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TOS-CR-4S-2011-0002
Project Number 0760

September 26, 2011

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Subject: Submittal of Technical Report "Evaluation for 4S Emergency Planning Zone"

Enclosed is a copy of the non-proprietary "Evaluation for 4S Emergency Planning Zone" for the 4S (Super-Safe, Small and Simple) reactor plant that is currently the subject of a pre-application review among NRC, Toshiba, and its 4S affiliates including Japan's Central Research Institute for Electric Power Industry (CRIEPI).

The pre-application review for the 4S reactor commenced in the fourth quarter of 2007. Pre-application review meetings were held among NRC, Toshiba and the 4S affiliates in October 2007, and February, May and August 2008.

Additional technical reports pertaining to the 4S design will be submitted as the pre-application review progresses. If you have any questions regarding this document, please contact Mr. Tony Grenci of Westinghouse at (623) 271-9992, or grencit@westinghouse.com.

Very truly yours,

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Enclosures: Technical Report "Evaluation for 4S Emergency Planning Zone"

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Document Number: AFT-2011-000080rev004(2)
PSN Control Number: PSN-2011-1042

Evaluation for 4S Emergency Planning Zone

September 2011

TOSHIBA CORPORATION

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LIST OF ACRONYMS AND ABBREVIATIONS

4S	Super-Safe, Small and Simple
ALWR	Advanced Light Water Reactor
ANS	American Nuclear Society
ASLBP	Atomic Safety and Licensing Board Panel
ATWS	anticipated transient without scram
CDE	committed dose equivalent
CEDE	committed effective dose equivalent
CFR	Code of Federal Regulations
CRBR	Clinch River Breeder Reactor
DA	Design Approval
DBA	design basis accident
EDE	effective dose equivalent
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
EPZ	emergency planning zone
FP	fission product
GIF	Generation IV International Forum
IAEA	International Atomic Energy Agency
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles
IRACS	intermediate reactor auxiliary cooling system
LWR	light water reactor
NEI	Nuclear Energy Institute
NPP	nuclear power plant
NRC	Nuclear Regulatory Commission
NUREG	U.S. Nuclear Regulatory Commission Regulation
PAG	Protective Action Guide
PRA	probabilistic risk assessment
PRISM	Power Reactor Innovative Small Module
PWR	pressurized water reactor
RG	Regulatory Guide
RHR	residual heat removal
RVACS	reactor vessel auxiliary cooling system
SAR	Safety Analysis Report
SFR	sodium-cooled fast reactor
SMR	Small and Medium Sized Reactor
SRM	Staff Requirement Memorandum
TEDE	total effective dose equivalent

1 INTRODUCTION

1.1 OVERVIEW OF EPZ CONCEPT IN THE UNITED STATES

Toshiba has been submitting technical reports to the U.S. Nuclear Regulatory Commission (NRC) as part of the pre-application review of the 4S sodium-cooled fast reactor (SFR) design. This report is the sixth in that series of reports.

One promising feature of the 4S design is the potential for its enhanced and passive safety features to permit licensing with modified requirements for emergency response to include reducing the size of the evacuation zone close to the plant boundary. This is a desirable feature where proximity to the plant for the end user is necessary, such as for water desalination or district heating applications.

Current regulations for nuclear power plants (NPPs) and their sites require adequate protective measures to be taken in the event of a radiological emergency. This requirement has been manifested as a requirement for emergency planning in an area within 10 miles of the plant boundary in the U.S. This prescribed distance is the same for all current operating NPPs, and was selected to conservatively cover all nuclear power plant designs existing at the time the rule was established.

Two licensees have previously petitioned the NRC to allow for a reduction in the size of the emergency planning zone (EPZ) for traditional light water reactors (LWRs) [1]. These requests were unsuccessful for reasons of completeness. In 1985, the licensee for Calvert Cliffs (PWR, Unit 1: 873 MWe, Unit 2: 863 MWe) requested exemptions and license amendments to allow for a reduction in the 10-mile EPZ to 2 miles. In 1986, applicants for the Seabrook nuclear power plant (PWR, 1244 MWe) requested a waiver to allow for reduction in the 10-mile EPZ to 1 mile. The technical argument supporting these requests was that a site-specific analysis of design basis and severe-accident risks showed a decrease in these risks relative to the risks considered in NUREG-0396 [2]. In regard to the Calvert Cliffs exemption request, the NRC staff concluded that it could not consider the request because the NRC was still studying severe-accident issues (April 11, 1988, letter from S. Varga [NRC] to J. Tiernan [BG&E]). In regard to the Seabrook petition, the Atomic Safety and Licensing Board Panel (ASLBP) concluded that *"there are a number of areas wherein it appears the Applicant had not presented full and complete results sufficient to inspire confidence that their motion deserves further consideration at this time"* (ASLBP 82-471-02-02).

So far, these historical developments have not resulted in new provisions in the regulations to account for the significant safety improvements in plant operation and design achieved in the nuclear industry and their impact on emergency planning zone size. On the other hand, the regulations of Appendix E to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50, Appendix E) indicate that "*The size of the EPZs also may be determined on a case-by-case basis for gas cooled nuclear reactors and for reactors with an authorized power level less than 250 MW thermal.*" Since all proposed versions of the 4S design are smaller than 250 MW thermal, this portion of Appendix E may be considered to apply to determining the EPZ size for 4S.

1.2 EPZ REQUIREMENTS FOR SMRS

The NRC has issued various papers with respect to the definition of emergency planning procedures for evolutionary reactors. For example, in SECY-93-092 [3], released in 1993, the staff raised the following issue: "*Should advanced reactors with passive advanced design safety features be able to reduce emergency planning zone and requirements?*" Although the staff suggested no change to existing regulations governing EP for advanced reactors, it mentioned that regulatory direction would be provided at or before the start of the design certification phase so that EP implications for design could be addressed.

Also, the Electric Power Research Institute (EPRI) has indicated that a significant reduction in the EPZ distance was possible for Advanced Light Water Reactors (ALWRs) by using a combination of probabilistic and deterministic approaches. This need was identified by the International Atomic Energy Agency (IAEA) International Project on Innovative Nuclear Reactor and Fuel Cycles (INPRO) as well as by the Generation IV International Forum (GIF).

Finally, the American Nuclear Society (ANS) established a Small and Medium Sized Reactor (SMR) President's Special Committee in 2010 to propose solutions to SMR generic licensing issues and prepare a set of white papers for use by the SMR community [4]. So far, the committee has identified 24 SMR generic issues, which includes emergency planning[1]. A white paper regarding emergency planning zone sizing for SMRs is scheduled for completion in the near future.

According to the ANS interim report, a clear trend emerges in the conclusions and recommendation of the completed white papers that the current U.S. nuclear regulations are primarily focused on the safety and security of large LWRs. The documented papers illustrate the incompatibilities of certain aspects of the current licensing rules with SMR design. In general, applicants would have three possible approaches for licensing SMRs where these incompatibilities exist [4]:

- Seek exemptions to current rules
- NRC rulemaking
- Legislative changes

Although, as noted above, the case-by-case exception to Appendix E based on thermal power currently provides a process to seek a smaller EPZ size for a 4S reactor, future rulemaking or legislation may provide additional options.

The Nuclear Energy Institute (NEI) SMR Licensing Task Force has also engaged in efforts to develop position papers on multiple generic licensing issues. The position paper regarding EPZ sizing is under development and scheduled to be submitted in mid-2011 [5]. This paper will be complementary to the initial paper on general EPZ issues. This paper will evaluate the current basis for EPZ sizing and is anticipated to propose a framework for developing EPZs appropriately sized for SMRs.

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1. Status in March 2011.

The conservative emergency planning requirements, especially for a small reactor, could cause an undue burden on plant utility both in construction and operation. For example, it may be required to build unnecessary infrastructure and develop and implement complicated procedures to satisfy the requirement for a large EPZ around the plant. An excessive EPZ around a small NPP, in spite of the reactor's compactness, may also send an incorrect message to the public with respect to the safety of advanced design SMRs.

The 4S design, having a low power level compared to large LWRs, has the potential to redefine the EPZ by taking into account its actual margin of safety. The 4S's advanced and safer passive design can further reduce risk to the public and therefore offer a possibility to reduce the emergency plan and evacuation requirements. Achieving licensing with a reduced EPZ provides significant benefits to the general public and plant owners as follows:

- Increased public acceptance of the nuclear power plant due to being treated more like any other non-nuclear power plant.

- Reduced costs (training, operational, administrative) to maintain simplified infrastructure and operational procedures.

- Supports a wider choice of site locations in areas with relatively high population density.

In summary, the rationale for different EPZ requirements for SMRs is identified as follows:

- Existing recognition of smaller plant characteristics
- Technical rationale for pursuing different EPZ, i.e., inherent safety characteristics
- Possible impact on stakeholders

Section 2 describes the purpose and scope of this report. Section 3 presents the process of EPZ and release scenario definition. In Section 4, release scenario definition is described and dose evaluation is performed in Section 5. The 4S EPZ identification is shown in Section 6. Section 7 summarizes the main conclusions of the report.

This evaluation concludes that a reduced EPZ, whose radius is remarkably small (about 200 m), can be proposed by taking consideration of the 4S's inherent safety characteristics.

2 PURPOSE AND SCOPE

2.1 PURPOSE

The purpose of this report is twofold:

1. To document the process and data regarding the radiological consequence analysis to define the EPZ for the 4S.

2. To obtain feedback from the NRC staff on the presented material either in writing or in a meeting at the staff's convenience. Such feedback will be utilized by the 4S project in completing its DA application.

2.2 SCOPE

This report describes the process for determining the release scenario and performing a dose evaluation analysis to define the EPZ size. The report identifies the innovative and passive 4S design features that enable the EPZ modification. The report explains the implementation of the modified EPZ for the 4S by reference to the dose evaluation results derived herein.

The methodology presented here is based on deterministic methods intended to justify licensing with reduced emergency planning requirements. It is articulated in the following steps:

Step 1: Specify the events affecting the EPZ.

Step 2: Select the most severe event.

Step 3: Carry out radiological consequence analysis for the selected events.

Step 4: Define the EPZ for the 4S by adapting the dose evaluation.

The scope of this report includes the following topics:

- Consideration of the events and accidents affecting the EPZ
- Identification of the process of release scenario definition
- Performance of the dose evaluation including reviewing calculation conditions
- The definition of the 4S EPZ based on results of the dose evaluation

3 PROCESS OF EPZ DEFINITION

3.1 SPECIFYING THE EVENTS AFFECTING THE EPZ

The current EPZ concept is based on NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants" [2], whose main purpose is described as follows: "*provide a basis for Federal, State and local government emergency preparedness organizations to determine the appropriate degree of emergency response planning efforts in the environs of nuclear power plants.*"

This guidance was prepared by the NRC and Environmental Protection Agency (EPA) Task Force on Emergency Planning (the Task Force) in 1978. One of the major recommendations of the Task Force was to take into consideration an enlarged accident spectrum, which not only considers the design basis accident (DBA), but also more severe accidents, commonly known as Class 9 events. This term refers to those accidents leading to "*total core melt and consequent degradation of the containment boundary and those leading to gross fuel cladding failure or partial melt with independent failures of the containment boundary.*" In summary, the Task Force recommended considering severe accidents described as follows:

- Total core melt and degradation of the containment, and
- Gross fuel cladding failure or partial melt with failures of the containment boundary.

The Commission requested for the staff to submit recommendations for proposed technical criteria and methods to use to justify simplifications of existing emergency planning requirements in a Staff Requirement Memorandum dated July 30, 1993 [6]. In this document, the Commission further stated that "*work on EP should be closely correlated with work on Accident Evaluation and Source Term, in order to avoid unnecessary conservatism. Also, the work on EP for advanced reactors should be coordinated with the approach for evolutionary and passive advanced reactors.*"

In response to the request, the staff issued SECY-97-020 [1] to perform an evaluation to develop technical criteria for EP for evolutionary and advanced reactor designs. The staff determined that the NUREG-0396 approach using consequences tempered by probability considerations was also appropriate for evolutionary and advanced plants. The staff also indicated that, due to the similarity in the potential consequences of severe accidents between those plants and current ones, rigid application of the technical criteria derived from the Task Force study against those advanced reactor designs indicated that no changes to EP requirements were warranted.

At the same time, by verifying how the innovative safety features and characteristics of these advanced plants tend to affect and modify the results that the Task Force or NUREG-0396 considered, the staff concluded that changes to EP requirements may be warranted only if the technical criteria were modified to account for the following factors:

- The lower probability of severe accidents
- The longer time period between accident initiation and release of radioactive material
- The most severe accident conditions associated with evolutionary and passive advanced LWRs

3.2 JUSTIFICATION OF REVISED EPZ CRITERIA FOR 4S

An evaluation follows of whether application of these criteria for evolutionary and advanced reactors, such as the 4S, might indicate that changes to EPZ requirements are warranted.

(1) Lower Probability of Severe Accidents

The 4S design has a lower, negligibly small probability of severe accidents leading to core melt due to its inherent safety aspects, including use of redundant shutdown systems and unique heat removal systems. The 4S has two independent shutdown systems of different design principles, namely a reflector and a shutdown rod. Six separately controllable segments combine to form the cylindrical reflector outside the core barrel. The subassembly at the reactor core center has a cylindrical reactor shutdown rod surrounded by six circumferentially divided, fan-shaped fixed absorbers, contained in a hexagonal wrapper tube. The reflector is the primary shutdown system and the shutdown rod is available as a backup shutdown system. The reactor shutdown system is composed of these two reliable, independent and diverse safety shutdown systems actuated by independent detection signals.

Also, the 4S residual heat removal (RHR) systems remove heat efficiently to keep the reactor temperature stable even in the case where both of the shutdown systems are not available. The reliable RHR systems (Figure 3.1) consist of the intermediate reactor auxiliary cooling system (IRACS) and the reactor vessel auxiliary cooling system (RVACS). The IRACS removes decay heat by making use of an air cooler in the intermediate heat transport system. The RVACS removes the decay heat with natural convection from air outside the reactor guard vessel, which serves as a heat collector between the cylindrical underground concrete wall around the guard vessel and the reactor vessel. Ambient cold air descends between the underground cylinder wall and the heat collector and turns upward at the lower end of the heat collector cylinder, then rises between the heat collector and guard vessel. As shown in Figure 3.2, radiation heat from the reactor vessel is removed with natural convection heat transfer in the gap between the guard vessel and the heat collector. This process takes place under all plant conditions and for all design events entirely by natural phenomena without the intervention of any active equipment. These RHR systems with inherent safety features have enough capacity to remove the decay heat even with only one system (either IRACS or RVACS) available.

Figure 3.3 shows the passive and inherently safe core characteristics of 4S. Core materials have negative temperature coefficients and low fuel temperature during the operation. These features provide a feedback mechanism, enhancing the inherent safety characteristics for some transients, and also provide power stability [7]. There could be several triggers to lead core melt accident as shown in Table 3.1, and the 4S has various provisions to mitigate this accident. Not only internal event, external natural event has a possibility to cause severe accident [8]. Table 3.2 shows the list of postulated extreme external events and the 4S design protection against these events.

Thus, the most severe accident corresponding to a Class 9 event (core melt) for the 4S could be mitigated by taking into account these characteristics.

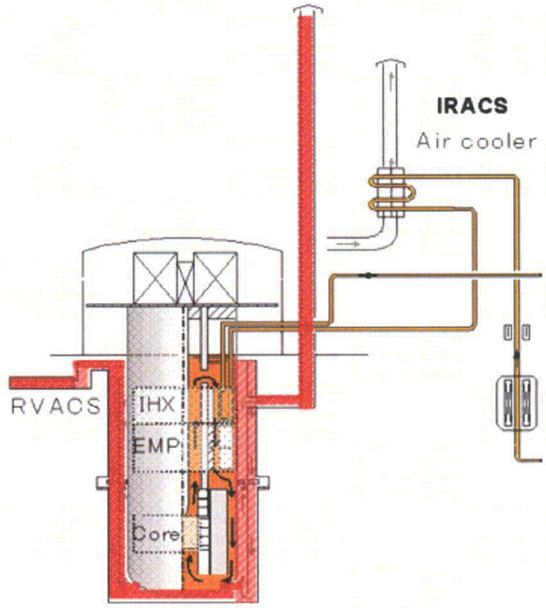


Figure 3.1 Passive Decay Heat Removal System,
RVACS: Natural Air Draft Outside the Guard Vessel,
IRACS: Natural Circulation of Sodium and Air Draft at Air Cooler

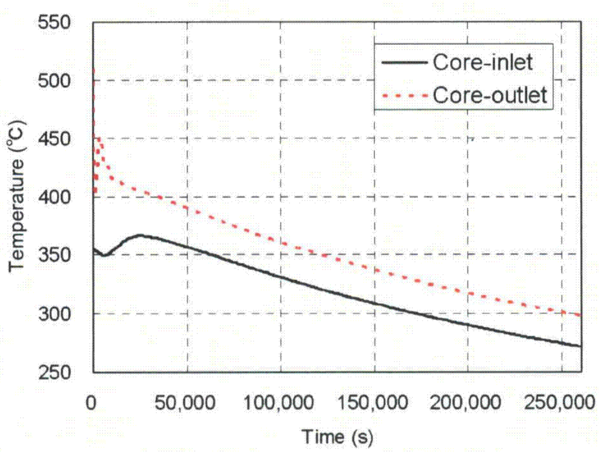


Figure 3.2 Core Temperature During Loss of Power with Only Natural Circulation

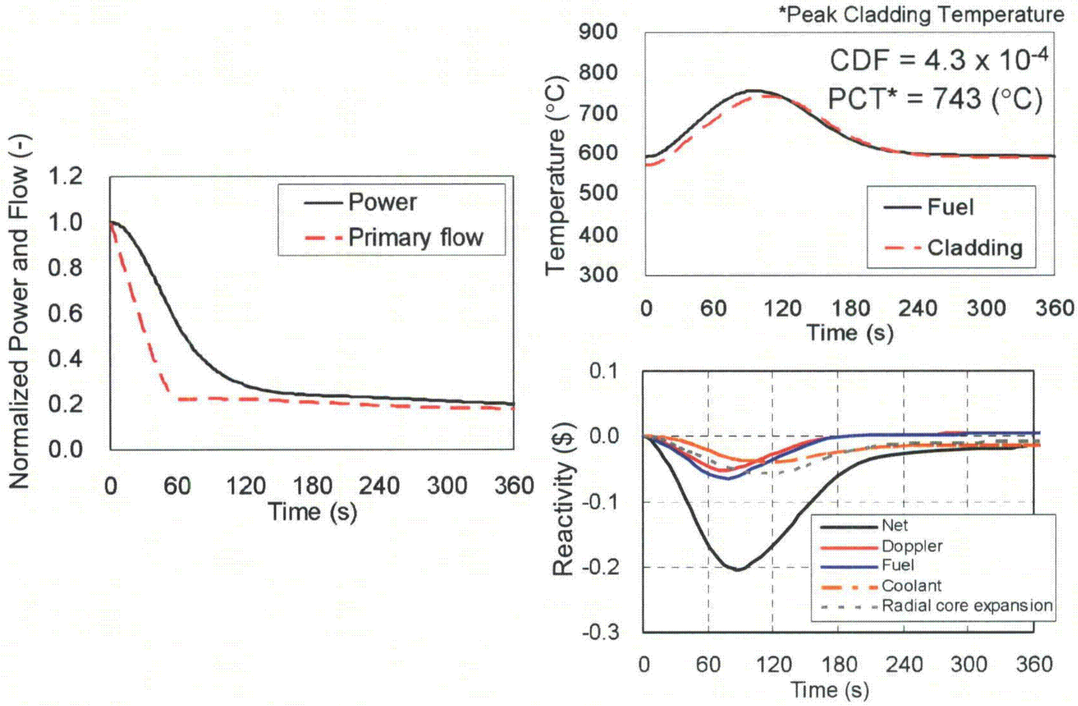


Figure 3.3 Passive Shutdown Capability (ATWS Reference)
(a) Power and primary flow, (b) Fuel and cladding peak temperature, (c) Reactivity

CDF: Cumulative Damage Fraction

Table 3.1
Selection of Initial Events for Possibly Core Melt in a Fast Reactor

	Initial Events	4S Design Against Core Melt
CRBR [9]	ATWS	Metallic fuel, negative feedback reactivity, and low power density
PRISM [10]	All control rods withdraw without scram	Very low reactivity addition rate at operation, heat removal with natural circulation, redundant mechanical shutdown system, negative feedback reactivity, low power density, and mechanical stop for rapid control rod withdrawal
	Instantaneous loss of flow without scram	Two-cascaded EM pump system to prolong flow coast down, metallic fuel, negative feedback reactivity, low power density, and natural circulation
	Fuel loading error	Similar enrichment level and similar flow rate for both core regions, low power density, mechanical prevention system
	Local blockage in a fuel assembly	Very low impurity occurrence due to no refueling and EM pump(non-mechanical) Redundant flow paths of inlet module
	75% blockage of RVACS ducting	Backup redundant and diverse systems (IRACS) and thermal inertia

Table 3.2
List of Postulated External Events and 4S's Protection

External Events	4S's Protection by Structural Design
Earth quakes	Supporting the reactor building by seismic isolator.
Extreme meteorological conditions (temperature, drought, et al.)	Use of redundant shutdown systems and passive decay heat removal system with natural circulation Air cooling is a final heat sink, not necessary emergency power supply system
Floods	
Lightning	
Cyclones	Built a reactor building into underground as shown in Figure 3.4 The roof of reactor building above the ground level contains a concrete-filled steel structure bio-shield plug and is designed to withstand tornado-class impact.

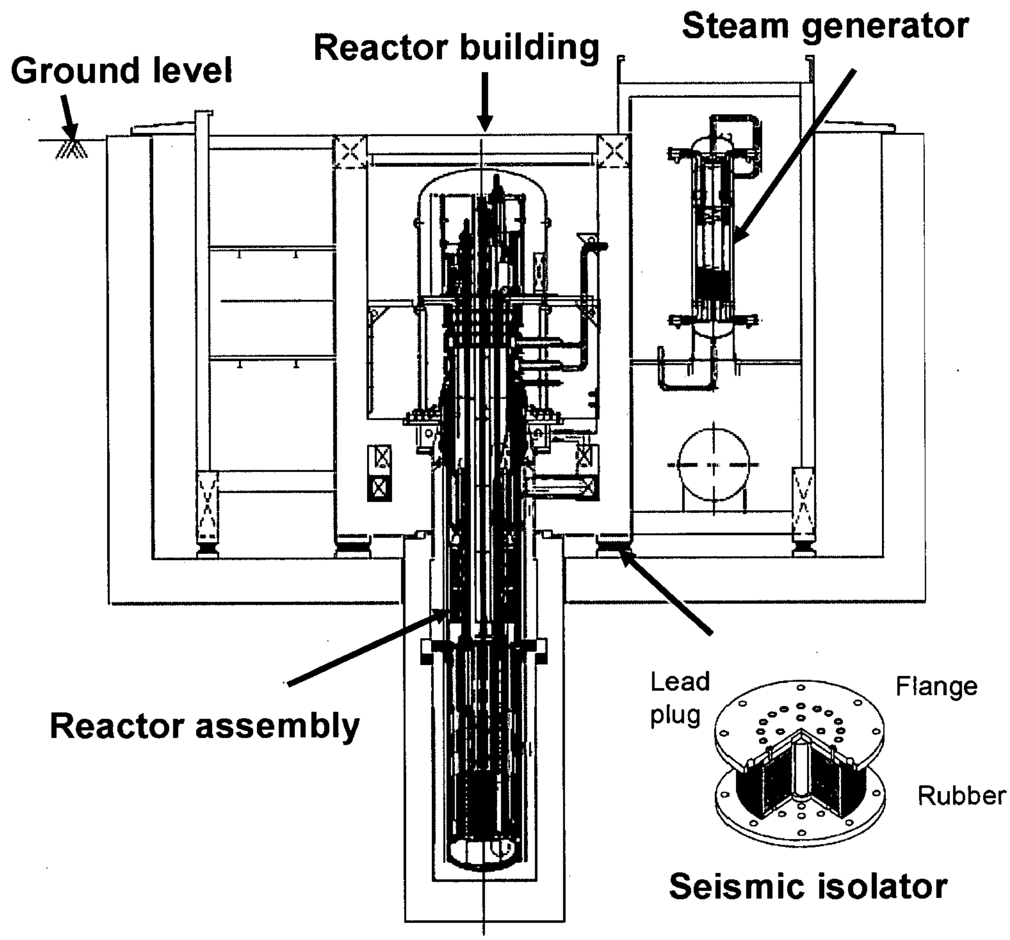


Figure 3.4 Reactor Building

(2) Longer Time Period between Accident Initiation and Release of Radioactive Material

In case of a severe accident, there could be a time lapse between recognition of an initial event and the start of the release with respect to consequent degradation of the containment boundary. For LWRs, the scenario of containment vessel degradation and release of radioactive material due to severe accident is assumed to be as follows:

- Accumulation of decay heat in the containment vessel
- Vapor pressure buildup in core due to heat removal system failure
- Degradation of the containment vessel (result of no counter measures taken against overpressure in the containment vessel)

In the case of SFRs, sodium has such a high boiling point compared to water that SFRs have enough margin to avoid overpressure in the reactor vessel caused by coolant boiling. Thus, with respect to the SFRs, degradation of the containment vessel due to overpressure is not assumed and no other degradation mechanisms have been identified.

Therefore, 4S and other SFRs should have a longer time period between the initial accident and any release of radioactive material by taking into account the avoidance of overpressure in the reactor vessel.

In the light of establishing EPZ, preventing 4S from large early release is ranked as the highest priority to keep necessary evacuation arrangement period. The 4S is designed to keep the radioactive material in the reactor in case of the prompt re-criticality accident.

(3) Most Severe Accidents Associated with Evolutionary and Passive Advanced LWRs and SMRs

As mentioned above, 4S has the potential to address the Class 9 event without a core melt (hereafter mod-Class 9 event) by taking into consideration its inherent safety aspects. As for 4S's containment vessel, or top dome could be protected against failure or degradation[2] by preventing from severe accident such as a core melt. As shown in Table 3.2, it is also designed to protect the reactor against extreme external natural event. Therefore, 4S's containment boundary could satisfy the design leak rate for a severe accident without failure or degradation. On the other hand, a fuel clad is supposed during operation. Although cladding fraction could be limited, gross fuel cladding is conservatively assumed. Therefore, the 4S mod-Class 9 event is identified as follows:

- Mod-Class 9 event for the 4S: "Gross fuel cladding failure in the core"

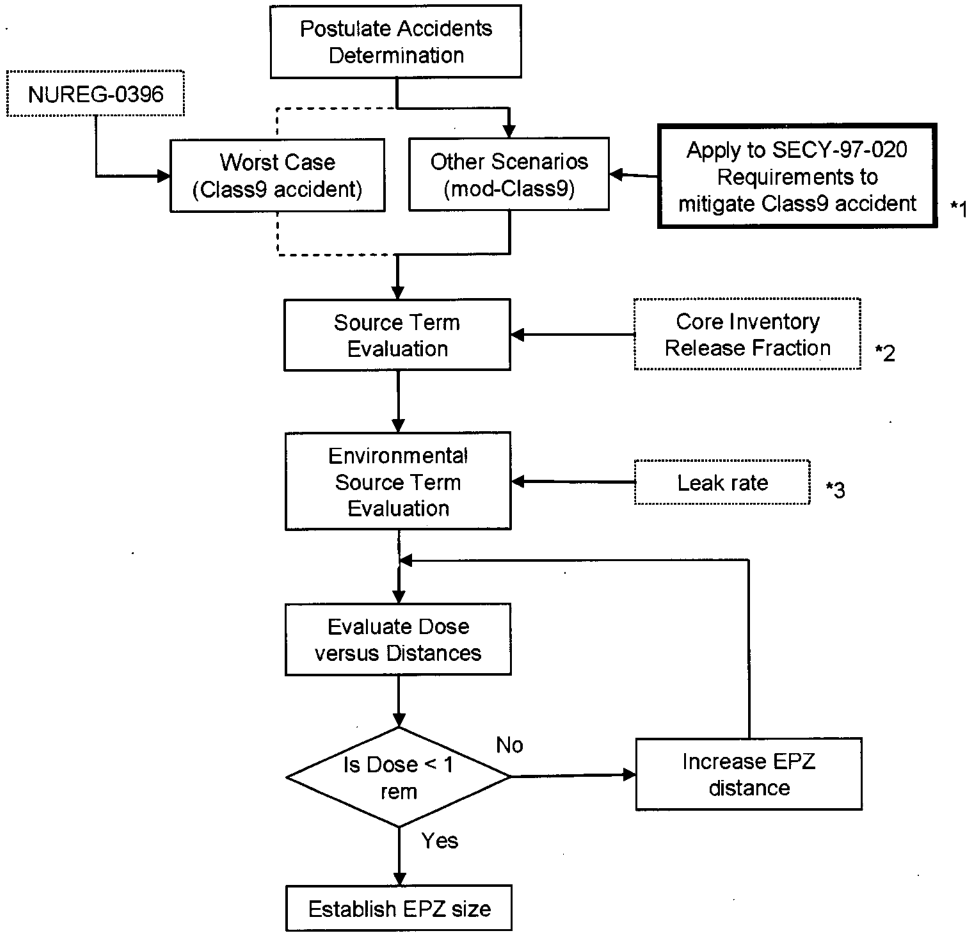
On the other hand, DBAs associated with the 4S that may result in a radioactive release have also been examined. DBAs are postulated accidents that are used to set design criteria and limits for the design and sizing of safety-related systems and components. Postulated accidents are unanticipated occurrences, that is, they are postulated but not expected to occur during the life of the nuclear power plant. The following DBAs have been identified by taking into consideration the 4S design [7]:

- Primary sodium reactor vessel leak
- Primary sodium cold trap leak
- Primary argon cover gas leak
- Radioactive material leak during fixed absorber withdrawal

After comparing these DBAs with the mod-Class 9 event, the mod-Class 9 event has been determined to be the most comprehensive and severe accident, since these other DBAs do not result in gross cladding failure and degradation of the containment.

Figure 3.5 presents EPZ development methodology applied to 4S.

[2] In this report, failure and degradation of containment are defined as loss of boundary and loss of design-base boundary function, respectively.



*1 Discussed in Sec. 3.2
*2 Presented in Table 4.2
*3 Presented in Table 4.3

Figure 3.5 EPZ Development Methodology for 4S

4 RELEASE SCENARIO DEFINITION

4.1 SOURCE TERM AND INVENTORY DEFINITION

For this evaluation, the radionuclide groups and the elements making up each group are set in reference to NUREG-1465 [11], except for the additional inclusion of uranium in the cerium group [11], here replaced with "Actinide".

The modified radionuclide groups and elements are shown in Table 4.1.

Table 4.1
Radionuclide Groups and Elements

Group	Element
Noble gases	Xe, Kr
Halogens	I, Br
Alkali metals	Rb, Cs
Tellurium group	Te, Sb, Se
Barium, strontium	Ba, Sr
Noble metals	Ru, Rh, Pd, Mo, Tc, Co
Actinide, Lanthanides	Ce, Pu, Np, U, La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Am, Cm

Generally, the radiological inventory in a core is proportional to the thermal power. Therefore, the radiological inventory value is conservatively estimated at the point where 30 years of operation have elapsed since plant startup. The radiological core inventory may include errors in excess of 10 percent in magnitude due to the uncertainty of the cross-section library, etc. To allow for these effects, the nominal value of the radiological inventory is multiplied by a factor of 2 as an uncertainty margin for this analysis.

4.2 LEAK PATH

Figure 4.1 shows the leak path for the 4S source term evaluation. The leak path is determined by considering the following stages: (1) migration from the core into the primary coolant; (2) migration from the primary coolant into the cover gas space; (3) leakage from the cover gas space to the top dome, or containment vessel, via the shielding plug and associated equipment; (4) leakage from the containment vessel into the environment.

The migration rate was established with consideration of the chemical reaction between each element and sodium as described in [7]. The migration rate from sodium to the cover gas space is calculated by taking into account the high holding ability of sodium due to its affinity for fission products (FPs). The sodium temperature during the accident is conservatively assumed to be up to 600°C. The migration rate is calculated on the basis of the 4S system design including the information on the primary coolant and cover gas space volumes as described in [7]. Table 4.2 shows the migration fractions from the core to the sodium coolant and from the sodium to the cover gas space. Table 4.3 shows the leak rate of cover gas and top dome.

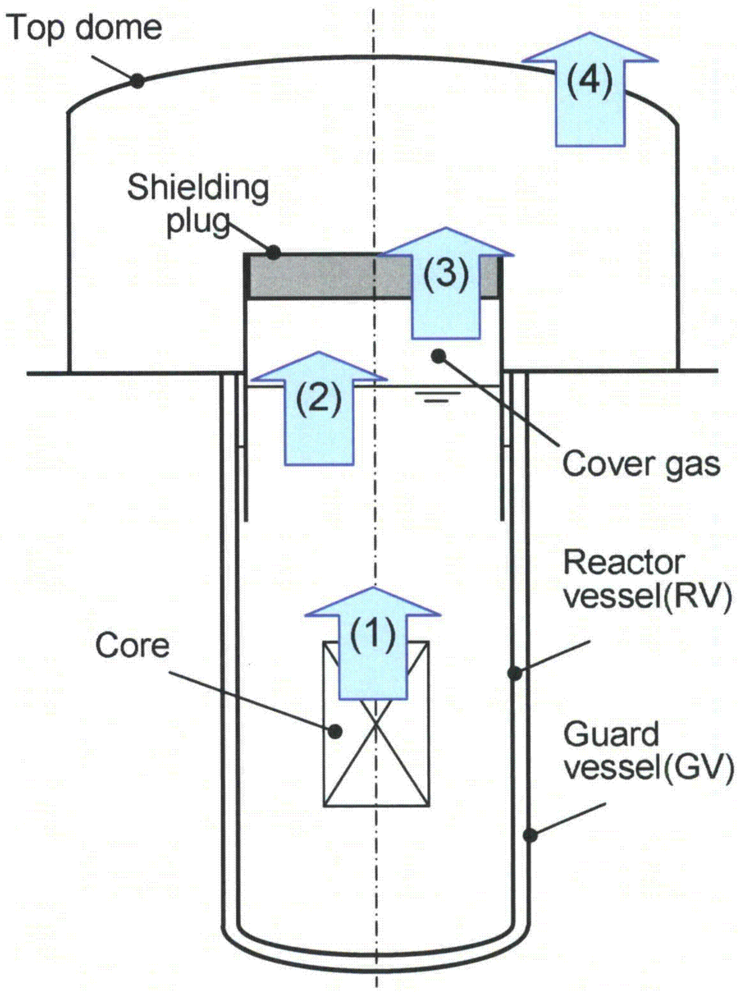


Figure 4.1 Leak Path for 4S Source Term:
(1) Core to Primary Sodium, (2) Sodium to Cover Gas,
(3) Cover Gas to Top Dome (Containment Vessel), (4) Top Dome to Environment

Table 4.2
Release Fractions from Core into Cover Gas

	(1) Migration Fractions* from Core to Primary Sodium (Core Inventory Fraction Released into Primary Sodium)	(2) Migration Fraction* from Primary Sodium into Cover Gas	(3) Total Migration Fraction*
Noble gases	1.0	1.0	1.0
Halogens	1.0	3×10^{-6}	3×10^{-6}
Alkali metals	1.0	5×10^{-5}	5×10^{-5}
Tellurium group	1.0	2×10^{-6}	2×10^{-6}
Barium, strontium	1.0	2×10^{-6}	2×10^{-6}
Noble metals	0.1	2×10^{-6}	2×10^{-7}
Actinide, Lanthanides	0.1	2×10^{-6}	2×10^{-7}

* Migration fraction is based on [7].

Table 4.3
Leak Rate of Cover Gas and Top Dome (Containment Vessel)

Case	Primary Cover Gas Leak Rate	Containment Vessel Leak Rate	Remarks
1	0.1%/day	1%/day	-
2	0.1%/day	10%/day	-
3	0.1%/day	100%/day	-
4	10%/day (*)	1%/day	(*) Design leak rate for a severe accident
5	10%/day	100%/day	-

As previously mentioned, SFRs can have a longer time period between accident initiation and the release of radioactive material as a result of the lack of any overpressure in the reactor vessel. However, the selected scenario and the resulting mechanisms do not assume credit for this characteristic. Therefore, the degradation condition of the containment boundary is given as a priori for the dose evaluation except design leak rate for a postulated accident as shown in Table 4.3. Shield plug, forming primary cover gas boundary, is designed to keep primary cover gas leak rate below 10%/day for a postulated accident. As for containment vessel, it is required to be tolerant of high pressure and temperature in case of a postulated accident. By taking into account these condition and capability of primary cover gas boundary, containment vessel leak rate is designed to keep below 1%/day for a postulated accident.

Table 4.4 presents a list of event scenarios and parameters for dose evaluation.

Table 4.4
Event Scenarios and Parameters for Dose Evaluation

Scenario		Parameter	Remarks
1	Gross fuel cladding failure	Fuel failure fraction: 100%	Based on Class 9 event.
2	Migration from core into primary sodium	Migration fraction is shown in Table 4.2	-
3	Migration from primary sodium into cover gas		
4	Leakage from primary cover gas boundary	Leakage rate: ranging from 0.1 to 10%/day shown in Table 4.3	Ranging up to design leak rate for postulated accident
5	Failure of the containment boundary	Leakage rate: ranging from 1 to 100%/day to instant release, shown in Table 4.3	Ranging from design leak rate to degradation of containment

5 DOSE EVALUATION

5.1 ATMOSPHERIC DIFFUSION MODEL

The basic equation for atmospheric diffusion from an elevated release is [12, 13]:

$$\chi/Q = \frac{1}{\pi \sigma_y \sigma_z u} \exp\left(-\frac{h^2}{2\sigma_z^2}\right)$$

where,

χ	=	centerline value of the ground-level concentration (Ci/m^3)
Q	=	amount of material released (Ci/m^3)
u	=	wind speed (m/s)
σ_y	=	horizontal standard deviation of the plume (m)
σ_z	=	vertical standard deviation of the plume (m)
h	=	effective height of release

For time periods of longer than 8 hours, the plume from an elevated release is assumed to meander and spread uniformly over a 22.5° (1/16) sector as shown in the following equation:

$$\chi/Q = \frac{2.032}{x \sigma_z u} \exp\left(-\frac{h^2}{2\sigma_z^2}\right) \quad , \quad 2.032 = \sqrt{\frac{2}{\pi}} \times \frac{16}{2\pi}$$

where,

x	=	distance from the release point (m)

The 0–8 hour ground-level release concentrations may be reduced by a factor ranging from 1 to a maximum of 3 for additional dispersion produced by the turbulent wake of the reactor building in calculating potential exposures. The basic equation for atmospheric diffusion from ground level and related atmospheric conditions is given as follows:

$$\chi/Q = \frac{1}{\pi \sigma_y \sigma_z u}$$: For 0–8 hour ground-level release

For time periods longer than 8 hours, the plume is assumed to meander and spread uniformly over a 22.5° (1/16) sector. The resulting equation and related atmospheric conditions are given as follows:

$$\chi/Q = \frac{2.032}{\pi \sigma_z u}$$: For time periods of greater than 8 hours

The atmospheric diffusion model for ground-level release expressed above is based on the information shown in Table 5.1.

Table 5.1
Atmospheric Conditions

Time Following Accident	Pasquill Type	Wind Speed	Wind Direction
0–8 hours	Pasquill F	1 m/s	Uniform direction
8–24 hours	Pasquill F	1 m/s	Variable direction within a 22.5°sector
24–96 hours	40% Pasquill D 60% Pasquill F	3 m/s 2 m/s	Variable direction within a 22.5°sector
After 96 hours	33.3% Pasquill C 33.3% Pasquill D 33.3% Pasquill F	3 m/s 3 m/s 2 m/s	33.3% frequency in a 22.5°sector

5.2 DOSE CONVERSION

Exposed dose is given by total effective dose equivalent (TEDE), which is the sum of the external exposures and internal exposures. External exposure is expressed as the effective dose equivalent (EDE), and internal exposure is expressed as the committed effective dose equivalent (CEDE), which is the sum of the products of the committed dose equivalents (CDE) for each of the body organs or tissues that are irradiated, multiplied by the weighting factors. The CEDE provides an estimate of the lifetime radiation dose, generally 50 years, to an individual from radioactive material taken into the body through either inhalation or ingestion.

These exposures are given as follows:

$$D_{EDE} = \sum_i K1_i \sum_j Q_{ij} (\chi / Q)_{ij} \,,$$

$$D_{CEDE} = \sum_i K2_i \sum_j Q_{ij} B_j (\chi / Q)_{ij}$$

where,

$K1_i$ = EDE dose conversion factor for radionuclide i (rem-m^3/Ci-s)

$K2_i$ = CEDE dose conversion factor for radionuclide i (rem/Ci)

B_j = breathing rate during time period j (m^3/s)

Q_{ij} = amount of material released for radionuclide i during time period j (Ci)

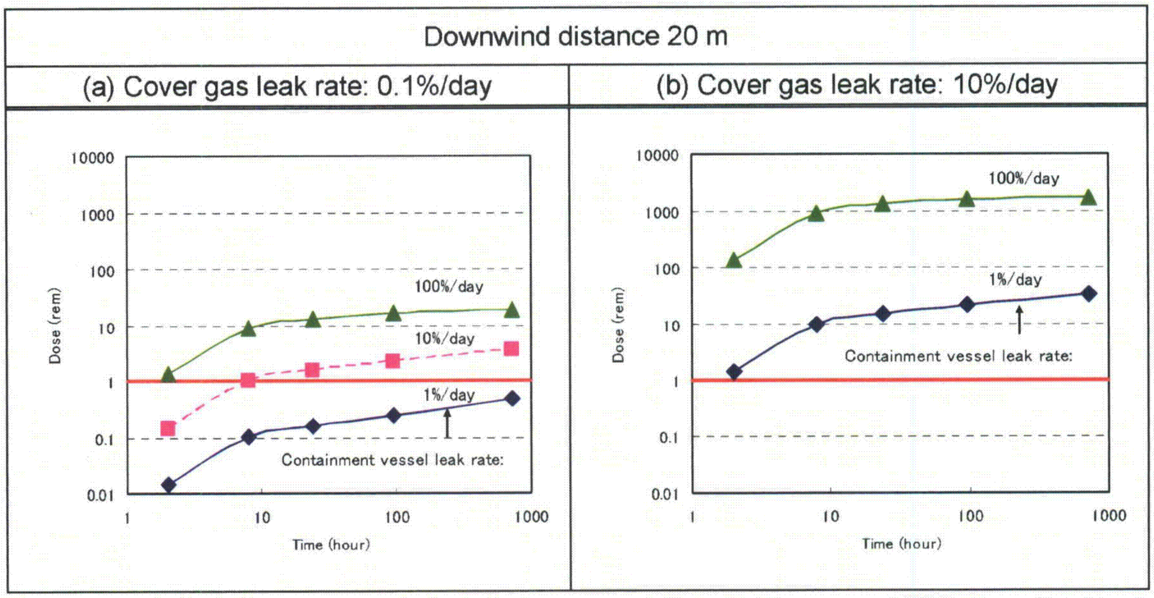
$(\chi / Q)_{ij}$ = atmospheric dispersion factor, or relative ground concentration for nuclide i during time period j (s/m^3).

EDE and CEDE dose conversion factors are given in EPA Federal Guidance Report No.12 and No.11, respectively [14, 15].

5.3 DOSE EVALUATION RESULTS

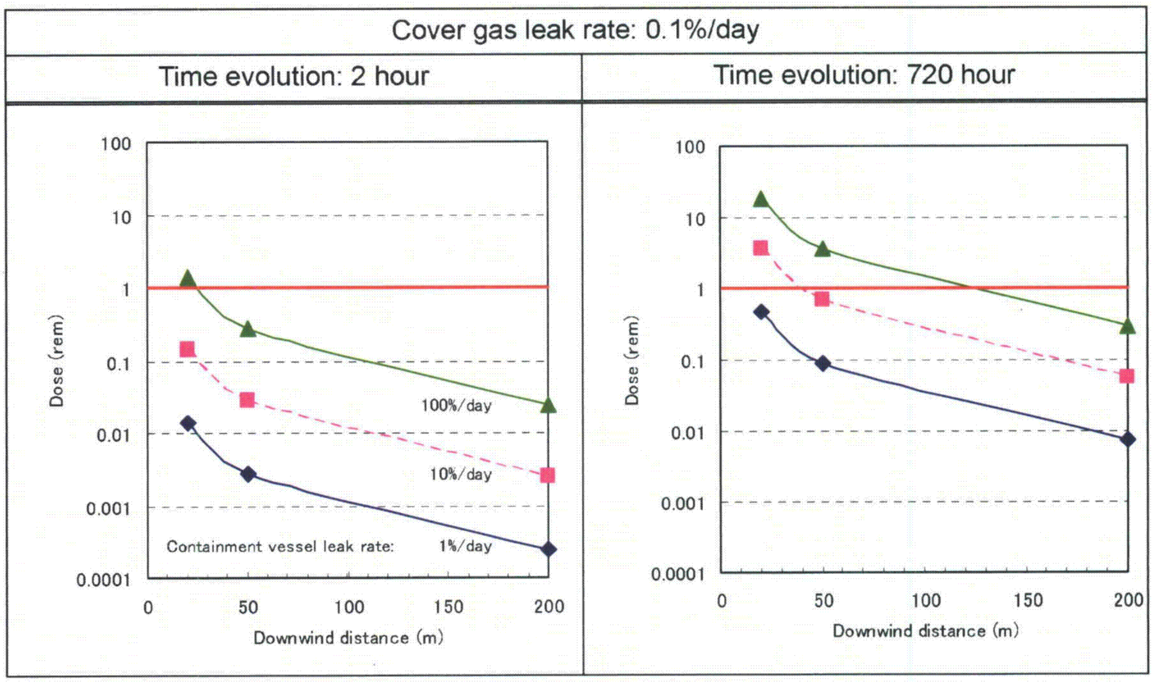
Radiological consequence evaluation results based on mod-Class 9 events are shown in this section. Figure 5.1 shows the dose rate (TEDE) of time evolution from 2 hours to 720 hours (1 week) at a downwind distance of 20 m. Figure 5.1 (a) and (b) correspond to the results for Cases 1 through 3, and Cases 4 and 5 in Table 4.3, respectively. Figure 5.2 also shows the same values at different distances and Table 5.2 though Table 5.4 summarize dose for various values of evolution time and leak rate of cover gas and containment vessel. These results show that the dose increases almost proportionally within 24 hours at the evolution time, and afterwards, the dose is almost saturated.

The breakdowns of EDE and CEDE at different evolution times are shown in Figure 5.3 and Figure 5.4. It is shown that the contribution of EDE to TEDE is predominant compared with CEDE though the evolution time. Figure 5.3 and Figure 5.4 also indicate that EDE is saturated in 96 hours (4 days) and CEDE is roughly proportional to time. These tendencies are related to the differences in half-lives for the nuclides. EDE is mainly related to noble gases whose half-life is relatively short, ranging from hours to days. The predominant nuclides contributing to CEDE have longer half-lives, measured in years.



Note: Red line represents allowable dose limit to evaluate the distance of EPZ, described in Section 6.

Figure 5.1 Time Evolution of Dose (TEDE) at Downwind Distance 20 m



Note: Red line represents allowable dose limit to evaluate the distance of EPZ, described in Section 6.

Figure 5.2 Dose (TEDE) at Downwind Distance

Table 5.2
TEDE at 20 m Downwind Distance (rem)

Cover gas leak rate: 0.1%/day						
	Containment Vessel Leak Rate	**2h**	**8h**	**24h**	**96h**	**720h**
Case 1	1%/day	1.42E-02	1.02E-01	1.58E-01	2.39E-01	4.82E-01
Case 2	10%/day	1.42E-01	1.01E+00	1.55E+00	2.28E+00	3.61E+00
Case 3	100%/day	1.38E+00	9.33E+00	1.34E+01	1.66E+01	1.86E+01
						Dose unit: rem
Cover gas leak rate: 10%/day						
	Containment Vessel Leak Rate	**2h**	**8h**	**24h**	**96h**	**720h**
Case 4	1%/day	1.41E+00	1.01E+01	1.55E+01	2.27E+01	3.49E+01
Case 5	100%/day	1.38E+02	9.24E+02	1.32E+03	1.59E+03	1.66E+03
						Dose unit: rem

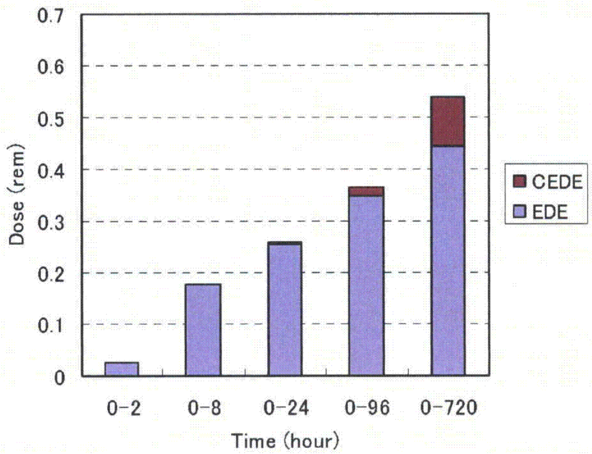
Table 5.3
TEDE at 50 m Downwind Distance (rem)

Cover gas leak rate: 0.1%/day						
	Containment Vessel Leak Rate	**2h**	**8h**	**24h**	**96h**	**720h**
Case 1	1%/day	2.84E-03	2.05E-02	3.09E-02	4.61E-02	9.12E-02
Case 2	10%/day	2.83E-02	2.03E-01	3.04E-01	4.40E-01	6.87E-01
Case 3	100%/day	2.77E-01	1.87E+00	2.64E+00	3.23E+00	3.60E+00
						Dose unit: rem
Cover gas leak rate: 10%/day						
	Containment Vessel Leak Rate	**2h**	**8h**	**24h**	**96h**	**720h**
Case 4	1%/day	2.83E-01	2.03E+00	3.04E+00	4.38E+00	6.65E+00
Case 5	100%/day	2.76E+01	1.85E+02	2.60E+02	3.10E+02	3.24E+02
						Dose unit: rem

Table 5.4
TEDE at 200 m Downwind Distance (rem)

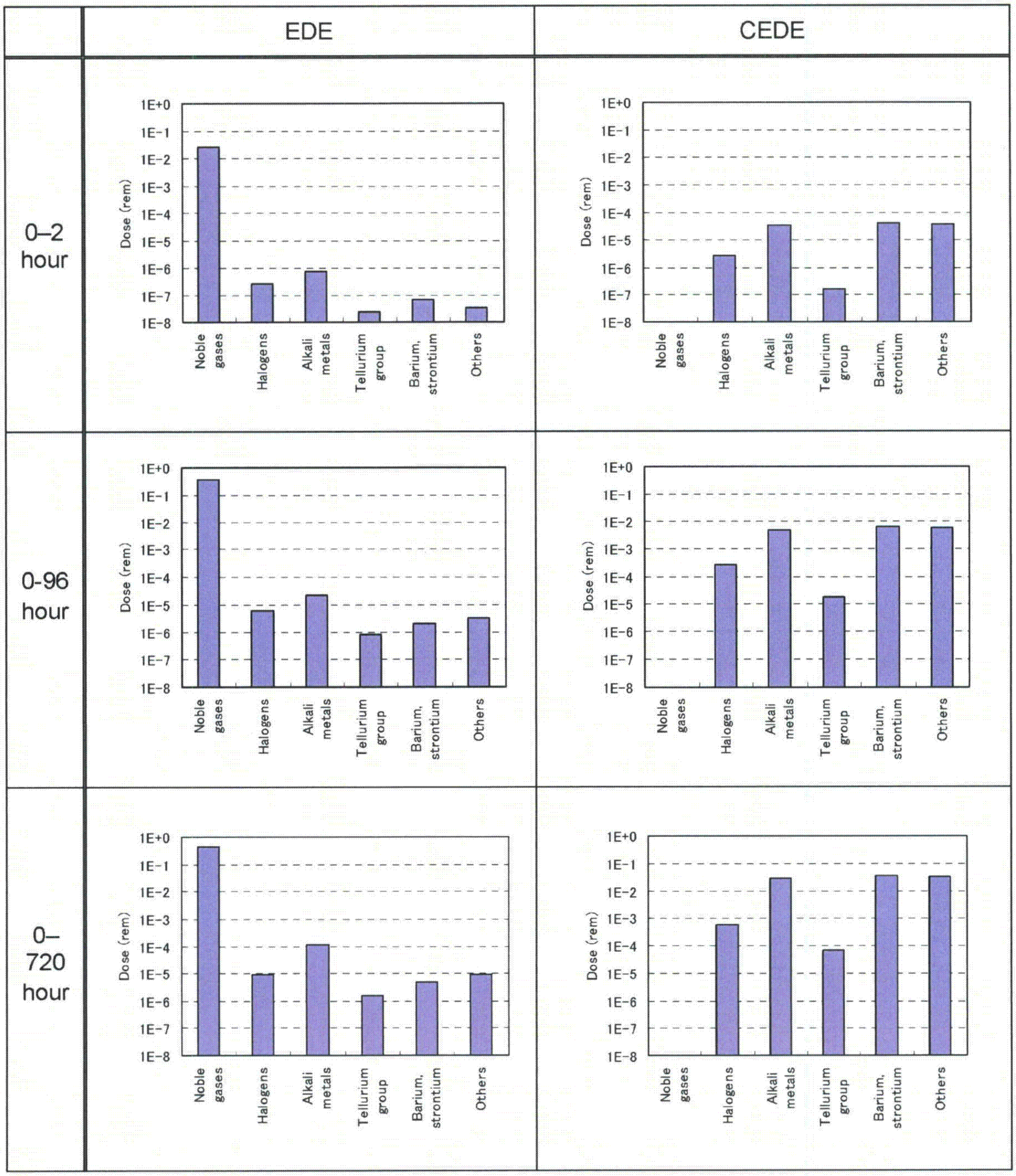
Cover gas leak rate: 0.1%/day						
	Containment Vessel Leak Rate	**2h**	**8h**	**24h**	**96h**	**720h**
Case 1	1%/day	2.49E-04	1.79E-03	2.62E-03	3.82E-03	7.33E-03
Case 2	10%/day	2.48E-03	1.78E-02	2.58E-02	3.65E-02	5.58E-02
Case 3	100%/day	2.43E-02	1.64E-01	2.25E-01	2.71E-01	3.00E-01
						Dose unit: rem

Cover gas leak rate: 10%/day						
	Containment Vessel Leak Rate	**2h**	**8h**	**24h**	**96h**	**720h**
Case 4	1%/day	2.48E-02	1.78E-01	2.58E-01	3.64E-01	5.40E-01
Case 5	100%/day	2.42E+00	1.62E+01	2.21E+01	2.61E+01	2.72E+01
						Dose unit: rem



Leak rate
Cover gas: 10%/day
Containment vessel: 1%/day
Downwind distance: 200 m

Figure 5.3 Breakdown of EDE and CEDE



Leak rate

Cover gas: 10%/day

Containment vessel: 1%/day

Downwind distance: 200 m

Figure 5.4 Breakdown of Dose by Nuclide Group

6 4S EPZ IDENTIFICATION

According to the guidance regarding protective actions for nuclear incidents, under normal conditions, evacuation of the general population should be initiated for most incidents at a projected dose of 1 rem, although the Protective Action Guide (PAG) is expressed as a range of 1–5 rem [16]. Therefore, 1 rem would be an appropriate allowable dose limit to evaluate the size of the EPZ.

Table 6.1 shows the criteria for leak rate at different distances to satisfy the allowable dose limit, set to be less than 1 rem. It is shown that an EPZ radius of 200 m could be satisfied, whose TEDE is calculated to be less than allowable dose limit, 1 rem, if the leak rates of the cover gas volume and containment vessel are successfully kept under 10%/day and 1%/day, respectively. The 4S is designed to keep the cover gas leak rate less than 10%/day, and the containment leak rate less than 1%/day for a severe accident.

Table 6.1
Criteria of Leak Rate for EPZ Definition

Downwind Distance	Limitation for Leak Rate		TEDE in 720h (rem)
	Cover Gas to Containment Vessel (Top Dome)	Containment Vessel (Top Dome)	
20 m	0.1%/day	1%/day	0.482
50 m	0.1%/day	10%/day	0.687
200 m	0.1%/day	100%/day	0.3
	10%/day	1%/day	0.54

7 CONCLUSIONS

The current basis for EPZ sizing was reviewed and the importance of a modernized EPZ for SMRs and advanced reactors was discussed in this report. Criteria for an appropriately sized EPZ were suggested based on mitigation of a Class 9 event, which requires core melt, due to 4S's inherent safety aspects.

The radiological consequences based on the Class 9 event without core melt (mod-Class 9) were evaluated. As shown in Table 7.1, an EPZ radius of 200 m could be satisfied, whose TEDE is calculated to be less than the allowable dose limit of 1 rem, if the leak rates of the cover gas volume and containment vessel (top dome) are successfully kept below 1%/day and 10%/day, respectively. The 4S is designed to keep the cover gas leak rate less than 10 %/day and the containment leak rate less than 1 %/day for a severe accident.

This suggests that an EPZ at 200 m could be technically justified.

Table 7.1
Design Leak Rate and Calculated Dose at 200 m

Cover Gas to Containment Vessel (Top Dome)	Containment Vessel (Top Dome)	TEDE at 200 m (rem)
10 %/day	1 %/day	2.48E-2 (2h) 2.58E-1 (24h) 5.40E-1 (720h)

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