (walls, floors, and their supports), including beams, joists, columns, penetration seals or closures, fire doors, and fire dampers that are rated by accredited laboratories in hours of resistance to fire and are used to prevent the spread of fire;
4. The term “fire area” means a building or part of a building comprising one or more rooms or spaces, that is separated from other areas by fire barriers, constructed to prevent the spread of fire to or from the remainder of the building for a given

period of time;

5. The term “fire zone” means a subdivision of a fire area in which fire separation between items important to safety is provided by limitation combustible materials, spatial separation, or fixed fire extinguishing systems;
6. The term “design basis fire” means those fires that may develop in local areas, assuming that no manual, automatic, or other fire-fighting action has been initiated, and that have passed flashover and have reached its peak burning rate. It is used as a measure to anticipate the maximum damage of the fire area;
7. The term “fire hazard analysis” means qualitative and/or quantitative hazard analysis to investigate potential in situ and transient fire hazards in each fire area and to assess measures for fire prevention, fire detection, fire suppression and fire containment in order to demonstrate that the plant will maintain the ability to perform safe shutdown functions and minimize radioactive releases to the environment in the event of a fire;
8. The term “fire load” means the sum of the calorific energies per unit area which could be released by the complete combustion of all the combustible materials in a spaces, including the facing of the walls, partitions, floors and ceilings; and
9. The term “fire resistance rating” means the grade of ability of an element of building construction, component or structure to fulfil, for a stated period of time, the required load bearing function, integrity and/or thermal insulation, and/or expected duties specified in a standard fire resistance test.

Article 4 (Establishment of Fire Protection Area) Fire protection area, which consists of fire area and fire zone, shall be established to secure proper isolation to mitigate the consequences of fire on structures, systems, or components important to safety. As to the fire protection area, the following items shall be included in the fire hazard analysis report.

1. Minimum technical background to prove that the postulated fire in each fire area will not compromise nuclear safety.
   a. Design basis fire and assessment of the fire hazard in each fire area.
   b. Arrangement status of structures, systems, or components important to safety, including cable and pumps.
   c. Fire resistance rating of fire barrier required by design basis fire
   d. Fire resistance rating of fire barrier, required in adjacent areas
2. Result of assessment of the suitability of the fire zones in cases where the fire barriers in a fire area do not meet the required fire resistance rating

a. Complementary effects of sufficient separation distance and fire extinguishing system

b. Influence of radiative or convective fire effects, smoke or other decomposition products of combustion, or spreading flammable/combustible liquids or gases on structures, systems, or components that are important to safety

c. Possibility of prevention of fire-fighting activities or spurious actuation of a fire extinguishing system by the smoke from the adjacent fire areas or zones

d. Adverse effects of the fire extinguishing system actuation (even in the event of spurious actuation) on the safety systems in the relevant areas (fire areas or fire zones) or adjacent areas

3. Items related to fire area

a. Physical dimension (length, width, height and characteristics), layout and structure of each fire area

b. Types, layout and redundancy of structures, systems, or components that are important to safety

c. Construction materials of walls, floors and ceilings

d. Designs and operating conditions of the heating, ventilation and air conditioning and smoke control system

e. Details of any drainage systems, including inlet and outlet, and any system for containing liquid leaks

f. Details of systems and components in the fire area containing any radioactive materials

g. Access to and egress from each fire area

4. Items related to fire barrier

a. Fire resistance rating of the fire barrier in each fire area

b. Characteristics of fire barrier and fire resistance performance

c. Fire resistance test standards applied to fire barriers

Article 5 (Types and Quantities of Combustible Materials) The fire hazard analysis shall describe measures including the followings to limit types and quantities of combustible materials and ignition sources for the purpose of preventing fire:

1. Status of combustible materials in each fire area

a. All machinery and equipment containing combustibles

b. Quantities of flammable/combustible liquids and gases, including their configurations and container types

c. Other combustible materials (e.g. wood and hydraulic fluids), together with their configurations

d. Location of cables, together with other relevant details (e.g. tray type and fill density, fire resistant standard, insulation and orientation)
e. Details of electrical and electronic equipment, including lighting, together with any fire protection standards with which they comply
f. All facings of walls, floors and ceilings

2. Status of ignition sources in each fire area
   a. All fixed equipment and processes that have, or are capable of generating, an ignition source, including open flames, sparks and hot surfaces
   b. Ignition sources resulting from equipment failures, including a sudden release of energy, frictional heat, electric arcs and spontaneous combustion
   c. Ignition sources caused by operation or construction activities, including welding, cutting, grinding and unsafe use of heat and flame sources
   d. All possible types of transient ignition sources
3. Guidance on housekeeping of combustible materials and ignition sources for each fire area, and necessity and adequacy of combustible materials in a fire area or adjacent area containing structures, systems and components important to safety
4. Prevention measures for potential fire induced by operation or malfunction of the equipment containing combustible material
5. Intermediate storage of combustible materials, and transient combustibles produced by maintenance work, etc.

Article 6 (Category of Design Basis Fire) Categories of design basis fire to be considered in fire hazard analysis shall be as follows:

1. On multiple-reactor sites, fires involving facilities shared between units and fires due to man-made site-related events (such as an aircraft crash) that have a reasonable probability of occurring and affecting more than one nuclear reactor unit shall be considered. Provided that, worst case fires need not be postulated to be simultaneous with non fire-related failures in safety systems, plant accidents, or the most severe natural phenomena and unrelated fires in two or more units need not be postulated to occur simultaneously.
2. Design basis fire in each fire area shall be established, and the potential effects of the design basis fire shall be evaluated on the structure, system, and component.
3. Characteristic and scenario of the design basis fire in each fire area shall be established and the protection plan shall be described.
4. Fire damage limits are shown in Table according to the safety function of the structure, systems, and components during design basis fire.

Article 7 (Fire Detection and Suppression System) With regard to fire detection and suppression system, the followings shall be included in the fire hazard analysis report.

1. The structure, system, and component important to safety, and the protection characteristic established by the potential fire hazard of each fire area (or fire zone), and types of fire detection and suppression system and its adequacy
2. Adequacy of the inventory and performance of the fire detection and suppression system to minimize the effect on the structure, system, and component important to safety
3. Items related to fire detection and suppression system
   a. Details of the specific design standards
   b. Types and locations of the fire detection systems and fire alarm systems, and their operation methods
   c. Fixed water extinguishing system
   d. Details of other fixed extinguishing system (foam, dry powder and gas)
   e. Types and locations of portable fire extinguishers
   f. Details of the smoke control systems
   g. Details of the explosion control system
   h. Types and locations of emergency lighting and communication apparatus

Article 8 (Fire Hazard Assessment) The followings shall be considered to analyze fire load and fire characteristic for each fire area and to evaluate the fire hazard.

1. One of the three methods shall be applied to analysis of fire load and fire characteristic: an evaluation on the basis of only practical experience and engineering judgement, manual calculation using empirical formulas, graphs or tables, and computer calculation with fire modeling. The analysis methods shall only be applied within the scope, limitations, and assumptions prescribed for that methods.
2. The followings shall be considered to analyze fire load and fire characteristic for each fire area:
   a. Combustion heat (calorific value);
   b. Ignition point and/or flash-point of combustibles;
   c. Mass burning rate of cables in a horizontal/vertical position, of oil in open area or in closed rooms, and other combustibles in open area or in closed rooms;
   d. Heat release rate;
   e. Percentage of oxygen that supports combustion;

f. Explosion limit of flammable liquids and gases;
g. Dependency of mass burning rate on oxygen content;
h. Whether flame spread is assumed to be possible if redundant cable trays have only a minimum horizontal separating distance or only a minimum vertical separating distance;
i. Whether ignition of combustibles is always assumed to be possible, or if exceptions are stated (e.g in connection with external events); and
j. Flame propagation data for vertical/horizontal cables as single cables or in cable bundles, for combustible liquid surfaces and for other conditions.

3. Fire hazard assessment shall be performed in consideration of the following items:
a. Temporary increases and concentrations of combustible materials expected to be used in normal operations such as refueling, mid-loop operation, maintenance, and modifications;
b. Continuity of combustible materials, furnishings, building materials, or combinations thereof in configurations conducive to fire spread;
c. Exposure fire, heat, smoke, or water exposure, including those that may necessitate evacuation from areas that are required to be attended for safe shutdown;
d. Fire in control rooms or other locations having critical safety-related functions;
e. Adequate access or smoke removal facilities that impede fire-fighting in safety-related areas;
f. Explosion-prevention measures;
g. Functions of electric power supply or control circuits; and
h. Inadvertent operation of fire suppression systems.

Article 9 (Capabilities of Reactor Safe Shutdown, Residual Heat Removal and Prevention of Radioactive Releases) The followings shall be considered in evaluating the capabilities of the nuclear reactor safe shutdown, residual heat removal and prevention of radioactive releases:

1. The structure, system, or component which is important to safety (including those required for reactor safe shutdown, residual heat removal) or which contains radioactive materials shall meet the fire damage limits in Table during design basis fire;
2. Effect analysis of design basis fire in a fire-affected fire area containing structures, systems, and components important to safety shall be conducted to evaluate the effect not only on the fire area, but also on the adjacent fire areas

which may affect adversely;

3. Fire suppression activities and the capabilities for reactor safe shutdown shall not be adversely affected by propagation of smoke, hot gas and fire suppressant (water, foam, CO_2 and halon, etc.) from a fire-affected area to non-fire-affected fire areas;
4. In the reactor containment, redundant shutdown systems shall be provided with fire protection to ensure, to the extent practical, that one shutdown division shall be free from fire damage; and
5. Alternative or dedicated shutdown capability shall be provided in case that the fire in a fire area has an effect on the safe shutdown capabilities.

Article 10 (Submission of Fire Hazard Analysis) The fire hazard analysis shall be submitted to the Minister of Science and Technology as a review document for construction permit and operation license.

Addendum

This notice shall enter into force on the date of its promulgation.

[Table]

Fire Damage Limits According to the Safety Function of the Structure, System, or Component Important to Safety

Functions	Fire Damage Limits
Hot Shutdown	One train of equipment necessary to achieve hot shutdown from either the control room or emergency control station(s) must be maintained free of fire damage by a single fire, including an exposure fire.
Cold Shutdown	Both trains of equipment necessary to achieve cold shutdown may be damaged by a single fire, including an exposure fire, but damage must be limited so that at least one train can be repaired or made operable within 72 hours using onsite capability. The equipment in any one fire area is assumed to be rendered inoperable by the fire and re-entry into the fire area for repairs and operator actions is not possible. The control room is excluded from this approach provided an independent alternative shutdown capability physically and electrically independent of the control room is provided.
Design Basis Accident	Both trains of equipment necessary for mitigation of accidents consequences following design basis accidents may be damaged by a single exposure fire.

【 24 】

Regulation on In-Service Test of Safety-related Pumps and Valves

Enacted : Notice No. 2004-14 (Jul. 02, 2004)

◎ **Notice of the Minister of Science and Technology No.2004-14 (MOST.reactor.033)**

The Regulation on In-Service Test of Safety-related Pumps and Valves as provided for in Subparagraph 2 of Article 63 (1) of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc. is hereby notified publicly as follows:

July 2, 2004
Minister of Science and Technology

Regulation on In-Service Test of Safety-related Pumps and Valves

Article 1 (Purpose) The purpose of this notice is to establish requirements for in-service testing to assess performance and monitor and evaluate degradation from the aging of safety related pumps and valves as set forth in Subparagraph 2 of Article 63 (1) of the Regulations on Technical Standards for Nuclear Reactor Facilities, etc.

Article 2 (Scope of Application) This notice shall be applied to pumps and valves with safety function.

Article 3 (Definitions) Definitions of the terms used in this notice shall be as follows:

1. The term "safety function" is a function to safely shut down a nuclear reactor, maintain the shutdown state and prevent accident or mitigate of its result;
2. The term "in-service testing" refers to testing to assess performance of safety-related pumps and valves of nuclear reactor facilities and to monitor and evaluate degradation from aging during nuclear reactor operation;
3. The term "pre-service testing" refers to testing to assess performance of safety-related pumps and valves of nuclear reactor facility and to perform upon installation to determine reference values used during in-service testing;
4. The term "reference values" refers to one or more testing values of parameters to be applied in determining safety function capability of pumps and valves;
5. The term "active valves" are valves that are required to change obturator position to perform the required safety function;
6. The term "passive valves" are valves that maintain obturator position and are not required to change obturator position to perform the required safety function;

7. The term "exercising test" means the demonstration test of valve moving parts that is based on direct visual or indirect positive indications;
8. The term "full stroke time" means the time interval from initiation of the actuating signal to the indication of the end of the operating stroke;
9. The term "category A valves" are valves for which seat leakage is limited to a specific maximum amount in the closed position for fulfillment of their required function;
10. The term "category B valves" are valves for which seat leakage in the closed position is inconsequential for fulfillment of their required function;
11. The term "category C valves" are valves that are self-actuating in response to some system characteristic, such as pressure(relief valves) or flow direction (check valves) for fulfillment of their required function;
12. The term "category D valves" are valves that are actuated by an energy source capable of only one operation, such as rupture disks or explosively actuated valves;
13. The term "design basis performance" means a performance at worst operating conditions including normal, abnormal operation and design criteria accident;
14. The term "operation margin" means a difference between a maximum capacity of valve actuating parts at design criteria condition and a capacity for fulfillment of their required function;
15. The term "group A pumps" are pumps that are operated continuously or routinely during normal operation, cold shutdown, or refueling operations;
16. The term "group B pumps" are pumps in stand by systems that are not operated routinely except for testing;
17. The term "comprehensive test" means a test that is performed every two years to check flow rates, pressures and vibrations of group A and B pumps;
18. The term "nonintrusive testing" means a testing performed on a component without disassembling or disturbing boundary of component; and
19. The term "condition monitoring program" means a program that optimizes preventive maintenance activities to maintain capabilities of check valves.

Article 4 (Establishment and Implementation of Test Program) (1) The installer of nuclear power reactor shall establish and perform pre-service test program during initial operation period to acquire base data for in-service test such as pump performance curve, reference values of pumps and valves needed to compare, analyze and evaluate performance change during plant operation. Provided, that for motor- and pneumatically- operated valves, design basis performance evaluation program shall be established and implemented additionally to evaluate design basis operation margin.

(2) The operator of nuclear power reactor shall establish and implement in-service test program for safety-related pumps and valves spanning every ten years after commencement of commercial operation. Provided, that for motor- and pneumatically- operated valves, additional periodical performance evaluation program shall be established and implemented to check design basis operation margin and performance maintenance.

(3) More details as regards in-service test shall follow "MOA (2000 edition) of Korea Electric Power Industry Code (KEPIC)" or "ASME OM Code, ISTA (1995 edition and addenda).

Article 5 (Submission of Pre-Service Test and In-Service Test Programs) (1) The operator of nuclear power reactor shall submit in-service test program that contains the following items to the Minister of Science and Technology:

1. General
   a. Test frequency
   b. Applicable technical codes and standards, etc.
   c. Identification of components subject to tests or examination and safety function of individual component
   d. Reference (applicable codes and standard, regulatory documents, P&ID, etc.)
2. Details
   a. Exemptions of test applied to each system
   b. List of components including system identification, code classification and nominal size
   c. Details of test (pump test schedule and valve test schedule for each system)
   d. List of test procedures including number and title of procedure
   e. Justification of cold shutdown test
   f. Justification of refueling shutdown test

(2) The installer of nuclear power reactor shall prepare pre-service test program including (1) 1. b, c, d and 2. a, b, c.

(3) Pre-service test program shall be submitted to the Minister of science and technology before start of initial operation test and in-service test program till three month prior to commencement of each in-service test.

Article 6 (Submission of Design Basis and Periodic Performance Evaluation Program of Motor- and Pneumatically-Operated Valves) (1) The installer of nuclear power reactor shall submit design basis performance evaluation program of motor- and pneumatically-operated valves including the following items to the Minister of Science and Technology.

1. General
   a. Test schedule
   b. Valves for test (includes selection principle)
   c. List of valves by systems including types and safety function, etc.
   d. Corrective action for valves not satisfying operation margin
   e. Reference (applicable standards, reference standards, regulatory documents, P&ID, etc.)
2. Details
   a. Method and content of design basis performance evaluation
   b. List of implementation procedures
   c. List of valves including valves subject to test, design basis differential pressure and nominal dimension
   d. Exemptions of test

(2) The operator of nuclear power reactor shall submit periodic performance evaluation program of motor- and pneumatically- operated valves including the following items to the Minister of Science and Technology.

1. General
   a. Frequency of performance evaluation or test schedule
   b. Valves subject to test (including selection principle)
   c. List of valves including types and safety function, etc.
2. Details
   a. Periodical performance evaluation method and content
   b. List of implementation procedures

(3) Design basis performance evaluation program and periodic performance evaluation program of motor- and pneumatically- operated valves shall be submitted to the Minister of Science and Technology till three month prior to commencement of test.

(4) The frequency of design basis performance evaluation test of motor- and pneumatically- operated valves shall be established on the basis of risk, operation margin and operating experience so as to meet the followings.

1. The first test shall be done during the third refueling outage or within five years

after design basis performance evaluation for each valve.

2. Within maximum ten years after first test for successive tests

Article 7 (Reference Values) (1) The initial reference values shall be determined from the results of pre-service test or initial in-service test.

(2) When a reference value or set of values for pumps or valves may have been affected by repair, replacement, or maintenance activities, a new reference values or set of values shall be evaluated and determined whether the values demonstrate adequate operability.

Article 8 (Data Collection and Record Keeping) (1) Records or data of pre-service test and in-service test shall be maintained as set forth in the quality assurance program during the facility life including at least followings:

1. Pre-service and in-service test program and its changes;
2. Design basis performance evaluation program, periodic performance evaluation program of motor- and pneumatically-operated valves and their changes;
3. Pre-service and in-service test results of pumps and valves and results of design basis performance and periodic performance evaluation of motor- and pneumatically-operated valves;
4. Reasons of resetting of reference values and evaluation result of operation status in Article 7 (2);
5. Causes of excessiveness of allowable range and evaluation in Article 10; and
6. History of request for exemption and records of basis in Article 11.

Article 9 (Test Parameters and Codes) (1) If there are not any special reasons, tests for Group A pumps and Group B pumps shall be run every 3 months and comprehensive test shall be run at least once every 2 years. The tests shall meet the followings:

1. Pump tests shall be run on hydraulic and mechanical performance and the tests shall measure parameters in the Table 1 under set test conditions; and
2. Details such as test method, test conditions and procedures, and deviations of measurements, etc., not specified in this notice as regards pump test, shall follow "KEPIC MOB(2000 edition)" or "ASME OM Code, ISTB (1995 edition and Addenda)".

(2) Valve tests shall meet the followings:

1. Valve test shall be run for check points in the Table 2.

2. Valves with remote position indicators shall be observed locally at least once every 2 years to verify that valve location is accurately indicated.
3. Valves with fail-safe actuators shall be tested by observing the operation of the actuator upon loss of valve actuating power in accordance with the exercising frequency.
4. If there are not any special reasons, active valve actuation test shall be conducted at full stroke required to fulfill its function every 3 month.
5. Category A valves shall be tested to verify their seat leakages within acceptable limits at least once every 2 years, and category A containment isolation valves shall be tested in accordance with "Standards for leakage rate tests of reactor containment".
6. Category C safety and relief valves shall be tested to verify set pressure and seat leakage, and their test frequency is as follows; every 5 years for safety class 1 and at least once every 10 years for safety classes 2 and 3, and once every 5 years for safety class 2 main steam safety valve.
7. Category C check valve shall be tested at least every 3 month to verify that the leakage device travel to its required position unless there is any specific reason. Alternative tests include overhaul, nonintrusive testing or condition monitoring program. In case that condition monitoring program is selected as an alternative, the applicable requirements in "Korea Electric Power Industry Codes (KEPIC) MOC(2000 edition)" or in "ASME OM Code, ISTC (1995 edition and addenda)" shall be modified as follows:
   a. In case that flow test or verifying methods such as nonintrusive testing or overhaul are used, valve open and close test shall be conducted;
   b. Initial test period shall not exceed 2 refueling periods or 3 years. In case that this period is extended, the extended period can not exceed 1 refueling outage at a time, and the total extension shall not exceed 10 years. The extension or reduction of test period have to be based on the evaluation of past test results and trend analysis; and
   c. If the condition monitoring program is discontinued, then the requirements of "MOC 4510 to 4540" or "ISTC 4.5.1 to 4.5.4" shall apply.
8. Category D valves shall be replaced at least every 5 years and the replace period can be reduced according to component history.
9. "KEPIC MON-1 (2000 edition)" or "ASME OM Code OMN-1" can apply as an alternative for pre-service and in-service tests of motor-operated valves. In case the "MON-1" or "OMN-1" applies, initial diagnose test shall be conducted

within 5 years. In case that the exercise test period is extended from 3 months, the fact that the frequency and risk of reactor core degradation do not increase due to extended test period shall be demonstrated.

10. Details of categories A, B, C check valve tests such as test method shall follow "KEPIC MOC (2000 edition)" or "ASME OM Code, ISTC(1995 edition and addenda)".
11. Details of category C (safety and relief) valve tests and category D valve tests such as test method shall follow "KEPIC MOD (2000 edition)" or "ASME OM Code, Appendix I (1995 edition and addenda)".

(3) In addition to the Paragraph 2, tests of design basis performance evaluation and periodical performance evaluation for motor- and pneumatically- operated valves shall meet the followings:

1. Design basis performance shall be evaluated considering inaccuracy of test apparatus and censors. Evaluation shall include at least the followings:
   a. Analysis of design basis for checking integrity of fragile side, minimum required capability and maximum operable capability;
   b. Diagnose test for checking the appropriateness of set value of control switch (or acceptable analytical method); and
   c. Evaluation of operation margin.
2. For valves for which the first design basis performance evaluation was completed, degradation of actuator capability that is usable and change of operation margin due to increase for required capability shall be monitored and evaluated.

   These monitoring and evaluation shall include the followings:
   a. Method of periodical performance evaluation and test frequency;
   b. Diagnose test for checking the appropriateness of set value of control switch (or acceptable analytical method); and
   c. Change of operation margin.

Article 10 (Evaluation of Test Results and Corrective Actions) (1) The test results for pumps and valves shall be compared with the reference values to confirm the results meet the allowable limits.

(2) In case that test results for pumps exceed the allowable limit of reference values, the following measures shall be taken:

1. The cause of deviation shall be determined in case that the result is within the warning range and the test frequency shall be reduced half by completion of

corrective action; and

2. The concerned pump shall be declared inoperable in case that the result is within the action range and proper corrective action shall be taken along with determination of cause of that deviation.

(3) In case that the test results for valves do not meet criteria or secure design basis margin, the following items shall be executed:

1. In case that the test results of valves such as stroke time and leakage rate exceed the criteria, the operability of concerned valves shall be evaluated and proper measures be taken; and
2. In case that the operation margin is not sufficient by the next test from the design basis performance evaluation and periodical performance evaluation, measures to maintain safety function shall be established and corrective action be taken.

Article 11 (Relief Request) (1) The operator of nuclear power reactor may request relief for the following cases:

1. In case that tests in accordance with the requirements can not be conducted due to difficulties and inappropriateness of design, configuration and material of components and systems subject to tests;
2. In case that unnecessary overexposure is anticipated due to environmental conditions in spite of protective measures;
3. In case that access to components subject to tests is not possible physically or tests in accordance with requirements can not be conducted due to other inevitable reasons; and
4. In case that tests in accordance with "KEPIC" or "ASME OM Code" requirements can not be conducted practically.

(2) In order to make a relief request as set forth in Paragraph 1, a relief request including the following items shall be filed with the Minister of Science and Technology:

1. Name of business place;
2. Name, number and quantity, and safety class of components subject to relief request(including pump group and valve category);
3. Safety function of components subject to relief request;
4. Applicable requirements;
5. Reasons and contents for relief request;
6. Documents providing evidence that the test requirements can not apply (drawings,

sketch, photos, etc.); and

7. Articipated changes to plant quality and safety when relief request is allowed, and justifications thereof.

(3) The relief request is in effect on the date when the Minister of Science and Technology approves it.

Addenda

Article 1 (Enforcement date) This notice shall enter into force on the date of its promulgation.

Article 2 (Transitional Measures) The in-service testing plan in accordance with Notice of the MOST No.98-15 and the design basis performance evaluation plan in accordance with the recommendation of the MOST (Wongom 71233-205, dated June 13, 1997) before the enforcement date of this notice, shall be deemed as having been applied according to this notice.

Article 3 (Design Basis Performance Evaluation for Pneumatically-Operated Valves) For Uljin 5, 6 and the nuclear reactor facilities in operation as of the enforcement date of this notice, design basis performance evaluation for pneumatically-operated valves shall be completed within 6 years from the enforcement date of this notice, and for new reactor facilities it shall be completed before issuance of operating license.

Article 4 (Repeal of Notice) Notice of the MOST No.98-15 "Rules on In-service Inspections and In-service Tests of Nuclear Reactor Facilities" shall be repealed at the time this notice enters into force.

[Table 1]

Pump Test Parameters

Test Items	Parameters of Measurement	Remarks
Hydraulic Capability	Speed Differential Pressure Discharge Pressure	in case of variable speed centrifugal pump (includes vertical line shaft pumps) displacement pump (excludes group B pumps)
	Flow	centrifugal pumps and displacement pumps
Mechanical Capability	Vibration	speed (peak value) or deviation (between peaks) (excludes group B pumps)

[Table 2]

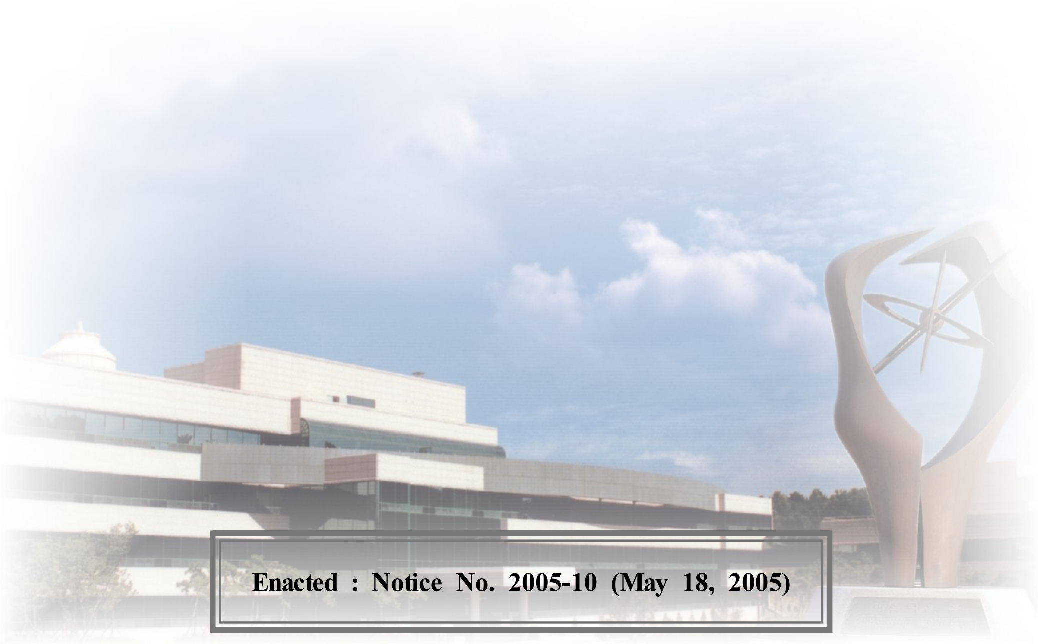
Valve Test Requirements

Category	Function	Test Items			
		Leak Test	Exercising Test	Position Indication Verification Test	Special Test
A	Active	○	○	○	
	Passive	○		○	
B	Active		○	○	
	Passive			○	
C (Safety, and Relief)	Active			○ (if applicable)	
C (Check)	Active			○ (if applicable)	
D	Active				○

Notice of the Minister of Science and Technology No.2005-10 (MOST.reactor.034)

【 25 】

Regulation on Items and Method of Periodic Inspection for Nuclear Reactor Facilities



Enacted : Notice No. 2005-10 (May 18, 2005)

◎ **Notice of the Minister of Science and Technology No.2005-10 (MOST.reactor.034)**

The Regulation on Items and Method of Periodic Inspection for Nuclear Reactor Facility as provided for in Article 19 (1) of the Enforcement Regulation of the Atomic Energy Act is hereby notified publicly as follows:

May 18, 2005

Minister of Science and Technology

Regulation on Items and Method of Periodic Inspection for Nuclear Reactor Facilities

Article 1 (Purpose) The purpose of this notice is to prescribe the items and method subject to periodic inspection of nuclear reactor facilities for power generation and for research, etc. (hereinafter referred to as "reactor facilities") as provided for in Articles 19 (1) and 28 of the Enforcement Regulation of the Atomic Energy Act.

Article 2 (Facilities and Field subject to Periodic Inspection) (1) Facilities and field subject to periodic inspection for operators of reactor facilities shall be the 12 facilities as provided for in Subparagraphs 1 through 12 of Article 19 (1) of the Enforcement Regulation of the Atomic Energy Act and the field of technical capabilities for operation as provided for in Article 50 (2) of Regulations on Technical Standards for Nuclear Reactor Facilities, etc.

(2) The periodic inspection items are established for each reactor type such as PWR, PHWR and nuclear research reactor as shown in Tables 1, 2 and 3, respectively. When deemed necessary for evaluation and verification of safety and performance of reactor facilities, some inspection items may be added.

Article 3 (Selection of Items and Method) (1) The inspection items shall be selected among those subject to inspection considering the impacts on the safety and performance.

(2) The inspection method for reactor facilities shall be as follows:

1. The inspection method for the selected items shall be documents review,

walk-down, direct observation inspection or interview with utility personnel;

2. For the preventive maintenance, performance (function) test, design change, etc. of the important safety related facilities or components, walk-down or direct observation inspection may be performed when necessary, considering the work process at site; and
3. The documents such as design change documents, related procedures or work plans may be required to be submitted for review before performing site inspection.

Article 4 (Inspection Items for Each Plant, etc.) Among the inspection items in Article 2, the inspection for the radiological emergency response facilities and the technical capabilities for operation may be conducted on a plant basis and the inspection for environmental radiation/radioactivity management and the inspection for meteorological measurement facilities may be conducted on a site basis.

Addendum

This notice shall enter into force on the date of its promulgation.

[Table 1]

Items subject to Periodic Inspection for PWR (related to Article 2 (2))

Facilities	Items Subject to Periodic Inspection
1. Reactor Pressure Vessel	Fuel Assembly Integrity, Zero power Test and Physics Test, Reactor ISI
2. Reactor Coolant System	Components and Pipes ISI within Pressure Boundary, Steam Generator Tube Inspection, Pressurizer Valve, Reactor Coolant Pump, Pressure Boundary Leakage test, Reactor Head and Penetration, RCS Leak Rate Test, Reactor Coolant Flow Rate Measurement, Water Chemistry
3. Instrument & Control System	Plant Integrity Monitoring system, Digital Rod Position Indication System, Reactor Protection System, ESF System, Control Rod Drop Time Measurement, Major I&C System Calibration, Seismic Monitoring System
4. Fuel Handling & Storage System	Fuel Transfer System, Spent Fuel Pool Cooling & Cleanup System
5. Radioactive Waste Disposal System	Radioactive Waste Management, Ventilation System Performance Test, Radiochemistry Management
6. Radiation Control System	Radiation Protection and Health Physics Planning, Radiation Detection & Monitoring System, Environmental Radiation/Radioactivity Management, Meteorological Observation and Facility Management
7. Reactor Containment System	Containment Leakage Rate Test, Containment Isolation System, Containment Spray System, Containment Combustible Gas Control System, ISI of Containment Post-Tensioning System, Metal Containment and Liner Plate.
8. Engineered Safety Feature	Shutdown Cooling System, Safety Injection System
9. Electric Power System	Emergency Diesel Generator System, Main Generator System, Transformer System, Switch Yard System, D.C Power System, Uninterrupted Power System,
10. Power Conversion System	Aux-feedwater System, Main Feedwater System, Condensate System, Steam Generator Blowdown System, Turbine & Turbine Auxiliary System, Turbine Bypass System, Auxiliary Steam System, Turbine Control & Protection System, Auxiliary Generator System, Main Steam Safety and Relief Valve, Main Steam & Main Feedwater Isolation Valve, Major I&C System Calibration, Water Chemistry.
11. Radiation Emergency Response System	Radiation Emergency Response System Test
12. Other Systems related to Safety	Structure Integrity, Essential Service Water System, Component Cooling Water System, Essential Chilled Water System, HVAC, CVCS, Compressed Air System, Sampling System, Fire Protection System, Safety & Relief Valve, Major I&C System Calibration, Supporter & Snubber, Safety Related Protective Coating, IST of Safety Related Pump & Valve, Inspection of Component & Pipe including ISI of Safety Class 2 & 3.
13. Technical Capabilities for Operation	Plant Operating Organization, Personnel Qualification & Training, Operating Procedure, Human Factor Management, Operating Experience Feedback, Test-Monitoring-Inspection and Maintenance Activities.

[Table 2]

Items subject to Periodic Inspection for PHWR
(related to Article 2 (2))

Facilities	Items Subject to Periodic Inspection
1. Calandria	Reactor Criticality, Fuel Power Distribution, Fuel Pressure Tube, Fuel Channel Flow.
2. Primary Heat Transport System	Components and Pipes ISI within Pressure Boundary, Steam Generator Tube Inspection, Primary Heat Transport Pump, Shutdown Cooling System, Heavy Water Feed & Recovery System, Feeder Grayloc, HT LRV, Water Chemistry
3. Instrument & Control System	Plant Integrity Monitoring System, All Rod Drop Test, Performance Test for SDS #1 & #2, Major I&C System Calibration, Seismic Monitoring System, PHT Capacity Test, Emergency Core Cooling System Capacity Test.
4. Fuel Handling & Storage System	Fueling Machine Interlock, Spent Fuel Pool Cooling & Cleanup System, Spent Fuel Discharge and Emergency Cooling System
5. Radioactive Waste Disposal System	Radioactive Waste Management, Ventilation System Performance Test, Radiochemistry Management
6. Radiation Control System	Radiation Protection and Health Physics Planning, Radiation Detection & Monitoring System, Environmental Radiation/Radioactivity Management, Meteorological Observation and Facility Management.
7. Reactor Containment System	Containment Leakage Rate Test, Containment Isolation System, Dousing System ISI of Containment Post-Tensioning System, Containment Non-metallic Liner.
8. Engineered Safety Feature	Emergency Water System, Emergency Core Cooling System, Poisonous Substance in SDS #2.
9. Electric Power System	Emergency Diesel Generator System, Main Generator System, Transformer System, Switch Yard System, D.C Power System, Uninterrupted Power System,
10. Power Conversion System	Aux-feedwater System, Main Feedwater System, Condensate System, Steam Generator Blowdown System, Turbine & Turbine Auxiliary System, Turbine Bypass System, Auxiliary Steam System, Turbine Control & Protection System, Auxiliary Generator System, Main Steam Safety and Relief Valve, Main Steam & Main Feedwater Isolation Valve, Major I&C System Calibration, Water Chemistry.
11. Radiation Emergency Response System	Radiation Emergency Response System Test
12. Other Systems related to Safety	Structure Integrity, Essential Service Water System, Component Cooling Water System, Essential Chilled Water System, HVAC, CVCS, Compressed Air System, Sampling System, Fire Protection System, Safety & Relief Valve, Major I&C System Calibration, Supporter & Snubber, Safety Related Protective Coating, IST of Safety Related Pump & Valve, Inspection of Component & Pipe including ISI of Safety Class 2 & 3. Moderator System, Containment Differential Settlement, Annulus Gas System
13. Technical Capabilities for Operation	Plant Operating Organization, Personnel Qualification & Training, Operating Procedure, Human Factor Management, Operating Experience Feedback, Test-Monitoring-Inspecion and Maintenance Activities

[Table 3]

Items subject to Periodic Inspection for Nuclear Research Reactor, etc. (related to Article 2 (2))

A. HANARO Facilities

Facilities	Items Subject to Periodic Inspection
1. Reactor (including Fuel)	Control Rod & Shutdown Rod Measurement
2. Reactor Coolant System	Safety Related Components ISI, RCP & Heat Exchanger Leak Test, Water Chemistry
3. Instrument & Control System	Reactor Trip Parameter Channel Operating Test, Reactor Protection System Response Time Measurement Test, Shutdown Rod Drop Time Measurement Test, Safety Related I&C Calibration, Seismic Monitoring System
4. Fuel Handling & Storage System	Fuel Refueling Crane Test
5. Radioactive Waste Disposal System	Radioactive Waste Management System, Ventilation Filter Performance Test, Radiochemistry Management
6. Radiation Control System	Radiation Protection and Health Physics Planning, Radiation Detection & Monitoring System, Environmental Radiation/Radioactivity Management, Meteorological Observation and Facility Management
7. Reactor Containment System	Reactor Isolation Damper Operating Time, Containment Leakage Rate Test
8. Engineered Safety Feature	Emergency Recovery Water Pump & Valve Test
9. Electric Power System	Safety Related Battery & Charger Test
10. Radiation Emergency Response System	Radiation Emergency Response System Test
11. Other Systems related to Safety	IST of Safety Related Pump and Valve, ISI, Fire Protection System, HVAC Emergency Ventilation Test
12. Technical Capabilities for Operation	Operating Organization, Personnel Qualification & Training, Operating Procedure, Human Factor Management, Operating Experience Feedback, Test-Monitoring-Inspection and Maintenance Activities

B. Irradiated Material Examination Facility (IMEF)

Facilities	Items Subject to Periodic Inspection
1. Radioactive Waste Disposal System	Radioactive Waste Management System, Ventilation Filter Performance Test, Radiochemistry Management
2. Radiation Control System	Radiation Protection and Health Physics Planning, Radiation Detection & Monitoring System
3. HVAC	HVAC Test
4. Fire Protection System	Fire Protection System Test
5. Transport Facility	Inspection of Irradiated Material Transport Container Handling Equipments

C. Radioisotopes Production Facility

Facilities	Items Subject to Periodic Inspection
1. Radioactive Waste Disposal System	Radioactive Waste Management System, Co-60 Storage Water Tank & Purification System, Ventilation Filter Performance Test, Radiochemistry Management
2. Radiation Control System	Radiation Protection and Health Physics Planning, Radiation Detection & Monitoring System
3. HVAC	HVAC Test
4. Fire Protection System	Fire Protection System Test
5. Transport Facility	Crane Test

【 26 】

Guidelines on Application of Technical Standards for Assessment of Continued Operation of Nuclear Reactor Facilities beyond Design Life

Enacted : Notice No. 2005-31 (Dec. 1, 2005)

◎ **Notice of the Minister of Science and Technology No.2005-31 (MOST.reactor.035)**

The Guidelines on Application of Technical Standards for Assessment of Continued Operation of Nuclear Reactor Facilities beyond Design Life as provided for in Article 19-3 (4) of the Enforcement Regulations of the Atomic Energy Act are hereby notified publicly as follows:

December 1, 2005
Minister of Science and Technology

Guidelines on Application of Technical Standards for Assessment of Continued Operation of Nuclear Reactor Facilities beyond Design Life

Article 1 (Purpose) The purpose of this notice is to prescribe the guidelines necessary for application of technical standards used for assessment of continued operation of nuclear reactor facilities as provided for in Article 19-3 (4) of the Enforcement Regulations of the Atomic Energy Act.

Article 2 (Scope of Application) These guidelines shall apply to the assessment of continued operation of pressurized water reactor beyond design life.

Article 3 (Application of Technical Standards for Assessment of Continued Operation) The matters necessary for application of technical standards for assessment of continued operation shall be as follows:

1. Matters reflecting the recent operating experience and research results
   a. assessment of scoping and screening results of aging management
   b. assessment of aging management program(AMP)
   c. assessment of time-limited aging analysis(TLAA) for continued operation
   d. matters necessary for reflecting operating experience and research results
2. Radiological environmental impact assessment in accordance with the latest technical standards

Article 4 (Matters concerning Assessment of Scoping and Screening Methodology for Aging Management) (1) The matters concerning assessment of scoping and screening results for aging management pursuant to Subparagraph 1.a of Article 3 and reference technical standards shall be presented in Table 1.

(2) Any person who wishes to apply for continued operation shall meet the following items in connection with scoping and screening results of aging management.

1. The method for determining the systems, structures and components(SSCs) within the scope of aging management shall be presented and its validity shall be proven.
2. The lists of SSCs subject to aging management and aging management program for each item shall be presented.

Article 5 (Matters concerning Assessment of Aging Management Program) (1) The matters for assessment of AMP pursuant to Subparagraph 1.b of Article 3 and reference technical standards are presented in Table 2.

(2) Any person who wishes to apply for continued operation shall meet the following items in connection with AMP.

1. The aging management program shall contain the scope of program, preventive actions, parameters monitored/inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls and operating experience.
2. The AMP shall contain matters on items to be reviewed for continued operation, measures to be taken prior to continued operation, additional measures to be taken following continued operation in consideration of characteristics of design and operation, and operating experience of relevant reactor.
3. The AMP shall be same as presented in the Table 2, but in a case where the AMP is necessary for reflecting the characteristics of design and operation, and operating experience of relevant nuclear reactor, those shall be added. Provided, that in a case where the aging management program presented in Table 2 is verified not to be applicable due to the characteristics of design and operation, and operating experience of relevant nuclear reactor, those may be excluded from assessment. And in a case where management program presented in Table 2 is established by means of combining similar items, the program shall be managed after verifying the adequacy.

Article 6 (Matters concerning TLAA for Continued Operation) (1) The matters for time-limited aging analysis falling under Subparagraph 1.c of Article 3 and reference technical standards are presented in Table 3.

(2) The time-limited aging analysis shall meet the followings:

1. An applicant shall present items subject to time-limited aging analysis and demonstrate one of the followings:
   a. The analyses remain valid for the period of continued operation;
   b. The analyses have been projected to the end of the extended period of operation; or
   c. The effects of aging on the intended functions will be adequately managed for the period of continued operation.
2. As for the individual nuclear reactor, all the exemptions actually based on time-limited aging analysis shall be presented. In this case, the technical validation for the above exemptions during the period of continued operation shall be verified.

(3) The reactor vessel neutron embrittlement shall meet the standards as set in the followings:

1. It shall be verified that there is safety margin for normal operation during the period of continued operation in response to change of Charpy upper-shelf energy caused by neutron irradiation in the beltline materials; and
2. The reactor pressure vessel shall be maintained within pressure-temperature limits established for any normal operating condition including heatup and cooldown.

(4) The assessment of environmental qualification of equipment shall meet the followings:

1. It shall be proved that the list of equipment to be qualified, qualification program and procedures are presented and the equipments are qualified during the period of continued operation;
2. The reports including criteria, methods and procedures of qualification, qualification results and qualification summary data sheets used during equipment qualification shall be prepared appropriately and be maintained in an auditable form during the period of continued operation at the plant central location; and
3. The assessment of equipment qualification shall be performed including qualification methods and implementation status, impact analysis of equipment failures, proper corrective actions to maintain equipment qualification, and protection measures of qualified equipment from adverse environmental conditions, etc.

(5) The time-limited aging analysis for continued operation shall consider the characteristics of design and operation, and operating experience of relevant reactor.

Article 7 (Matters Necessary for Reflecting Operating Experience and Research Results) (1) The matters necessary for the operating experience and research results pursuant to Subparagraph 1.d of Article 3 and reference technical standards shall be presented in Table 4.

(2) The assessment of fire protection system shall meet the followings.

1. The fire protection systems of nuclear power plants for continued operation shall be designed, installed, maintained/repaired, tested and inspected in accordance with the technical standards applied at the time of reviewing the safety report for operating license.
2. The fire protection program shall be established including each item of the followings:
   a. The fire test report and data shall be evaluated to ensure that the information is applicable and representative of the condition for which the information is being applied. As variations from the manufacture of the test specimen or test condition may substantially change the performance characteristics of test assembly the test condition shall be eralnated appropriately;
   b. Plant personnel shall be adequately trained in administrative procedure related with emergency procedure implemented in accordance with fire protection program;
   c. Modifications of the fire protection program shall be reviewed by personnel in the fire protection organization to ensure that fixed fire loadings are not increased beyond those accounted for in the fire hazards analysis, or if increased, the suitable protection shall be reflected in the fire hazards analysis; and
   d. Administrative controls and procedures shall be established to minimize fire hazards in areas containing structures, systems, and components important to safety.
3. The system for fire detection and suppression shall be provided.
4. The building design and fire barrier shall be reviewed in fire hazard analysis, and habitability of control room, post-fire safe shutdown areas, personnel access and egress pathway shall be considered.
5. For reactor facilities in which the corrective or supplementary measures related to fire hazard analysis are implemented, the measures shall be completed prior to continued operation.

(3) The assessment for seismic and dynamic qualification shall meet the followings:

1. The list of equipment and procedures for seismic qualification shall be presented;
2. The reports including the criteria for qualification, methods and procedures of qualification, qualification results, and qualification summary data sheets used for performing the equipment qualification shall be prepared appropriately and auditable records must be available and maintained by the applicant during period of continued operation at a central location;
3. The procedures for maintaining the quality including quality requirements, quality program and documents on qualification measures shall be adequately established; and
4. The assessment of seismic and dynamic qualification shall be performed including the method and implementation status of equipment qualification, impact analysis of equipment failure, appropriate corrective or supplementary actions, and protection measures of qualified equipment from harmful environmental condition, etc.

(4) It shall be proved that pressurized thermal shock related to the characteristics of neutron irradiation embrittlement of reactor vessel beltline materials has safety margin during the period of continued operation.

(5) The evaluation for anticipated transients without scram shall meet the followings:

1. The diverse protection systems with the functions such as reactor shutdown, emergency auxiliary feedwater initiation and turbine trip shall be installed in reactor facilities to cope with of possibility of an occurrence of transient not to trip even in such conditions that the reactor has to be shutdown; and
2. The diverse protection systems shall be installed independent of the reactor protection system from sensor output detecting initiation condition to the final actuation device. Provided, that this shall not apply to the Westinghouse-type reactor.

(6) The management program for the active components shall satisfy the matters falling under each of the followings:

1. The effective record of the active components subject to evaluation shall present the status of components;
2. The evaluation scope shall be determined from safety related components and non-safety related components whose failure could prevent safety related components from fulfilling their safety related function, and take into account probabilistic safety analysis result, operating experience and research results;
3. The function and performance of components shall be maintained;
4. For the components requiring corrective actions based on the prediction of component performance, the appropriate corrective action program shall be established; and

5. The maintenance management program shall be established.

(7) The assessment of thermal stratification for pipe shall meet the followings:

1. In any unisolable section connected to reactor coolant system, it shall be examined whether thermal stratification or change of temperature occurs due to the leakage of valve, and the stress caused by it is reflected on piping design;
2. Stress relief and structural damage for pressurizer surge line, pipe supports, pipe whip restraints, and anchor bolts etc. shall be examined by conducting visual inspection; and
3. The visual inspection, stress and fatigue analysis as prescribed in Subparagraphs 1 and 2 and corrective or supplementary measures shall be conducted in accordance with the relevant laws and quality assurance procedure for each plant and the records thereof shall be kept during the period of continued operation.

(8) The assessment for ignition of combustible gas shall meet the followings:

1. The accident sequences for analysis from severe accident sequences shall be selected by considering the results of engineering judgement and probabilistic safety analysis. And the quantity of producing combustible gas shall be decided by means of realistic analysis method for these accident sequences. At this time the operators' realistic management measures for severe accident may be taken into consideration;
2. The concentration of combustible gas in each compartment of the containment building during or after the accidents releasing combustible gas shall be as low as the extensive flame acceleration or de-flagration detonation transition does not occur; and
3. If Subparagraph 2 fails to satisfy, it shall be proposed reasonably and practicably that the components essential to secure integrity of the containment may be able to perform safety function while they are in the environmental conditions (including the conditions of rapid local combustion occurrence and the environmental conditions of operating combustible gas control system) of release of combustible gas.

(9) The evaluation of capability for coping with station blackout shall meet each of the followings:

1. The nuclear power plant shall be able to withstand for a specified duration of station blackout and recover from it which means the complete loss of offsite electric power supplied through the switchyard and loss of onsite AC power(turbine trip and unavailability of emergency diesel generator) to the essential and nonessential busses;

2. The reactor core and associated coolant, control and protection systems, including station batteries and an other necessary support systems shall provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration;
3. If it is proven by the analysis that a plant has capability to be adequately operated by alternate AC source and initiation of the components necessary for safe shutdown at the time of station blackout, the alternate AC source may be allowed to be a facility for coping with station blackout;
4. The capacity, capability, reliability, independence, testability and diversity of the alternate AC source used for coping with station blackout shall meet the relevant technical standards and guides; and
5. The procedures, education and training programs for coping with and recovering from station blackout shall be established and implemented.

(10) When the recent operating experience, research results and regulatory requirements of international level are reflected, the characteristics of design and operation, and operating experience of relevant reactor shall be considered.

Article 8 (Radiological Environmental Impact Assessment according to Latest Technical Standards) Radiological environmental impact assessment as set forth in Subparagraph 2 of Article 3 shall follow technical standards applied at the time of radiological environmental impact assessment which was conducted most recently in the same site, and shall contain the followings. Provided, that in the case of the nuclear power plant where radiological environmental impact assessment was performed at a previous operating licence stage, the evaluation shall be limited only to the parts different from previous evaluation for operating license.

1. Matters concerning continued operation plan;
2. Matters concerning environmental status;
3. Matters concerning status of nuclear power plant;
4. Matters concerning environmental effects of continued operation;
5. Matters concerning effects of accidents; and
6. Matters concerning environmental monitoring programs.

Article 9 (On-Site Inspection) The on-site inspection may be conducted in order to confirm the measures taken in accordance with the applied technical standards for evaluation of continued operation and the adequacy of the measures taken.

Addendum

Article 1 (Enforcement Date) This notice shall enter into force on the date of its promulgation.

[Table 1]

Matters concerning Assessment of Scoping and Screening Methodology for Aging Management

Detailed Matters	Reference Technical Standards
1. Scoping and Screening Results for Mechanical System 2. Scoping and Screening Results for Structures 3. Scoping and Screening Results for Electrical and Instrumentation and Control System	- NUREG-1800 : Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants - NUREG-1801 : Generic Aging Lessons Learned Report

[Table 2]

Matters concerning the Assessment of Aging Management Program

Detailed Matters	Reference Technical Standards
1. Safety Class 1,2,3 Component Inservice Inspection 2. Safety Class Support Inservice Inspection 3. One-Time Inspection 4. Reactor Vessel Surveillance 5. Loose Part Monitoring 6. Neutron Noise Monitoring 7. Water Chemistry 8. Boric Acid Corrosion 9. Thermal Aging Embrittlement of Cast Austenitic Stainless Steel 10. Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) 11. Flow-Accelerated Corrosion 12. Selective Leaching of Materials 13. Reactor Head Closure Studs 14. Nickel-Alloy Nozzles and Penetrations 15. Reactor Vessel Internals 16. Bolting Integrity 17. Steam Generator Tube Integrity 18. Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems 19. Buried Piping and Tanks Surveillance 20. Aboveground Steel Tanks 21. Fuel Oil Chemistry 22. Open-Cycle Cooling Water System 23. Closed-Cycle Cooling Water System 24. Compressed Air Monitoring	- NUREG-1800 : Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants - NUREG-1801 : Generic Aging Lessons Learned Report

[Table 2]

Matters concerning Assessment of Aging Management Program (continued)

Detailed Matters	Reference Technical Standards
25. Fire Protection 26. Fire Water System 27. Containment Liner Plate, Metal Containment 28. Containment 29. Integrated Leakage Rate Testing 30. Masonry Wall Program 31. Structures Monitoring Program 32. Inspection of Water-Control Structures Associated with Nuclear Power Plants 33. Protective Coating Monitoring and Maintenance Program 34. Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements 35. Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits 36. Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements 37. Bus Ducts 38. Fuse Holders 39. Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	- NUREG-1800 : Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants - NUREG-1801 : Generic Aging Lessons Learned Report

[Table 3]

Matters for Time-Limited Aging Analysis

Detailed Matters	Reference Regulations and Technical Standards
1. Identification of Time-Limited Aging Analysis	-10CFR 54.21
2. Reactor Vessel Neutron Embrittlement Analysis	-10CFR 50. Appendix G
3. Metal Fatigue Analysis	-10CFR 54.21
4. Environmental Qualification of Electric Equipment	-10CFR 50.49
5. Concrete Containment Tendon Prestress Analysis	-10CFR 54.21
6. Containment Liner Plate, Metal Containment, and Penetrations Fatigue Analysis	-10CFR 54.21
7. Other Plant-Specific Time-Limited Aging Analysis	-10CFR 54.21

[Table 4]

Matters Necessary for Reflecting Operating Experience and Research Results

Detailed Matters	Reference Regulations and Technical Standards
1. Evaluation for Fire Protection	- 10CFR 50.48
2. Seismic Qualification of Equipments	- Regulatory Guide 1.100
3. Pressurized Thermal Shock of Reactor Vessel	- 10CFR 50.61
4. Anticipated Transients Without Scram	- 10CFR 50.62
5. Management Program for Active Components	- ASME OM
6. Evaluation for Thermal Stratification of Piping	- US NRC Bulletin 88-11
7. Safety Assessment for Ignition of Combustible Gas	- IAEA Safety Guide, NS-G-1.10
8. Evaluation of Capability for Coping with Station Blackout	- 10CFR 50.63

【 27 】

Objects of Consultations Concermg Installation of Industrial Facilities, etc. around Nuclear Facilities

Enacted : Notice No. 2006-05 (Mar. 29, 2006)

◎ **Notice of the Minister of Science and Technology No.2006-05 (MOST.reactor.036)**

The facilities, etc. subject to consultations concerned as a possible cause of a serious trouble to the safety of a nuclear reactor and related facilities, nuclear fuel cycle facilities or waste disposal facilities by means of explosion, vibration and the release of toxic chemicals as provided for in Subparagraph 4 of Article 297-8 (2) of the Enforcement Decree of the Atomic Energy Act are hereby notified publicly as follows:

March 29, 2006
Minister of Science and Technology

Objects of Consultations due to Installation of Industrial Facilities around the Nuclear Facilities, etc.

Article 1 (Purpose) The purpose of this notice is to prescribe the objects of consultation for the facilities, etc. which are deemed to be feared to cause a serious trouble to the safety of a nuclear reactor and related facilities, nuclear fuel cycle facilities or disposal facilities(hereinafter referred to as "nuclear facilities") by means of explosion, vibration and the release of toxic chemicals as provided for in Subparagraph 4 of Article 297-8 (2) of the Enforcement Decree of the Atomic Energy Act, and for which the heads of administrative agencies should, at the time of permission, authorization and approval, consult with the Minister of Science and Technology.

Article 2 (Scope of Facilities subject to Consultations) The scope of installation of facilities for which the heads of administrative agencies should, at the time of permission, authorization and approval, consult with the Minister of Science and Technology shall be of up to 8km radius from the center of nuclear facilities.

Article 3 (Explosive and/or Vibratile Facilities subject to Consultations) (1) The facilities, etc. which are feared to cause a serious trouble to the safety of nuclear facilities due to explosion and vibration shall be as follows:

1. The facilities for manufacture, sale, storage and transportation of the explosives to be permitted in accordance with the provisions of Articles 4, 6, 18, 25 and 26 of the Act of Control of Firearms, Swords and Explosives, etc;

2. The facilities for filling-up, collective supply and sale of liquefied petroleum gas to be permitted in accordance with the provisions of Article 3 (1) and (2) of the Safety Control and Business of Liquefied Petroleum Gas Act, and the liquefied petroleum gas storage to be permitted in accordance with the provisions of Article 5 (1) of the Act;
3. The facilities for manufacture, supply and storage of gas to be permitted or approved in accordance with the provisions of Articles 3 and 11 (1) of the Urban Gas Business Act;
4. Manufactory, selling agency or repository of high pressure gas to be permitted in accordance with the provisions of Article 4 (1) and (3) of the High-pressure Gas Safety Control Act, and Subparagraphs 1 and 3 of Article 3 (1) of the Enforcement Decree of the Act;
5. The basic programme of railroad construction in accordance with the provisions of Article 7 of the Railroad Construction Act, and the railroad construction project to be approved in accordance with the provisions of Article 9 of the Act;
6. The route of the road in accordance with the provisions of Articles 2 and 23-2 of the Road Act; and
7. Creation of extraction right to be permitted in accordance with the provisions of Article 15 (1) of the Submarine Mineral Resources Development Act.

(2) Among the facilities as provided for in each item of the forgoing Paragraph 1, the facilities subject to consultations shall be those treating not less than the quantity depending on the distance given under the Table 1 in the course of manufacture, sale, storage and transportation.

(3) Among the facilities as provided for in each item of the forgoing Paragraph 1, the facilities subject to consultations shall be those being attended with blast exceeding the quantity of blasting powder depending on the distance given under the Table 2 in the course of construction.

Article 4 (Facilities Releasing Toxic Chemicals subject to Consultations) (1) The facilities which are feared to cause a serious trouble to safety of nuclear facilities due to releasing toxic chemicals shall be the followings treating or storing the materials diffusible to the air.

1. Toxic materials or restricted materials to treat which are subject to registration or permission in accordance with the provisions of Article 20 (1) and Article 34 (1) of the Hazardous Chemicals Control Act;
2. Among the materials to be permitted in accordance with the provisions of Article

38 of the Industrial Safety and Health Act, the materials specified by Article 30 of the Enforcement Decree of the Act.

(2) Among the facilities as provided for in Paragraph 1, the facilities subject to consultations shall be those storing and treating quantity not less than the quantity depending on distance and toxic limit given under Table 3.

Article 5 (Other Objects under Consultations) Other objects under consultations which are feared to cause a serious trouble to the safety of nuclear facilities due to fire and developments of hot spring or underground water as well as other activities shall be as follows:

1. The reserved oil facilities according to reserved oil programme in accordance with the provisions of Article 15 of the Petroleum and Substituted Fuel with Petroleum Business Act;
2. The final disposal facilities of wastes to be permitted in accordance with the provisions of Articles 26 and 30 (2) of the Wastes Control Act;
3. Designations of protective area of hot spring source in accordance with the provisions of Article 3 of the Hot Spring Act, protective area of hot spring hollow in accordance with the provisions of Article 4 of the Act, and development programme of hot spring to be approved in accordance with the provisions of Article 7 of the Act;
4. The development/utilization facilities for groundwater to be permitted in accordance with the provisions of Article 7 of the Groundwater Act;
5. The fundamental program of harbor in accordance with the provisions of Article 5 of the Harbor Act, and the implementation program of harbor construction to be notified publicly or approved in accordance with the provisions of Article 10 of the Act; and
6. The oil pipeline construction programme to be authorized in accordance with the provisions of Article 3 of the Safety Control of Oil Pipeline Act.

Article 6 (Documents Necessary for Consultations) (1) The heads of the relevant administrative agencies shall submit to the Minister of Science and Technology the relevant data of the followings in order to conduct consultations for permission, authorization or approval of the facilities subject to consultations:

1. Location and scope of the facilities and business places, the distance, direction and characteristics of medium area from nuclear reactor facilities (including drawings and photos);

2. Maximum quantity or reported quantity of use, management, storage, sale, manufacture and transportation which are to be permitted, approved or authorized; and
3. The evaluation report of event affecting nuclear reactor facilities (in case of facilities or business places where the release of toxic chemicals, explosion and fire are anticipated to arise).

(2) The evaluation report of event as provided for in Subparagraph 3 of Paragraph 1 may be evaluated by application of the "Guidelines for investigating and evaluating man-made events for site selection" attached to Table 1 of Article 2 of the "Technical Standards on the Locations, Structures and Installations of Nuclear Reactor Facilities", which was notified by the Minister of Science and Technology.

Addenda

Article 1 (Enforcement Date) This notice shall enter into force on the date of its promulgation.

Article 2 (Transitional Measures) The facilities subject to consultations, which have been permitted, authorized or approved by the heads of the relevant administration agencies prior to enforcement of this notice shall be regarded as having been consulted in accordance with this notice.

[Table 1]

Quantity of Storage or Transportation of Explosives by Distance Required for Consultations

Classification / Distance(m)[1]	Explosives(kg)	Gas(kg)	Remarks
less than 500	5,000	10,000	
500~700	5,400	10,900	
700~1,000	15,000	30,000	
1,000~1,200	60,000	120,000	
1,200~1,500	111,000	220,000	
1,500~1,700	230,000	460,000	
1,700~2,000	350,000	700,000	
2,000~4,000	500,000	1,000,000	
4,000~8,000	5,000,000	10,000,000	

Note 1) The distance refers to that from the center of nuclear reactor in case of nuclear reactor and its related facilities; that from the center of the building in case of nuclear fuel cycle facilities; that from the center of the intermediate storage facilities, taking-over facilities, and permanent disposal facilities for wastes in case of wastes facilities.

[Table 2]

First Blasting Quantity Required for Consultations in case of Blasting Construction

Classification / Distance(m)[1]	Quantity of charged Gundpowder(kg)	Remarks
less than 500	25	
500~700	30	
700~1,000	50	
1,000~1,200	120	
1,200~1,500	180	
1,500~1,700	300	
1,700~2,000	400	
2,000~4,000	550	
4,000~8,000	1,000	

Note 1) The distance refers to that from the center of nuclear reactor in case of nuclear reactor and its related facilities; that from the center of the building in case of nuclear fuel cycle facilities; that from the center of the intermediate storage facilities, taking-over facilities, and permanent disposal facilities for wastes in case of wastes facilities.

[Table 3]

Minimum Quantity of Storage or Management by Distance owing to Toxic Limit (IDLH Limit) of Hazardous Chemicals Which Require Consultations

IDLH Limit1) (mg/m^3) / Distance(m)2)	1~9	10~49	50~99	100~499	500~999	more than 1000
less than 800	1kg	10kg	45kg	90kg	450kg	900kg
800~1,100	2kg	20kg	100kg	200kg	1,000kg	2,000kg
1,100~1,600	8kg	80kg	400kg	800kg	4,000kg	8,000kg
1,600~3,200	22kg	220kg	1,100kg	2,200kg	11,000kg	22,00kg
3,200~8,000	116kg	1,160kg	5,800kg	11,600kg	58,000kg	116,000kg

Note 1) The limit that is likely to cause death or immediate or delayed permanent adverse health effects if no protection is afforded within 30 minutes (published by NIOSH).

2) The distance refers to that from the center of nuclear reactor in case of nuclear reactor and its related facilities; that from the center of the building in case of nuclear fuel cycle facilities; that from the center of the intermediate storage facilities, taking-over facilities, and permanent disposal facilities for wastes in case of wastes facilities.