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Subject: Submittal of Technical Report "4S Safety Analysis"

Enclosed is a copy of the non-proprietary "4S Safety Analysis" 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 "4S Safety Analysis"

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4S Safety Analysis

July 2009

TOSHIBA CORPORATION

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LIST OF ACRONYMS AND ABBREVIATIONS

AC	air cooler of the intermediate reactor auxiliary cooling system
AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
BOP	balance of plant
CDF	cumulative damage fraction
CEDE	committed effective dose equivalent
CPT	component performance test
CRBR	Clinch River Breeder Reactor
DBA	design basis accident
DBE	design basis event
EAB	exclusion area boundary
EBR-II	Experimental Breeder Reactor-II
EDE	effective dose equivalent
EM	electromagnetic
EMDAP	Evaluation Model Development and Assessment Process
EOL	end of life
EPA	Environmental Protection Agency
FCA	fast critical assembly
FCC	failure of a cavity can
FCTT	Fuel Clad Transient Tester
FFTF	Fast Flux Test Facility
FMEA	failure modes and effects analysis
FOM	figure of merit
FP	fission product
GV	guard vessel
IC	inner core
IET	integral effects test
IHTS	intermediate heat transport system
IHX	intermediate heat exchanger
IRACS	intermediate reactor auxiliary cooling system
JAERI	Japan Atomic Energy Research Institute
LMP	Larson-Miller parameter
LMR	liquid metal reactor
LOCA	loss-of-coolant accident
LOF	loss of flow
LOHS	loss of heat sink
LOSP	loss of offsite power
LPZ	low-population zone
LWR	light water reactor
MC	middle core
MG	motor-generator (set) used for pump coastdown
MLD	master logic diagram

LIST OF ACRONYMS AND ABBREVIATIONS (cont.)

MWe	megawatts electrical
MWt	megawatts thermal
NRC	United States Nuclear Regulatory Commission
OC	outer core
PA	postulated accident
PHTS	primary heat transfer system
PIE	postulated initiating event
PIRT	Phenomena Identification and Ranking Table
PRISM	Power Reactor Innovative Small Module
RBCB	run beyond cladding breach
RG	Regulatory Guide
RHR	residual heat removal
RI	radioisotope
RIR	reactivity insertion by uncontrolled motion of segments of reflector at startup
RPS	reactor protection system
RVACS	reactor vessel auxiliary cooling system
SET	separate effects test
SLIP	sodium leak from intermediate piping
SRP	Standard Review Plan (NUREG-0800)
SSC	structures, systems, and components
TEDE	total effective dose equivalent
TOP	transient overpower
ULOF	unprotected loss of flow

EXECUTIVE SUMMARY

This report is the fourth in a series of reports that Toshiba Corporation is submitting to the U.S. Nuclear Regulatory Commission (NRC) as a part of the pre-application review of the 4S (Super-Safe, Small and Simple) reactor.[1] The purpose of this report is twofold:

- Inform the NRC of the 4S safety analysis approach and insights from its application to eight events that were selected to represent *anticipated operational occurrences* (AOOs), *design basis accidents* (DBAs), and *anticipated transients without scram* (ATWS).
- Obtain NRC feedback on the safety analysis approach and the way it has been applied to the 4S liquid metal reactor (LMR).

The 4S safety analysis reported in this submittal was done during the preliminary design phase of the 4S reactor that has been recently completed. The process for optimizing the design to meet its goals of super safety performance, security, and simplicity and the innovative features that resulted from this process were presented to the NRC in the first pre-application meeting [1]. Of particular interest to this report is the thorough search that was performed in that process to identify events that could challenge the fission product barriers and the response of safety systems and structures to these events. The techniques used for that search include failure modes and effects analysis (FMEA) and master logic diagram (MLD) which are discussed in the body of this report. Those techniques were complemented by a preliminary probabilistic risk assessment (PRA) and evaluations of the applicability of the lessons learned from past LMR and light water reactor (LWR) experience to the 4S design. The document summarizing the results of the PRA will be submitted separately as part of the design approval application.

Predictably, the above search resulted in a very large number of events. This required reducing the events to a manageable number that covers the full spectrum of challenges to the 4S reactor safety. The reduction was accomplished by applying the guidance of SRP Section 15.0 [11] to: 1) categorize the events by expected frequency and type; and 2) select a bounding set of events to evaluate the reactor response to them. The eight events selected for safety analysis in this report have been picked from the above set of bounding events.

The 4S safety analysis uses metrics that have been defined to ensure the structural integrity of the 4S fission product barriers (fuel cladding, core coolable geometry, primary coolant boundary, and containment) and to protect the public health and safety from undue risk of radioactive release. Table ES-1 shows the selected metrics and the basis for their selection.

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1. The following 4S reports have already been submitted to the NRC: (1) "Long-Life Metallic Fuel for the Super Safe, Small and Simple (4S) Reactor," June 2008 [12]; (2) "4S Design Description," May 2008 [13]; and (3) "4S Seismic Base Isolation Design Description," ADAMS: ML090650235, February 2009.

Table ES-1 Safety Metrics and their bases

Safety Issue	Metric	Basis
• Fuel Integrity • Core coolable geometry	Clad cumulative damage fraction (CDF), No fuel melting	EBR-II safety case experience and irradiation tests at ANL and HEDL.
• Primary coolant boundary Integrity	Stress and temperature judged by ASME code service levels (Selected level commensurate with the expected frequency of the event category)	Accepted practice in LWR and LMR
• Containment Integrity	Containment stress and temperature judged by ASME code service levels	ASME code section III subsection NE Class MC component
• Radioactive Release	EAB and LPZ dose	10CFR 50.34(1)(ii)(D)

The report develops surrogate criteria of the metrics for calculation purposes. It also identifies the metric that has the most limiting criteria.

The methodology used to evaluate the 4S design against the above criteria includes two primary computational tools:

1. Toshiba's ARGO computer code - The code simulates the plant response to the selected events.

2. A set of mathematical expressions and correlations for use in estimating the radiological release fraction from the 4S metallic fuel-sodium coolant system, and the resulting offsite doses.

The reported safety analysis accounts for uncertainties in: 1) the response of safety systems, structures, and components to the postulated events, 2) physical phenomena relied upon to passively shut down the reactor and remove the heat from the core and reactor vessel, 3) input data and assumptions used in the analysis, and 4) reactor response simulation models, correlations, and analytical procedures. For design basis events (AOOs and DBAs) and radiological release, the above uncertainties were accommodated by use of conservative assumptions and models. For ATWS events, which are beyond the design basis, the analysis estimated the expected values of parameters related to the safety criteria as well as statistical upper bound values (95% probability at 95% confidence level).

Results of the above analysis show significant margins to specified 4S safety criteria. Although the analysis is a result of preliminary safety analysis effort for the preliminary design, it supports the viability of the 4S design margins, features that enhance safety, and the set-points used in the analysis for safety systems. Feedback from the NRC on this report is highly needed to proceed with the preparation of the 4S Design Approval application.

1 INTRODUCTION AND SAFETY ANALYSIS SUMMARY

1.1 Purpose

The contents of this report present a sample of the contents of Chapter 15 of the Safety Analysis Report (SAR) for design approval of the 4S reactor. The purpose of this report is twofold:

- Inform the NRC of the 4S project approach to satisfy regulatory requirements for safety analysis to be included in the 4S reactor SAR Chapter 15, in particular:
  - Completeness of the contents of Chapter 15
  - Proper selection and classification of the design basis events (DBEs) and beyond DBEs
  - Acceptability of the analysis methods, computer codes, data, and assumptions
  - Adequacy of operational experience, test data, and treatment of uncertainties for safety assurance of the 4S reactor
- Obtain NRC feedback on the items above to identify and resolve any outstanding issues and facilitate the 4S design approval process.

1.2 Scope

The scope of this technical report includes the following topics:

- Methodology and procedures used for safety analysis of the 4S reactor
- Results of the safety analysis of a selected set of events that illustrate how the 4S reactor meets its safety criteria

These topics were discussed during the NRC pre-application review meetings for the 4S [1–4]. This technical report provides more information that supplements the presentations at the third NRC pre-application review meeting [3].

In developing the 4S safety analysis, the guidance of the Standard Review Plan (NUREG-0800) [5] and Regulatory Guide (RG) 1.206 [6] has been used if applicable to a sodium-cooled fast reactor. Guidance not applicable to sodium cooled fast reactors was modified or replaced by new guidance as necessary.

1.3 Identification of 4S-specific Transients and Accidents

The first step in the 4S safety analysis approach is to conduct a comprehensive search for possible transients and accidents that can lead to radioactive release. The search covers the lifetime of the plant, all radioactive sources on site, and all operations related to these sources. This report focuses only on the radioactive sources within the reactor vessel and only on radioactive releases following the occurrence of a transient or an accident.

The 4S approach to identify a comprehensive set of transients and accidents includes methods and data that have been used in probabilistic risk assessments (PRAs) of first-of-a-kind reactors to ensure its completeness, namely:

- Failure modes and effects analysis (FMEA): This is a bottom-up tabular format that starts from failure modes at the component level and evaluates their impact on interfacing structures, systems, and components (SSCs) [7].
- Operating LMR transient and accident experience, as discussed in [8].
- Clinch River Breeder Reactor (CRBR) and Power Reactor Innovative Small Module (PRISM) transients and accidents used in safety analyses submitted to the NRC [9, 10].
- 4S design reviews.
- Master logic diagram (MLD) [7]: This is a top-down fault tree structure with the top event defined as radiological release from the containment. The bottom level of the MLD shows combinations of equipment failures and malfunctions that are necessary and sufficient to lead to the top event.

1.4 Initiating Events Categorization

The second step in the 4S safety analysis approach involves grouping the initiating events identified above into three frequency categories similar to those of SRP Section 15.0 [11], namely:

- *Anticipated operational occurrences* (AOOs): This category includes transients and accidents of moderate frequency (expected to occur one or more times during the life of the plant). It also includes infrequent initiating events (may, though not expected to, occur during the lifetime of the plant).
- *Design basis accidents* (DBAs): These are postulated accidents that:
  - — Are not expected to occur during the lifetime of the plant
  - — Present more severe challenges to plant safety than any transient or accident in the AOO category

— Are used as design basis for evaluating the effectiveness of safety-related SSCs

- *Anticipated transients without scram* (ATWS): Similar to SRP Section 15.0, the 4S reactor beyond DBE category includes ATWS events. It is to be noted, however, that the 4S reactor mitigates the impact of ATWS through passive reactivity feedback mechanisms and passive heat removal that protect the fuel integrity and maintain long-term core coolability.

Within each of the above frequency categories, transients and accidents are grouped by type. The types of interest refer to local or core-wide transients or accidents that result in heat generation/heat removal mismatches or failure of a radioactive-material barrier. Examples include local or core-wide loss of flow (LOF), transient overpower (TOP), or loss of heat sink (LOHS). The types also include complements of these examples (increase in core flow, transient power reduction, or excessive heat removal). The type categorization used in the 4S reactor safety analysis approach is similar to that of Section 15.0 of the SRP.

1.5 Selection of Bounding Initiating Events for Safety Analysis

The 4S safety analysis approach for selecting bounding initiating events for safety analysis is similar to that of SRP Section 15.0. Namely, within each frequency category, initiating events of the same type are compared and the one that presents more severe challenges to the safety of the reactor is selected for safety analysis. This helps reduce the number of analysis cases to a more manageable value.

1.6 Definition of 4S-specific Safety Analysis Acceptance Criteria

Safety analysis acceptance criteria are composed of two parts:

- Safety criteria
- Safety analysis requirements

The 4S safety criteria are defined in terms of six safety performance parameters for the spectrum of identified DBEs and beyond DBEs. As seen in Table 1-1, the following safety performance parameters are used:

- Absence of fuel melting

- Maintaining the integrity of the three radioactive release barriers:

  — Fuel cladding
  — Primary coolant boundary
  — Containment

- Maintaining a coolable core geometry
- Not violating the regulatory dose limits at the exclusion area boundary (EAB) and low-population zone (LPZ) for the site assumed for the 4S

The safety criteria for AOOs include no fuel melting, maintaining cladding integrity and core coolability, and meeting ASME Service Level B or C for the primary coolant boundary. These criteria ensure quick recovery from an AOO, with minimum impact on the fuel properties or design life.

For AOO, DBA, and ATWS events, the 4S analysis uses a fuel cladding integrity criterion based on the cumulative damage fraction (CDF < 0.1). By maintaining the CDF below 0.1, the failure probability of a fuel pin is less than 1/3000 at the 95 percent confidence level, based on the tests to failure of HT-9 clad that were conducted in the Fuel Clad Transient Tester (FCTT) [20,21].

The 4S reactor core has 18 assemblies, with 169 fuel pins in each, for a total of 3042 pins. This implies that no more than one fuel pin failure is to be expected during the 4S reactor life. Meeting the CDF criterion also ensures coolable geometry of fuel pins and fuel bundles for DBEs.

The 4S safety analysis requirements are discussed in the following section.

1.7 Safety Analysis Assumptions

The 4S reactor has the unique feature of no required refueling during the 30-year life of the plant. This report shows that the highest radiological consequences are to be expected if a transient or accident occurs at the end of life (EOL) of the plant. Based on this finding, the safety analysis in this report assumes that the transient or accident occurs at EOL.

As shown in Table 1-1, analysis of DBEs is required to be conservative, while analysis of beyond DBEs is required to provide best estimates as well as uncertainty evaluations. The assumptions used for these analyses and for the radiological consequences are described below.

Design Basis Analysis Assumptions

- Initial power at 102 percent
- Hottest pin calculated under conservative conditions

- Single failure of active components in safety systems

  — Scram: one stuck reflector segment

  — Residual heat removal (RHR): intermediate reactor auxiliary cooling system (IRACS) – lose one of two dampers functioning

- Reactivity feedback

  — Combination of reactivity coefficients assumes the worst of each coefficient

- RHR

  — No coastdown for intermediate loop, i.e., the intermediate flow stops instantaneously
  — Steam generator (SG) instantaneously isolated at time of accident
  — Natural circulation in primary, intermediate, and IRACS

Beyond-Design-Basis (ATWS) Analysis Assumptions

- 100 percent power
- Nominal reactivity coefficients
- Hottest pin calculated under nominal conditions

Radiological Consequence Analysis Assumptions

The EAB and LPZ doses are estimated using a nonmechanistic, conservative source terms:

- Fraction of fuel damaged:

  - 1 % (~ 30 fuel pins which is conservative for the 4S reactor DBEs as discussed earlier)

- Radioactivity inventory:

  - 150 % of the estimated inventory at the end of life, which is very conservative but used to cover uncertainties in estimating the fission products and activated primary sodium inventory during the preliminary design phase.

- Release into primary sodium

  — No fuel retention is assumed for the following fission product (FP) groups (see Section 4.1 for the chemical elements in these groups)

- — Noble gases
- — Halogens (no fuel retention although retention in the metallic fuel as UI_3 or NaI is likely)
- — Alkali metals
- — Te group
- — Sr and Ba
- — 0.1 % release fraction is used for noble metals, Ce group and lanthanides

- Release into cover gas region
  - High primary sodium temperature (650°C) is used for estimating the release fraction. This temperature bounds the sodium temperature estimates for all the cases analyzed. Appendix A contains the correlations used to estimate the release fractions.
- Leak rates from cover gas region and containment:
  - The analysis uses design leak rates which are demonstrable under the design pressure and temperature limits of these boundaries. None of these limits was reached under the most severe conditions of the analyzed events.
- Radiation dose:
  - Please see Appendix A for the regulatory basis of the site weather conditions and dose calculation.

1.8 Safety Analysis Methods and Data

Simulation of plant response to the initiating events

- ARGO code: Used for simulating the plant response and estimating the core damage fraction for DBEs and beyond DBEs. Appendix A describes the code architecture and methods used as well as the verification and validation of the code.

Statistical analysis – performed for ATWS analysis

- Parameters for statistical analysis: Determined as follows:

  — Five experts selected and ranked plausible phenomena.

  — Experts selected 17 parameters from 88 plausible phenomena identified in the phenomena identification and ranking table (PIRT) process.

  — The 17 parameters were further reduced to 10 parameters after sensitivity analysis.

Radiation dose

  — Non-mechanistic source term

    - Release fractions, primary sodium temperature and leak rates as presented earlier.

    - Release from primary sodium to the cover gas region – Please see Appendix A for the correlations used.

  — Weather conditions – Please see Appendix A for the regulatory basis used for EAB and LPZ dose estimation.

  — Dose estimation – Please see Appendix A for the procedure and data used for estimating the EAB and LPZ doses.

1.9 Analysis Results

Tables 1-2, 1-3, and 1-4 present the peak fuel and clad temperatures and the CDF for AOOs, DBAs, and ATWS. These results show sufficient margin to the 4S acceptance criteria for the fuel and clad. Consequently, the 4S design also meets the acceptance criteria for coolant boundary integrity with sufficient margin.

1.10 Conclusions

The 4S safety analysis approach is consistent with the SRP approach for light water reactors (LWRs). The specific initiating events, safety criteria, and plant responses to initiating events are different between LWRs and the 4S, as should be expected. The most important differences are summarized below:

- 4S features that prevent core damage
  - — Reactor vessel at atmospheric pressure (no loss-of-coolant accident (LOCA))
  - — Coolant thermal capacity
  - — Metallic fuel high conductivity
  - — Fuel/coolant compatibility
  - — Two passive heat removal systems
  - — Negative reactivity feedback
- 4S features that reduce radiological consequences
  - — Small inventory due to low power
    - – Short-lived FP inventory: A reduction factor of 100 relative to 3000 MW LWR
    - – Long-lived FPs proportional to power multiplied by irradiation time: A reduction factor of about 10 relative to 3000 MW LWR
  - — Retention of FPs in the core and sodium
  - — Design margin against clad failure for DBEs and beyond DBEs
  - — No sodium boiling; core remains covered by ~ 11 m of sodium
  - — No accident energetics
- Release reduction
  - — Sealed reactor vessel and containment
  - — Minimal number of penetrations and isolation valves
  - — Minimal threat to containment

Although the analysis in this report did not take credit for all the above features, the analysis confirms adequacy of the assumed plant protection system setpoints, response time, and safety systems performance. Feedback from the NRC on the analysis and these conclusions will be of great help for detailed design of the 4S to proceed.

Table 1-1
Safety Criteria and Analysis Requirements

Event \ Region		No Fuel Melting	Fuel Cladding Integrity	Core Coolable Geometry	Primary Coolant Boundary Integrity	Containment Integrity	Radiation Dose at EAB and LPZ	Analysis Requirements
DBE	AOO	✓	✓	*	✓ ASME SL† "B," "C"	*	*	Conservative
	DBA	-	✓	✓*	✓ ASME SL "D"	✓	*	Conservative
Beyond DBE	ATWS	-	✓	✓*	✓ ASME SL "D"	✓	✓*	Best estimate plus uncertainties

Notes:
✓ Explicit safety criterion is defined.
* Meeting the safety criterion is expected if the previous criterion is met.
† SL = Service Level

Table 1-2
Analysis Results of AOO

Event	Fuel Peak Temp (°C)	Clad Peak Temp (°C)	CDF
Loss of offsite power	639	613	6E-9
Decrease of primary coolant flow	644	621	2E-8
Reactivity insertion by uncontrolled motion of segment of reflector at startup	613	607	7E-8
Inner or Outer tube failure of the SG	Similar to loss of offsite power		

Table 1-3
Analysis Results of DBA

Event	Fuel Peak Temp (°C)	Clad Peak Temp (°C)	CDF
Failure of a cavity can	664	629	3E-8
Sodium leakage from intermediate piping	639	613	3E-8
Primary cover gas boundary failure	Normal reactor shutdown Dose at LPZ ≤ 0.01 rem for assumed 1% fuel failure		

Table 1-4
Analysis Results of ATWS

Event	Case	Analysis Condition	Clad Peak Temp (°C)	CDF
Loss of offsite power without scram	Reference	Best estimate	Hottest pin: 743	Hottest pin: 4.4E-4
	Statistical	Best estimate plus uncertainty	Average value: 745 95/95% probability/confidence level value: 798	Average value: 4.6E-4 95/95% probability/confidence level value: 1.2E-2

2 OVERVIEW OF 4S SAFETY SYSTEMS

Safety systems used in this safety analysis include: (1) reactor protection system (RPS), (2) reactor shutdown systems, (3) RHR systems, and (4) containment system. These systems are described in Reference [13].

2.1 Reactor Protection System

2.1.1 System Identification

Figure 2-1 includes an overview of the RPS. The RPS consists of sensors, logic circuits, trip breakers, and actuators, configured as three separate identical sensors and logic channels each for both the main and backup systems.

2.1.2 Sensors and Logic Circuit of Reactor Protection System

Figure 2-2 shows sensors of RPS. The RPS has three signal sensing channels (Channels I, II, and III), which are physically and electrically separated from each other. The various process signals, which are grouped into the three channels, are gathered into the separate signal conditioning panels designated for each channel. When a sensed signal exceeds its setpoint, scram signals from bistable circuit outputs are transmitted to the RPS voter logic panel. In the RPS voter logic panel, those scram signals are input to the two-out-of-three voter logic circuit so that if two or more signals come into this voter, reactor trip initiating signals (divided into three logic train signals – A, B, and C) are sent to the reactor trip breakers of each shutdown system. The reactor trip initiating signals are also transmitted to the primary and intermediate-loop sodium coolant circulation electromagnetic (EM) pump drive systems, the intermediate reactor auxiliary cooling system (IRACS) air cooler damper drive circuits, and the feedwater pump drive system. After the trip initiation, the reactor will shut down, the primary and intermediate-loop EM pumps and feedwater pumps will trip, and the IRACS air cooler dampers will open.

The RPS is actuated when a threshold is attained by any one of the following safety function parameters:

- Reactor core neutron flux
- Liquid sodium level in the reactor vessel
- Supply voltage/current of primary loop EM pumps
- Primary outlet temperature of the intermediate heat exchanger (IHX)
- Voltage of power line
- Seismic acceleration

2.2 Reactor Shutdown Systems

Figure 2-3 shows the reactor shutdown system. The reactor shutdown system consists of the following reactivity control elements:

- Reflector
- Shutdown rod
- Drive equipment, including drive units for each segment of reflector and the reactor shutdown rod

Six separately controllable segments combine to form a cylindrical reflector outside the core barrel. The subassembly at the center of the reactor core 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 with the shutdown rod available as a backup shutdown system.

2.3 Residual Heat Removal Systems

Figure 2-4 shows a schematic diagram of the RHR systems. The IRACS removes decay heat by using an air cooler in the intermediate heat transport system (IHTS). The reactor vessel auxiliary cooling system (RVACS) removes the decay heat with natural convection of air outside the reactor guard vessel. The RVACS serves as a heat collector between the cylindrical concrete wall around the guard vessel (below ground level) and the reactor vessel. Ambient cold air descends between the cylinder wall and the heat collector, turns upward at the lower end of the heat collector cylinder, then rises between the heat collector and guard vessel. Radiation heat from the reactor vessel is removed with natural convection heat transfer in the gap between the guard vessel and the heat collector. The process takes place under all plant conditions and for all design events entirely without the intervention of any active equipment. The exhaust duct is elevated approximately 13 m from the reactor core. Because the estimated RVACS exhaust air temperature may exceed the allowable concrete temperature, the structure is insulated to meet the temperature limit of the building concrete.

2.4 Containment System

The containment system consists of the guard vessel, top dome, airlock, cable penetrations, and piping penetrations. The containment system is shown in Figure 2-5. The guard vessel surrounds the reactor vessel. If the reactor vessel should leak below the sodium level, the guard vessel would retain the sodium, the core would still be immersed, and the sodium flow circuit would be maintained. During reactor operation, the annulus between the reactor vessel and guard vessel and the top dome is filled with nitrogen gas. Containment monitoring is provided by a radiation monitor in the top dome. Temperature monitors, pressure gages, and sodium leak detectors are also provided in containment to detect failures of the primary heat transfer system (PHTS) or IHTS.

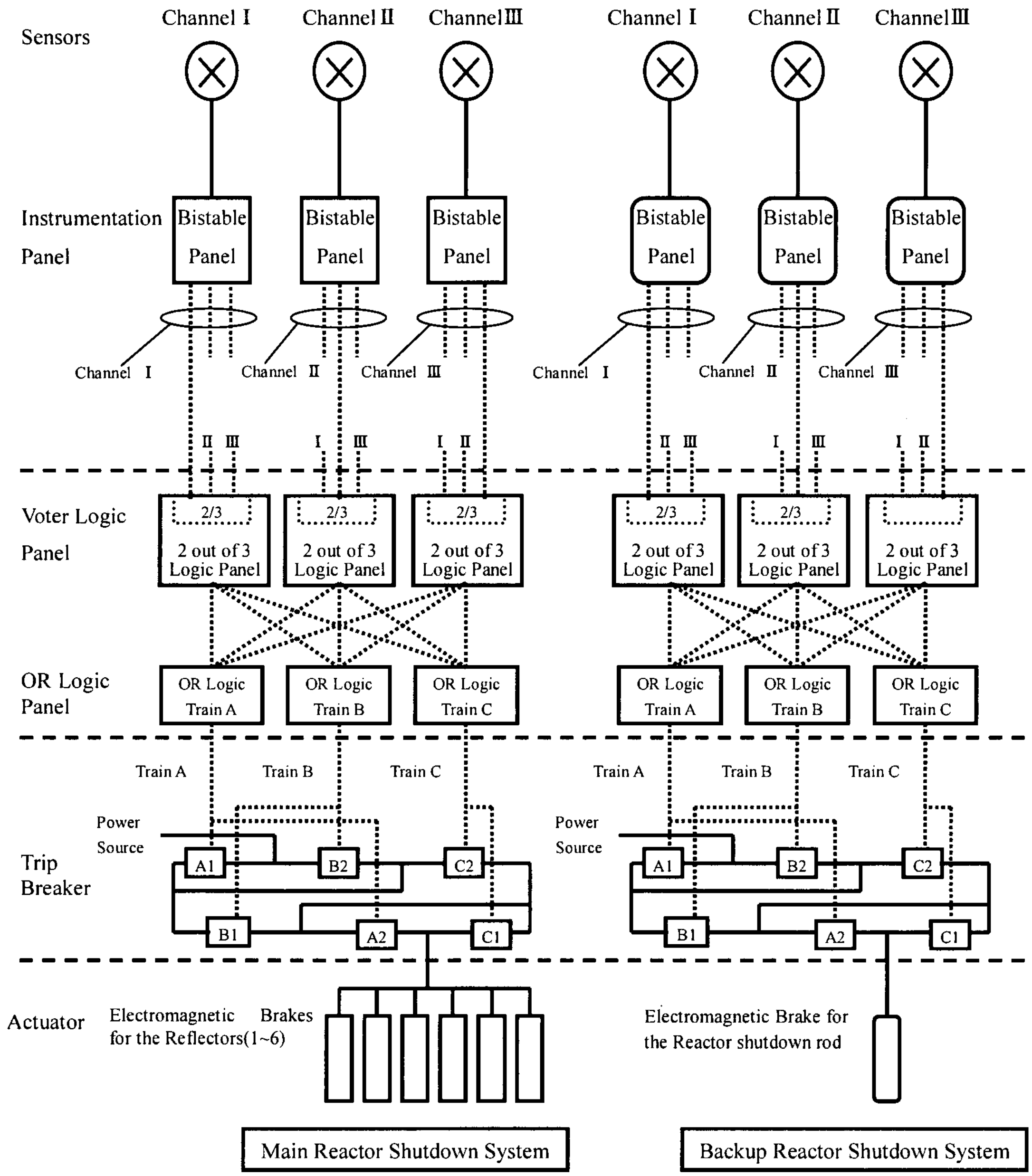


Figure 2-1 Reactor Protection System Logic Circuit

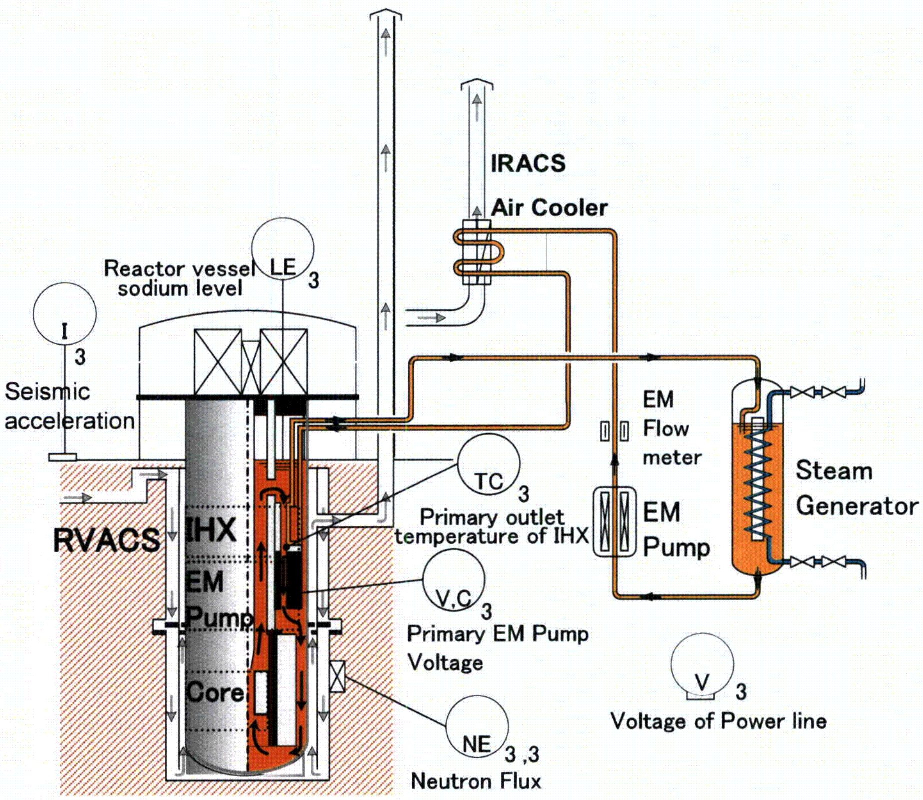


Figure 2-2 Sensors of Reactor Protection System

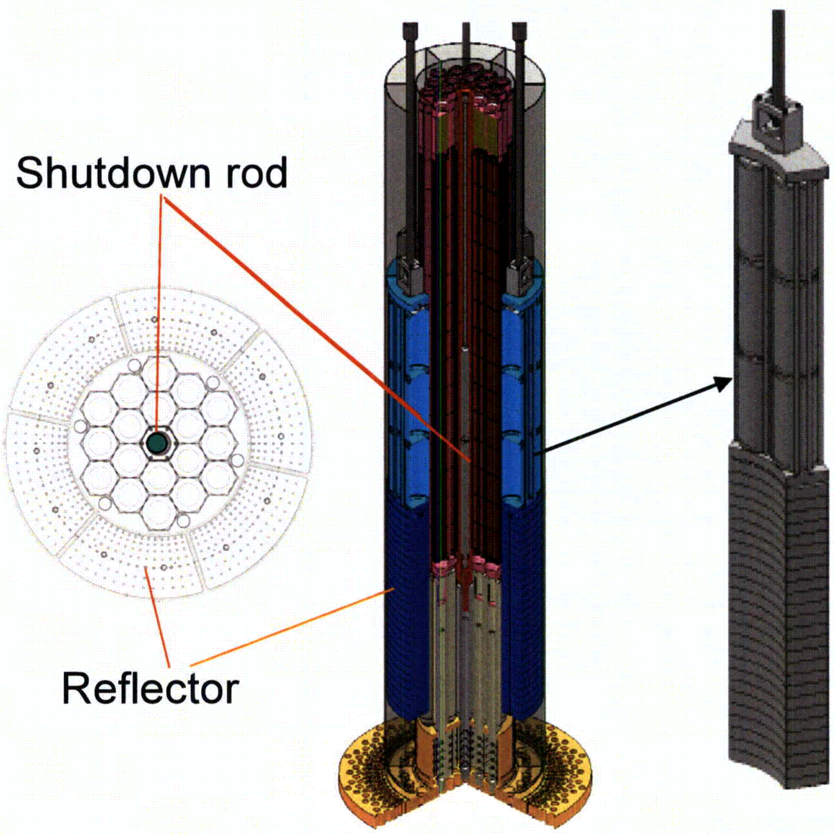


Figure 2-3 Two Reactor Shutdown Systems

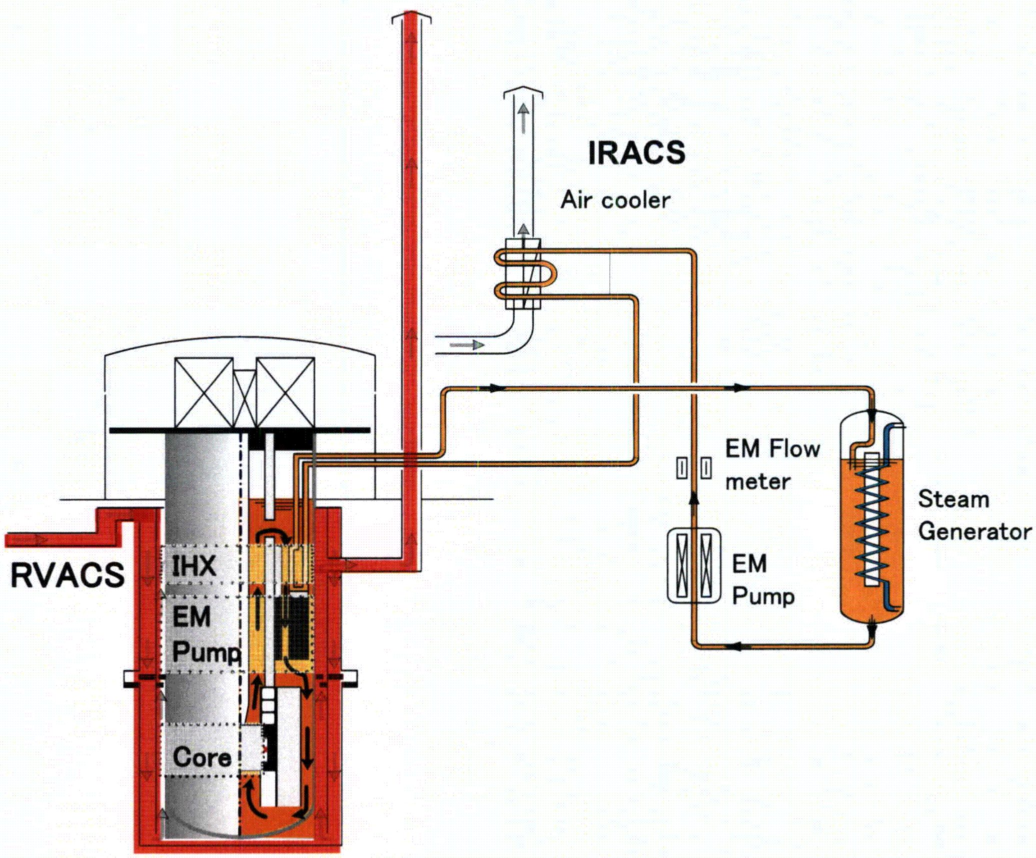


Figure 2-4 Schematic Diagram of Residual Heat Removal Systems

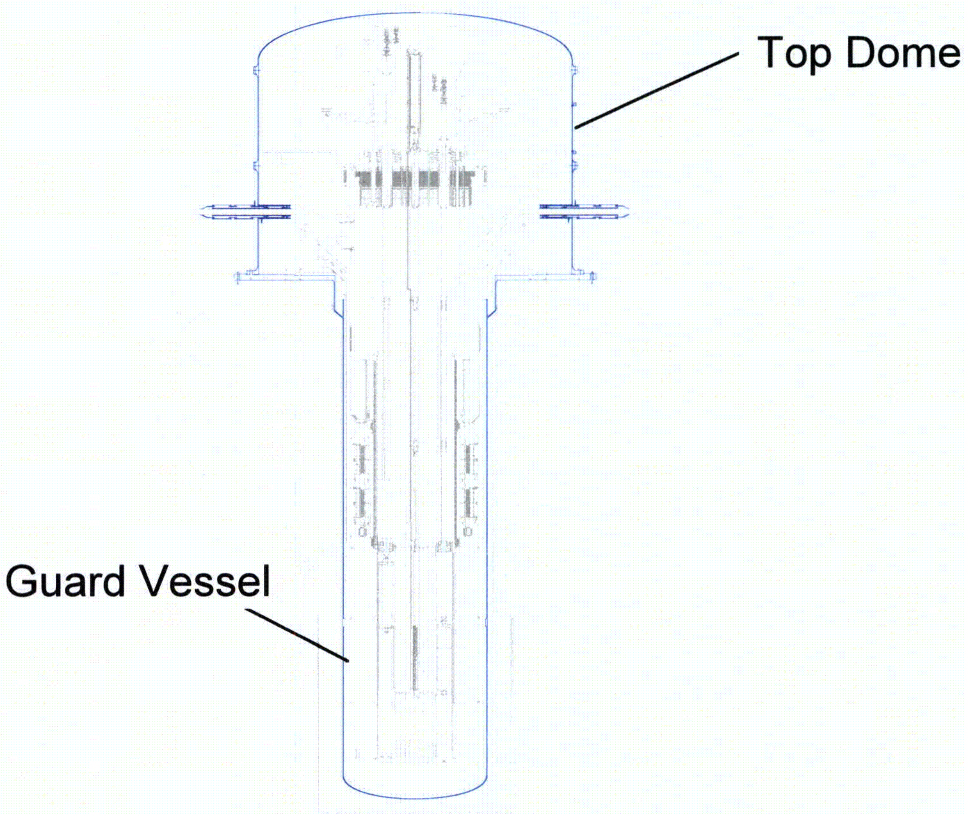


Figure 2-5 Containment Boundary

3 ANALYSIS APPROACH

This section presents two basic elements of the 4S safety analysis approach:

- The approach used for the selection of an appropriate set of events for the 4S safety analysis and

- The acceptance criteria for plant response to these events.

3.1 Overview of Analysis Procedure

Chapter 15.0 of NUREG-0800 [11] and Chapter C.I.15 of RG 1.206 [6], in which the concepts for nuclear power plant safety analysis are provided, have developed for a light water reactor (LWR). For the 4S reactor safety analysis, the events to be evaluated and the analysis acceptance criteria were determined by modifying and/or replacing the guidance from the regulatory documents in consideration of the different design features of a sodium-cooled fast reactor.

The following steps were performed:

1. The categories of events to be evaluated were established as anticipated operational occurrences (AOOs), design basis accidents (DBAs), and anticipated transients without scram (ATWS).

2. The events to be evaluated were selected for deterministic analysis; that is, based on the expected frequency of occurrence and event type.

3. The acceptance criteria for AOOs, DBAs, and ATWS are defined in Section 3.4.

3.2 Categorization of Transients and Accidents

DBEs include the AOOs and DBAs, and they are treated conservatively. ATWS events are beyond DBEs and are treated on the basis of best estimate.

3.2.1 Anticipated Operational Occurrences (AOOs)

An AOO is an event anticipated to occur once or more during the service life of the nuclear power unit.

3.2.2 Design Basis Accidents (DBAs)

DBAs are postulated accidents (PAs) 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 [11].

3.2.3 Anticipated Transients Without Scram (ATWS)

ATWS are AOOs in which a reactor scram is demanded but fails to occur because of a common-mode failure in the reactor scram system [11].

3.3 Selection of AOOs, DBAs, and ATWS

The events to be evaluated for the safety analysis were selected based on the following procedure:

1. The structures, systems, and components (SSCs) of the 4S reactor were all identified [13] (see Figure 3.3-1).

2. Failure modes and effects analyses (FMEA) were performed for each SSC described above. The function, failure mode, causes of failure, effect on availability, effect on safety, compensating provisions, detection methods, and occurrence frequency were extracted. The features of each event progression caused by an SSC failure were categorized for each type by focusing on the effect on safety.

3. The set of postulated initiating events (PIEs) for selecting AOO and PA candidates was categorized based on the results of the FMEA. The past failure rate data [14–16] and the evaluation results [9, 10] of events having occurred in conventional reactors were referenced to determine the failure frequency of a component element.

    a. The events that are anticipated to occur once or more during the life of a nuclear power unit are designated as AOOs.

    b. The events that are not anticipated to occur during the life of a nuclear power unit are designated as PAs.

4. For events of the same type, the system response, event progression, and accident effects were evaluated to determine the most severe event as a representative event, according to the analysis acceptance criteria of section 3.4.

5. Among the events selected as PA candidates, any event that prescribes the design criteria and limit for deciding the design and sizing of a safety system and component was selected as a DBA.

   Safety-related (safety) structures, systems and components means those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

   - *The integrity of the reactor coolant 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 which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.34(a)(1) or 10 CFR 100.11, of this chapter, as applicable (excerpted from 10 CFR 50.2).* [17]

6. ATWS Selection

   ATWS were selected from the set of AOOs by assuming a reactor scram is demanded but fails to occur because of a common-mode failure in the reactor scram system.

7. To confirm the results of the previous steps, a master logic diagram (MLD) [7] was developed with the top event prescribed as the effect of the event on the public, namely, the release (Level 1) of a radioactive material. Lower levels were expanded to extract events that are the result of a system failure.

   The events obtained from these MLD expansions were compared with the events categorized for each type by focusing on the effect on safety, extracted from the FMEA by the process described in all previous steps, to confirm the sufficiency of the event selection to be evaluated for safety analysis.

The events selected to be evaluated for the safety analysis by the preceding process are shown in Table 3.3-1.

The events selected for the PRISM [10] safety analysis are shown in Table 3.3-2 for comparison.

Two events selected for PRISM that were not selected for 4S and the related 4S design differences are as follows:

- Primary sodium cold trap leak – At startup of the 4S, the primary purification system is initiated once before operating the reactor. After this purification, since the reactor is operated with the reactor vessel sealed, the purification system is not used [13].

Therefore, the event of primary sodium cold trap leak is not applicable for the 4S operating plant lifetime.

- Fuel transfer cask cover gas release – After the initial fuel loading, there is no refueling for the life of the reactor. Therefore, the event of fuel transfer cask cover gas release is not applicable for the 4S operating plant lifetime.

Table 3.3-1
Events Selected for the Safety Analysis

Anticipated Operational Occurrences (AOOs)
Reactivity insertion by uncontrolled motion of segments of reflector at full-power operation
Reactivity insertion by uncontrolled motion of segments of reflector at startup
Decrease of primary coolant flow
Decrease of intermediate coolant flow
Increase of primary coolant flow
Increase of intermediate coolant flow
Inner or outer tube failure of the SG
Increase of feedwater flow
Decrease of feedwater flow
Loss of offsite power

Table 3.3-1
Events Selected for the Safety Analysis (cont.)

Design Basis Accidents (DBAs)
Failure of a cavity can
Reactor vessel leakage
One primary EM pump failure (instantaneous loss of power to one pump, equivalent to a mechanical pump seizure)
Sodium leakage from intermediate piping
Local fault in a fuel assembly
Primary cover gas boundary failure

Table 3.3-1 Events Selected for the Safety Analysis (cont.)
Anticipated Transients Without Scram (ATWS)
Unprotected loss of primary and/or intermediate coolant flow (ULOF)
• Decrease of primary coolant flow without scram
• Decrease of intermediate coolant flow without scram
• Loss of offsite power without scram
Unprotected transient overpower
• Reactivity insertion by uncontrollable motion of segments of reflector at full-power operation without scram
• Reactivity insertion by uncontrollable motion of segments of reflector at startup without scram

Table 3.3-2 Events Selected for the Safety Analysis of the PRISM
Reactivity insertion DBEs
• Uncontrolled rod withdrawal at 100 percent power
Undercooling DBEs
• Loss of normal shutdown cooling
Local fault tolerance
Sodium spills
• Primary sodium cold trap leak
Fuel handling and storage accidents
• Fuel transfer cask cover gas release
Other design basis events
• Cover gas release accident

- 4S Reactor
  - Reactor and core
    - Core
      - Fuel (Subassembly)
    - Reactivity control and shutdown system
      - Reflector
      - Reflector drive
      - Shutdown rod
      - Shutdown rod drive
      - Fixed absorber
  - Reactor coolant system and connected systems
    - Reactor vessel
    - Shielding plug
    - Guard vessel
    - Top dome
    - Reactor internal structures
      - Core support structures
      - Upper vertical shroud
      - Fixed shield
      - Instrumentation
    - Primary heat transport system
      - Intermediate heat exchanger
      - Primary EM pump
    - Intermediate heat transport system
      - Steam generator
      - Air cooler
      - Piping
      - Intermediate EM pump
      - Instrumentation
    - Residual heat removal system
      - IRACS
      - RVACS
  - Engineered safety features
    - Containment system
      - Guard vessel
      - Top dome
    - Containment isolation
      - Penetration (cable and piping)
      - Extension
  - Instrumentation and control
    - Reactor protection system
    - Safety related instrumentation
    - Instrumentation and monitoring systems of interfacing systems
  - Electric power
    - Offsite power system
    - Onsite power system
  - Auxiliary systems
    - Process auxiliaries
    - Heating, Ventilation, and Air conditioning system
    - Auxiliary liquid metal system
    - Sodium piping and equipment heating and insulation system
    - Plant fire protection system
  - Steam and power conversion system
    - Turbine generator
    - Main and auxiliary steam system
  - Radioactive and waste management
    - Gaseous waste management system
    - Process and effluent radiological monitoring and sampling system
  - Buildings

Figure 3.3-1 Structures, Systems, and Components

3.4 Acceptance Criteria

3.4.1 Definition of Acceptance Criteria

The acceptance criteria for the safety analysis of the 4S reactor are defined as follows, considering the guidance of SRP 15.0, and making allowance for differences between LMRs and LWRs [11].

1. Acceptance criteria for AOOs

   - Maintaining the fuel cladding integrity
   - Maintaining the primary coolant boundary integrity

2. Acceptance criteria for DBAs

   - Maintaining the core coolable geometry
   - Maintaining the primary coolant boundary integrity
   - Maintaining the allowable radiation exposure at an EAB

3. Acceptance criteria for ATWS

   - Maintaining the core coolable geometry
   - Maintaining the primary coolant boundary integrity
   - Maintaining containment integrity

The acceptance criteria and their derivation are described in Section 3.4.2.

3.4.2 Acceptance Criteria

3.4.2.1 Fuel Cladding Integrity and Core Coolable Geometry

This section is based on the design criteria developed for the Experimental Breeder Reactor-II (EBR-II) Mark-V metallic fuel pins, accounting for the 4S design. These criteria were developed in 1994 [18, 19] as part of the safety case for the introduction of that fuel into EBR-II' based on experience from the PRISM and Clinch River Breeder Reactor (CRBR) projects They also reflect experience with the behavior of metallic fuel clad in HT9 steel gained from irradiations in EBR-II and the Fast Flux Test Facility (FFTF). The evaluation criteria establish the detailed criteria that the fuel pin design must satisfy to meet the principal design requirements. These criteria were selected so that the fuel pins will satisfy their functional requirements and performance objectives in a safe and reliable manner based on current technology. The functional requirements of the fuel design are discussed first, followed by the design criteria that are put in place to ensure that the fuel pins meet those functional requirements.

1. Functional Requirements

The primary functions of the fuel elements (often referred to as "fuel pins") in a reactor are to position and retain the nuclear fuel in an array such that a critical mass can be achieved, control rods can move within the array to control the reactivity, and coolant can pass through to remove the heat generated by fission in the fuel. The fuel pin cladding acts as the primary barrier against fission product release into the coolant, and is therefore the first barrier against release of fission products to the environment. The design requirements for the fuel help ensure that cladding integrity will be maintained and its safety functions will be fulfilled. The two principal requirements that have been established to meet the safety and reliability functions for the 4S fuel elements are based on experience with EBR-II. These requirements are:

a. Ensure that sufficient fuel pin reliability is maintained so as to statistically prevent a significant number of fuel pin breaches during normal and off-normal reactor operation including postulated accidents. A significant number of breaches may be defined as that which challenges the safe operation or performance goals of the reactor. Here, it is defined as ensuring that no more than one fuel breach is expected per effective full-power year of operation.

b. Maintain a coolable geometry of both the fuel pin and the fuel pin bundle for the useful lifetime of the subassembly, including normal operation and all off-normal events (AOOs and postulated accidents).

During normal operation, anticipated events and postulated accidents, the low probability of fuel pin failure provides added confidence that the fuel pins will remain intact and, therefore, that the coolable pin bundle geometry will be maintained. The design criteria for normal operation are reported in [12]. In addition, established limits on fuel melting (see next section), fuel/cladding eutectic formation, maximum cladding stress, and maximum cladding strain restrict the number of pin failures throughout the life of the core.

2. Design Criteria for Transients

The design criteria and analysis methods presented in this section address the geometry, temperature regimes, and loading mechanisms that are expected to influence fuel performance during transients. They are applied to the most limiting fuel pin in the core.

Although it may be desirable to have a single set of criteria for all conditions (steady state and transient), it is recognized that, because of the widely different time scales that are involved, separate criteria are necessary.

The two principal functional fuel design requirements are related to fuel reliability and maintaining a coolable geometry of both the fuel pin and the fuel pin bundle during all off-normal events, including AOOs, DBAs, and ATWS. Table 3.4-1 summarizes the transient design criteria associated with these off-normal events and their association with the functional requirements.

a. Fuel Cladding Integrity

As shown in Table 3.4-1, the fuel cladding integrity-related criteria do not allow fuel melting during anticipated events and limit the CDF accumulated during anticipated events and the single most damaging DBA to a value that is less than 0.1. These criteria ensure that the cladding integrity is maintained with sufficient margin. The effects of steady-state and transient wastage due to metallurgical interaction between the fuel and cladding are accounted for by assuming that interaction zones are strengthless.

The transient CDF calculations for HT9 are based on the results of FCTT tests that provide transient time-to-rupture data for HT9 [20, 21]. In these tests, cladding tubes cut from irradiated fuel pins are pressurized and subjected to transient heating at various temperature ramp rates. The measured failure temperatures are used to determine time-to-rupture correlations as a function of stress and temperature. The transient failure strains are also measured. The basis for the allowable transient CDF value of 0.1 is a statistical analysis of the FCTT test data, where it was shown that a transient CDF less than 0.10 is sufficient to ensure, with 95 percent confidence, that the failure probability will be less than 1/3000. This probability would imply only a single failure over the life of the 4S core, which would satisfy by a large margin the functional requirement that allows one failure per full-power year.

b. Coolable Geometry

The functional requirement of coolable geometry is satisfied if cladding deformation due to creep strain is less than or equal to that required to result in touching of the cladding of adjacent fuel pins. Creep strain is accumulated during normal operation, AOOs, DBAs, and ATWS events. Since DBAs and ATWS events are not expected to occur during the plant lifetime, it is conservative to limit the total strain from normal operation and all AOOs, plus that from the most damaging DBA and ATWS to a value that maintains the coolable geometry. Given the pin pitch configuration of the 4S fuel, a creep strain of 8 percent would result cladding contact between adjacent pins. Figure 3.4-1 shows the coolable geometry of the fuel where the 8 percent strain corresponds to the strain at which the pin-to-pin space is eliminated. In this configuration, approximately 40 percent of the original flow area remains, so that coolable geometry is maintained. Note that, since the design criteria apply to the most limiting pin, the creep strain limit would only be realized in one or several fuel pins. Thus, the situation depicted in Figure 3.4-1 would apply only locally.

While cladding strain is the primary criterion relied upon to ensure a coolable geometry throughout core life, including consideration of an ATWS event, CDF-related criteria are retained in Table 3.4-1. This is done because the CDF accumulation can be used as a surrogate (or substitute) criterion for the cladding strain from thermal creep. Cladding strain and CDF are interrelated (typically cladding failures take place at thermal creep strains that are much less than 8 percent), so that satisfying the CDF-related criteria will ensure that the cladding strain criterion is also satisfied. The calculation of CDF is much simpler than that for cladding strain, so using the surrogate CDF criterion simplifies the calculation procedure.

c. Fuel Melting

To ensure that the reactor fuel can continue to be used to its design lifetime following an AOO, it is considered necessary that fuel relocation that might occur with significant fuel melting be avoided. A transient design criterion to achieve this objective is stated simply as "no fuel melting" during AOOs and is included in Table 3.4-1. This simply-stated criterion should be understood to mean that there should be no centerline fuel melting of the fuel slug and no significant formation of a liquid eutectic phase at the fuel/cladding interface.

The first criterion means that the centerline temperature should not exceed the solidus temperature of the U-Zr fuel alloy, which should be approximately 1100-1150°C, depending on the composition of the alloy.

Formation of an uranium-iron eutectic at the fuel cladding interface is a complex function of time, temperature, fuel and cladding composition, burnup, and conditions during irradiation. An experiment using U-19Pu-10Zr fuel and HT9 cladding, irradiated at higher linear power (~0.5 kW/cm) and to higher burnup (11 at%) than would be the case in 4S, showed no eutectic formation for 36 hours at 650°C. An experiment on a similar sample irradiated to 6 at% burnup at the same conditions showed no eutectic formation after 1 hour at 738°C. These data give a conservative indication of the conditions to which 4S fuel could be subjected in an AOO without eutectic formation.

3.4.2.2 Integrity of the Primary Coolant Boundary

The Class 1 component standards of The ASME at elevated temperatures are applied to the requirement to maintain the primary coolant boundary integrity. The codes to be applied are the ASME Code Section III, subsection NB (Class 1 components) and subsection NH (Class 1 components in elevated temperature service appendices). The limits for the primary coolant boundary depend on the pressure and temperature histories and also on the number of event occurrences. The limit levels for each category are shown in Table 3.4-2; these service levels are equivalent to those for a light water reactor.

3.4.2.3 Offsite Radiation Dose

The 4S design complies with the offsite dose guidelines for DBAs as follows from 10 CFR 50.34 [22], 10 CFR 52.47 [23], and 10 CFR 100 [24]:

1. *An individual located at any point on the boundary of the exclusion area (EAB) for any 2 hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE).*

2. *An individual located at any point on the outer boundary of the low population zone (LPZ), who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE).*

3.4.2.4 Containment Integrity

The design criteria for containment integrity are excerpted from SRP 15.8:

> *Following a failure to scram, the containment pressure and temperature must be maintained at acceptably low levels based on GDC 16 and 38* [25]. *The containment pressure and temperature limits are design dependent; but to satisfy GDC 50, those limits must ensure that containment design leakage rates are not exceeded when subjected to the calculated pressure and temperature conditions resulting from any ATWS event.*

Table 3.4-1
Transient Design Criteria for the Fuel

Functional Requirement	Off-Normal Event		
	AOO	**DBA**	**ATWS**
Fuel Cladding Integrity	No fuel melting	N/A	N/A
	$\sum_{i=1}^{M}\sum_{j=1}^{N_i}(CDF_{AOO})_{ij} + CDF_{DBA} \leq 0.1$		
Coolable Geometry	$\sum_{i=1}^{M}\sum_{j=1}^{N_i}(CDF_{AOO})_{ij} + CDF_{DBA} \leq 0.1$		$CDF_{ATWS} \leq 0.1$
	$\left(\sum_{i=1}^{M}\sum_{j=1}^{N_i}(\varepsilon_{AOO})_{ij} + \varepsilon_{DBA} + \varepsilon_{ATW} + \varepsilon_{SS} \leq 8\%\right)$		

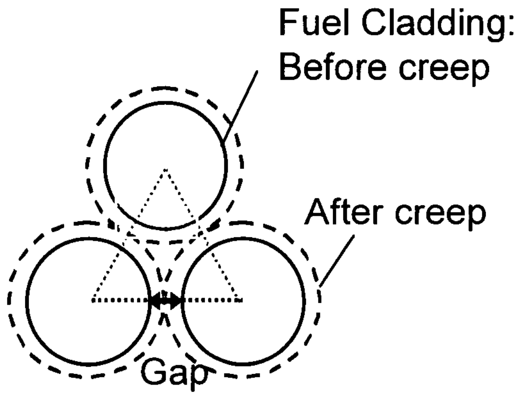
Notes:

$\sum_{i=1}^{M}\sum_{j=1}^{N_i}(CDF_{AOO})_{ij}$	=	cumulative damage function during all M anticipated events and all N_i occurrences
CDF_{DBA}	=	cumulative damage function during the single most damaging design basis event
CDF_{ATWS}	=	cumulative damage function during the single most damaging anticipated transient without scram
$\sum_{i=1}^{M}\sum_{j=1}^{N_i}(\varepsilon_{AOO})_{ij}$	=	strain associated with all M anticipated events and all N_i occurrences
ε_{DBA}	=	plastic hoop strain for the single most damaging design base event
ε_{ATW}	=	plastic hoop strain the ATWS event
ε_{SS}	=	steady-state plastic hoop strain

Table 3.4-2
Acceptance Limits for Primary Coolant Boundary

Category	Primary Coolant Boundary Limits
AOO	Service Level B Limits or Service Level C Limits(1)
DBA	Service Level D Limits(1)
ATWS	Service Level D Limits(1)

Note:
1. Refer to ASME B&PV Code Section III, Division 1, NCA 2142-4, 2007.



Notes:

1. Area of subchannel after 8 percent diametral strain.

2. In 4S fuel specification, pin diameter is 14 mm and pin-pin pitch is 15.1 mm. The area of the nominal inner subchannel is 21.8 mm^2. If the pin diameter expands by 8 percent, the inner subchannel area is 9.2 mm^2. The ratio of the flow area after expansion to that in nominal geometry is 0.43:1. Thus, an area for axial flow is maintained after expansion. Note that this situation would apply only locally in the core.

Figure 3.4-1 Pin Pitch and Coolable Geometry of Fuel Pin

4 ANALYSIS METHODOLOGY

This section describes the plant conditions and assumptions used in the 4S safety analysis (Section 4.1), and the methods used in that analysis (Section 4.2). These conditions, assumptions, and methods have been developed specifically for the 4S liquid metal reactor and the metal fuel used in that reactor.

As seen in Section 4.1, the plant conditions and assumptions depend on the severity of the initiating events (DBE or ATWS) and cover factors that affect the consequences of these events, e.g.,

- The time at which the initiating event occurs
- Availability and effectiveness of safety systems, structures, components and passive features (e.g. single failure criteria, reactivity feedback mechanisms, convective heat transfer by natural circulation)

Section 4.2 describes:

1. The computer code used for dynamic analysis (ARGO code), and
2. The analytical procedures and correlations used for evaluating radioactive material release and transport, and the dose at the exclusion area boundary and low-population zone.

4.1 Plant Conditions Assumed in the Safety Analysis

DBEs include AOOs and DBAs, and they are treated conservatively. ATWS events are beyond DBEs and are treated on the basis of best estimate. AOO, DBA, and ATWS analysis conditions are as shown in Table 4.1.1-1.

4.1.1 Conditions Assumed for DBEs (AOOs and DBAs)

4.1.1.1 Items to be Considered in the Analysis

1. Time in Reactor Lifetime to be Considered during Analysis

   The entire 30-year operational term from the completion of performance tests to just before decommissioning the reactor is usually considered when analyzing the assumed event. The most severe initial state during core life is selected according to the acceptance criteria and in consideration of the long-term changes due burnup and the various operating modes assumed during operation.

   A given transient calculation is continued until it is clear that the reactor can be brought to the shutdown sate.

2. Assumptions for the Safety Functions

a. The safety functions of the safety structures, systems, and components were considered in the analysis conditions.

b. The single failure of an active component that is most limiting with respect to performance of the basic safety functions, such as reactor shutdown, core cooling, and containment of radioactivity, is assumed for the system.

c. Appropriate time is considered for performance of any operator actions required to cope with the event.

d. The type of signal and time in a transient at which a signal is generated for the operation of the plant protection system are clearly defined. Also, if operation of a system other than the plant protection system has a significant influence on the result of an analysis, the type of signal and the time at which it is generated should be specified alike.

e. When calculating the scram effect, the type of signal triggering a scram is determined and a proper scram delay time is specified.

f. Cases in which offsite power cannot be used are also considered when calculating the effect of operation of engineered safety systems.

4.1.1.2 Plant Conditions

1. Plant Heat Balance

The plant thermal-hydraulic parameters of the 4S are as shown in Table 4.1.1-2.

The establishment of thermal-hydraulic parameters for the analysis is as described below:

a. Use heat-balanced temperature by focusing on consistency with other system design section

b. Consider 2 percent uncertainty for the reactor thermal power

Uncertainties of coolant flow rate and temperature are considered in determining a coefficient that multiplies the hottest fuel pin temperature calculated in a given safety analysis case.

4.1.1.3 Core Conditions

1. Core State

   The 4S reactor requires no refueling for 30 years. Movement of a reflector and fixed absorber, compensate for the burnup reactivity in order to maintain full power operation capability. The representative core states during reactor lifetime are as follows:

   - Beginning of life: BOL (state in which the reflector covers the bottom of the core and the fixed absorber is inserted)
   - Middle of life I: MOL I (state in which the reflector covers the core and the fixed absorber is inserted)
   - Middle of life II: MOL II (state in which the reflector covers the bottom of the core and the fixed absorber is withdrawn)
   - End of life: EOL (state in which the reflector covers the core and the fixed absorber is withdrawn)

   During the core states described above, there is little difference in the distribution of reactivity coefficients and the peak value of power distribution. However at the end of life, internal pin pressure and cladding corrosion are at their highest value and presents the most severe conditions for the fuel cladding. Thus, EOL is selected as the time in core life to be analyzed.

2. Power Distribution

   The circumferential power distribution of the 4S reactor is almost uniform because the core and control elements are axisymmetric and the burnup proceeds uniformly in the circumferential direction. Therefore, in the analysis, the reactor core is divided into three zones (inner core [IC], middle core [MC], and outer core [OC]) as shown in Figure 4.1.1-1. The maximum linear heat generation is shown in Table 4.1.3-1.

   The integrated power and axial distribution in each zone were evaluated for each time in life described above, and the power distribution at EOL was chosen for analysis. The maximum linear power and the maximum cladding temperature of the selected pin (representative fuel pin) for analysis were determined, so as to select the highest values of the thermal characteristics from the four core burnup states given above. Figure 4.1.1-2 shows the axial distribution at EOL in the three zones.

   The radial heat transfer between assemblies and the periphery of the core region is not considered.

3. Reactivity Coefficients

a. Reactivity Feedbacks to be Considered

For analysis, the reactivity changes caused by the Doppler effect and the density change in core constituent materials (fuel, coolant, and structural material), are considered. The reactivity change due to the radial expansion of the core support plate is also considered.

The reference values of the reactivity coefficients, density coefficients, and geometry coefficients are shown in Table 4.1.1-3.

b. Regions in which the Reactivity Feedback Effects Occur

Reactivity feedback in the core region is considered while reactivity feedback from other regions is treated as follows:

As shown in Table 4.1.1-3, the coolant density coefficients are positive in the reflector and inside the core barrel region. In case of temperature rise in these regions, negative reactivity is inserted. Thus, the reactivity feedback from these regions is not considered in conservative calculations. On the other hand, the coolant density coefficient in the shutdown rod region and the structure density coefficient in the shutdown rod region are negative. In case of temperature rise in these regions, positive reactivity is inserted. Therefore, the reactivity feedback from these regions is considered during the analysis.

c. Uncertainty of the Reactivity Coefficients

The reactivity coefficients used in the safety analyses were selected from the characteristics of the EOL, through comparison of the characteristics among BOL, MOL1, MOL2 and EOL. For each of these four combinations of burnup and reflector positions, reactivity uncertainties associated with fuel performance were taken into account in the analysis model. These uncertainties are caused by the slight shift of the top position of reflectors above and below the reference position for criticality. Those shift of the top position of reflectors in the four core stages, inclusive of EOL, resulted in deviation of distribution of the reactivity. The amount of the deviation was treated as an uncertainty for the safety analysis. Prediction accuracy such as correction factors of the core characteristic data were estimated considering the results of analysis of experiments [26] conducted at the Fast Critical Assembly (FCA) facility of the Japan Atomic Energy Research Institute (JAERI).

d. Combination of the Reactivity Coefficients

The reactivity coefficients are set with consideration of uncertainty. The reactivity coefficients (Table 4.1.1-4), used for analysis are the summation of the reference value of the reactivity coefficient (Table 4.1.1-3) and associated uncertainty. Considering the temperature rise and fall of each element (e.g., fuel pin, cladding, coolant, core, support, etc.), the uncertainty of the reactivity is set to add conservatism to prediction of the reactor power changes prior to scram. For example, in case of negative temperature reactivity coefficient associated with an element temperature rise, its uncertainty is set to make the absolute value of reactivity coefficient smaller. This means the negative reactivity contribution due to fuel temperature increase is smaller. The Doppler coefficients, density coefficients, and geometry coefficient used for analysis are shown in Table 4.1.1-4 for the 4S core.

4. Fuel Pin Temperature Conditions

The conditions of the representative fuel pin, i.e., hottest pin, for evaluating the fuel cladding integrity are as described below.

a. Maximum Temperature of the Fuel Cladding at Rated Power

The maximum temperature of the fuel cladding at rated power is determined to be that of the nominal hottest pin with consideration of the uncertainties of power, flow, and fabrication tolerance. The nominal hottest pin is the fuel pin with the highest temperature in a subassembly under nominal core fuel design condition, and it is evaluated from the power distribution, flow rate and distribution in a subassembly [27].

b. Maximum Linear Power

Considering estimated uncertainties as the design margin, 97 W/cm is selected as the maximum linear power rate.

c. Heat Transfer Between the Fuel and Cladding

For the 4S metallic fuel, the gap between the fuel slug and cladding is filled with a sodium thermal bond. At EOL, the fuel slug will have expanded so it adheres closely to the cladding at some locations in the core. In either case, the thermal resistance includes the thermal resistance of the sodium bond in the fuel pin.

5. Decay Heat Conditions

The decay heat after 30-year burnup was obtained from nuclear calculations. An uncertainty of +20 percent was included for use in the analysis.

4.1.1.4 Plant Protection System Conditions

The setting values and response times of the plant protection systems used for the analysis are collectively shown in Table 4.1.1-5. The plant protection system has a two-out-of-three configuration [13].

For a plant protection system signal, the first signal generated during the transient is usually considered in the analysis.

The insertion reactivity during a reactor scram, a scram reactivity insertion curve, and a reactor trip sequence are based on the description below.

1. Insertion Reactivity Conditions during a Reactor Scram

   The insertion reactivity during a reactor scram is prescribed as the reactivity value with one reflector segment stuck.

2. Scram Reactivity Insertion Curve

   Set scram reactivity worth and insertion rate value to be conservative based on the pre-scram position of reflector, insertion amount (scram stroke), reactivity worth curve of reflector, and insertion time (design value less than 8 seconds).

3. Reactor Trip Sequence

   The reactor trip sequence is shown in Table 4.1.1-6.

   To prevent a pump trip, before the reflector moves to shutdown the reactor, the design incorporates a timer so that the pump trip sequence is delayed to the same time as the delatch delay time of the reflector (dt2 = dt3).

   During analysis, it was assumed that the pump trip precedes lowering of the reflector; it trips with a delay time of 0 seconds, as shown in Table 4.1.1-6. This assumption is conservative, since the coolant temperature increases due to the delatch delay of the reflector drive mechanism.

4.1.1.5 RHR System Conditions

1. Modeling of the RHR System

   a. The air cooler (AC) of the IRACS is sized considering forced-circulation conditions. For the heat removal capacity during natural circulation operation, the specific geometric shape of the AC is modeled.

   b. RVACS is modeled so that the heat removal during the transient by the RVACS is evaluated from the wall temperature of the reactor vessel and the natural convection air flow for the designed heat transfer configuration (reactor vessel, guard vessel, collector and airflow path).

2. RHR Sequence

   a. The primary and intermediate-loop circulation pumps are tripped by a reactor scram.

   b. After trip, power to the primary- and intermediate-loop circulation pumps is provided by a motor-generator (MG) set and coastdown is performed with a flow rate halving time of 30 seconds [13].

   c. The steam water system is non-safety grade and residual steam in the steam water system removes the residual heat of the heat transport system. In the analysis, residual steam in the steam water system is not made to contribute to the heat removal.

3. IRACS Start

   The ordinary start sequence of the IRACS is shown in Table 4.1.1-7. After the reactor trips, the AC damper is opened using a decay heat removal start signal, initiating the heat removal. The startup logic for the blower considers the intermediate-loop coolant temperature to prevent an excessive thermal transient of the structure and the potential for overcooling the sodium. If necessary, the protection system will start the blower and intermediate-loop circulation pump to remove the decay heat by forced circulation.

4. Conditions to be assumed by the Safety Analysis

   a. The coastdown equipment of the intermediate-loop circulation pump in step (2)(b) is not safety grade; therefore, no coastdown is assumed after the pump trips.

b. The heat removal by the residual steam in steam generator in step (2)(c) is not considered. On the conservative side, the steam generator is instantaneously thermally insulated after tripping.

c. For heat removal by IRACS, heat is removed from the primary and intermediate loops and air side using natural circulation flow. The failure of one air-side damper (one of either of the two intake or two exhaust dampers) is assumed.

4.1.1.6 Single Failure Conditions

A single failure is presumed for an active component of safety-related SSC. In the sequence after the reactor shutdown, the systems with the function described above are the plant protection system, flow coastdown system, main reactor shutdown system, and decay heat removal system. Table 4.1.1-8 shows the list of single failures to be considered during the safety analysis and the single failures assumed during event analysis.

4.1.1.7 Radiological Consequence Analysis Conditions

In this section, the source term conditions in the radiological analyses are described and the effects of source term reduction and radioactive release reduction are taken into account.

1. Reduction of Source Term and Radioactive Release

   The safety characteristics of 4S related to source term reduction and radiological release reduction are as follows:

   a. Source Term Reduction

      - Low power results in low fission product (FP) inventory.
      - Sodium's affinity for FPs minimizes release.
      - There is no significant release to containment due to the absence of energetic and pressurization events.

   b. Radioactive Release Reduction

      - Reactor vessel and containment are sealed.
      - Penetrations and isolation valves are kept to a minimum.
      - Threat to containment integrity is minimal due to absence of damaging phenomena (direct containment heating, steam explosion, hydrogen burning or detonation, missiles).

2. Source Term Definition

Radionuclide Groups

The elements to be evaluated and their element grouping were specified based on RG 1.183 [28]. Because uranium is not defined in RG 1.183, it was included in the cerium group. Since the operation of liquid metal-cooled reactors results in the activation of the sodium coolant, a separate grouping was added for the coolant.

The element grouping names and breakdowns are indicated as follows:

- 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
- Cerium group: Ce, Pu, Np, U
- Lanthanides: La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Am, Cm
- Coolant: Na

Nuclides with a half-life of more than 1 minute are considered.

3. Inventory

In general, the radiological inventory in a core is proportional to the thermal power; the inventory is gradually accumulated, depending on the nuclides, over the 30-year lifetime. Therefore, the radiological inventory value is assumed at the point where 30 years of operation have elapsed since the plant startup. The inventory of each radionuclide is evaluated by ORIGEN code.

The radiological core inventory may include errors in excess of 10 percent in magnitude by taking in account various uncertainties associated with fuel mass in the core. Consequently, the nominal value of the radiological inventory is multiplied by a factor of 1.5 as an uncertainty margin to give the radiological inventory for this analysis.

4. Release Path

The 4S metallic fuel permits run beyond cladding breach (RBCB). Therefore, a fuel failure rate of 1 percent is applied as the operational allowable limit. For the 4S core, it is normally difficult to postulate fuel failure resulting from the operating conditions, transient conditions, and analysis evaluation.

Figure 4.1.1-3 shows the radioactive release path used in the 4S source term evaluation. The release path includes transport of the fission products from the damaged fuel to the primary coolant, release of fission products and activated sodium from the primary coolant to the cover gas space, leakage from the cover gas space to the top dome via the shielding plug and its associated equipment, and leakage from the top dome into the environment.

The fractions of release from the fuel to the primary sodium and from the primary sodium to the cover gas space are discussed in items (a) and (b) below. Items (c) and (d) present the assumed leak rates for the cover gas and containment dome boundaries.

a. Release from the Core to Primary Sodium

i. Noble Gases

The fission gas behavior in the 4S metallic fuel is discussed in the report submitted to the NRC [12]. The report shows that the fission gas is retained mostly in the fuel for burnups less than 1-2 %. At higher burnups, passageways are created within the fuel allowing transport of the fission gas to the fuel pin gas plenum. At the end of life, the fraction of the fission gas residing in the gas plenum reaches 70 to 80 % [29]. The remaining 20 to 30 % are retained in the fuel. In this analysis, no credit is taken for fuel retention and 100% of the fission gas is assumed to be released instantaneously to the primary sodium on clad failure.

ii. Halogens and Alkali Metals

The behavior of iodine and other halogens in the sodium-bonded metallic fuel used in the 4S reactor is quite different from that in oxide fuels such as those used in LWRs. While the formation of cesium iodide is possible for both types of fuel, the possibility of having elemental iodine in the 4S fuel is made extremely remote by the presence of uranium metal (to form UI_3 [uranium tri-iodide]) and sodium (to form NaI [sodium iodide]). Results of the RBCB experiments for sodium-bonded metallic fuel [30] show no measurable amount of iodine release from the breached cladding although cesium has been released. This suggests that iodine has been retained in the fuel as UI_3.

Cesium iodide is less volatile than cesium and iodine. Therefore, its formation reduces the release fraction of both Cs and I. In this analysis, the following conservative assumptions are made:

1. No iodine is retained in the fuel as UI_3

2. 100% of the cesium inventory is released from the fuel to the primary sodium as elemental Cs (No CsI is formed)

3. 100% of the iodine is released from the fuel to the primary sodium[1]

The same release fractions are used for the halide and alkali metal fission product groups. The release from the fuel is assumed to be instantaneous at the time of fuel failure.

iii. Tellurium Group

Elements in this group interact with sodium to form Na_2X compounds, e.g., Na_2Te for tellurium. It is conservatively assumed in this analysis that 100% of the inventory in this group is involved in this reaction with the fuel bond sodium. It is also assumed that the release of the bond sodium to the primary sodium occurs instantaneously at the time of fuel failure.

iv. Barium, Strontium

Barium and strontium dissolve in sodium at the coolant temperatures for cold shutdown and normal operation [31]. However, the melting points of Ba and Sr are higher than the peak fuel temperature estimated for DBAs and most of the inventory will likely be retained in the fuel. This limits the fraction that could be dissolved in the bond sodium. In this analysis, it is conservatively assumed that 100% of the inventory of this group is dissolved in the bond sodium and released instantaneously from the fuel at the time of fuel failure.

v. Noble Metals

Noble metals have melting points that are significantly higher than that of the metallic fuel (ranging from 400°C higher for Pd to ~1,500°C for Mo). Elements in the group do not react substantially with sodium and have low solubility. Consequently, a release fraction of 0.1% is assumed in this analysis which is believed to be conservative. Similar to other groups, the release of this fraction into the primary sodium is assumed be instantaneous at the time of fuel failure.

[1] Due to the presence of Na inside the 4S metallic fuel, a possible chemical form of iodine (other than UI_3) is sodium iodide. Even if some elemental iodine is released from the fuel, the abundance of the primary sodium will make it extremely unlikely that elemental iodine will be released to the cover gas space in the reactor vessel.

vi. Cerium Group and Lanthanides

The elements in this group hardly react with sodium and have low solubility. According to the Central Research Institute of Electric Power Industry's (CRIEPI's) research, the solubility of plutonium per 1 gram of sodium is 0.048 micrograms per month at 600°C [32]. This analysis assumes a solubility of 1 microgram of plutonium per gram per month. Amount of plutonium dissolved into bond sodium of fuel pins during the life time (i.e., about 360 months) is less than 0.1% of total plutonium amount. A release fraction at 1% pin failure is estimated to be less than 0.001% at the fuel end of life, conservatively. The same release fraction is assumed for other elements in the group. Similar to other groups, the release of this fraction into the primary sodium is assumed to be instantaneous at the time of fuel failure.

b. Release from Primary Sodium to the Cover Gas Space

i. Noble Gases

Noble gases hardly react with sodium. Therefore, 100% of the noble gases released from the damaged fuel are assumed to be instantaneously released to the cover gas region.

ii. Primary Sodium

The fraction of primary sodium that resides as vapor in the cover gas space (F_{Na}) is estimated using: 1) the sodium partial pressure at the 650°C assumed for the analysis, 2) volume of the cover gas space, and 3) mass of the liquid sodium in the reactor vessel. The analysis treats the sodium vapor as ideal gas. The fraction F_{Na} is derived using the following equation for the molar ratio of the sodium gas to sodium liquid:

$$F_{Na} = \frac{n_\ell^g}{n_\ell^\ell} = \frac{\dfrac{P_\ell^0}{RT} V_g}{\dfrac{\rho}{A_{Na}} V_\ell} \qquad (4.1\text{-}1)$$

where,

n_ℓ^g = sodium mole number in the cover gas (mol)
n_ℓ^ℓ = sodium mole number in the sodium (mol)
P_ℓ^0 = saturated vapor pressure of the sodium (Pa)
V_g = cover gas volume (m^3)

V_ℓ = sodium volume (m^3)
ρ = sodium density (g/m^3)
T = temperature (K)
A_{Na} = sodium atomic mass (g/mol)

The sodium saturated vapor pressure was estimated using the following equation [33].

$$P_\ell^0 = 1.01329 \times 10^5 \times \exp(18.832 - \frac{13113}{T} - 1.0948\ln T + 1.9777 \times 10^{-4} T) \quad (4.1\text{-}2)$$

Figure 4.1.1-4 shows the temperature dependency of the sodium mass fraction in the cover gas space using equation (4.1-1). As seen in the figure, the fraction is 2.2×10^{-6} at the assumed temperature of 650 °C. A rounded value of 3×10^{-6} is used in the calculation of the source term.

iii. Halogens and Alkali Metals

The fraction of halogens and alkali metals released to the cover gas (F) is estimated using the following equation, where K_d is the gas-liquid equilibrium constant for halogens and alkali metals:

$$F = K_d \cdot F_{Na} \quad (4.1\text{-}3)$$

The release factions of iodine and cesium are representative of the halogen and alkali metals groups. The gas-liquid equilibrium constant, K_d, for iodine and cesium, is based on the maximum value found in existing literature [34, 35]. Iodine exists in the primary sodium as sodium iodide (NaI), thus the gas-liquid equilibrium constant of NaI is used. Cesium remains in its elemental form in the primary sodium.

The gas-liquid equilibrium constants of NaI and Cs are shown in equations (4.1-4) and (4.1-5) below:

$$\text{NaI:} \quad \log_{10} K_d = \frac{592}{T} + 0.5 \log T - 1.79 \quad (4.1\text{-}4)$$

$$\text{Cs:} \quad \log_{10} K_d = \frac{987}{T} + 0.490 \quad (4.1\text{-}5)$$

Figure 4.1.1-4 shows the temperature dependency of the release fraction for NaI and cesium in the 4S reactor system, as calculated using equations 4.1-3, 4.1.4 and 4.1.5. For the assumed primary sodium temperature of 650°C, the release fractions from primary sodium to the cover gas space are 4.8×10^{-6} for NaI and

8.1x10^{-5} for cesium. Rounded release fraction values of 5x10^{-6} and 1x10^{-4} have been used for source term analysis. Similar to Noble gases, the release of the halogens and alkali metals is assumed to occur instantaneously at the time of fuel failure.

iv. Tellurium Group, Barium, Strontium, Noble Metals, Cerium Group, and Lanthanides

Elements in these groups have very small saturated vapor pressure compared with the sodium, halogens, and alkali metals. Therefore, the release fraction of these elements should be as low as that of sodium or lower. The release fraction of sodium (3x10^{-6}) is used for estimating the release of the above groups to the cover gas space. As for previous groups, the release is assumed to occur instantaneously at the time of fuel failure.

Table 4.1.1-9 presents the estimated release fractions from the damaged fuel to the primary sodium and from the primary sodium to the cover gas space. Table 4.1.1-9 also shows the release fractions in LWRs for reference [28]. Compared to the instantaneous release assumed for the 4S, the LWR release occurs in two phases, an early (gap) release and a later phase [28].

c. Leakage from the Cover Gas Space to the Top Dome

The shielding plug and its peripheral block between the cover gas space and the top dome are designed with a double boundary. The source term analysis uses the design leakage rate of 1 percent per day at the cover gas design pressure and temperature. This leakage rate is conservative for two reasons: 1) The analyzed DBAs show cover gas pressure and temperature that are smaller than their design values, and 2) The used leak rate does not account for the possibility of blockage of leak passages by sodium and other aerosols.

d. Leakage from the Top Dome to the Environment

Similar to the leakage from the cover gas, the design leakage rate of the top dome of 1 percent per day is used in the analysis. This leak rate is also conservative.

4.1.1.8 Analysis Procedure for AOOs and DBAs

AOOs and DBAs were analyzed using conservative conditions based on the described analysis conditions from subsections 4.1.1.1 to 4.1.1.7. The analysis results were compared with the safety criteria described in Chapter 3, and the validity of the safety design of the 4S was demonstrated.

Table 4.1.1-1
AOO, DBA, and ATWS Analysis Conditions

Attributes \ Analysis Event	AOO and DBA	ATWS
Reactivity Feedback	Combination of minimum and maximum to obtain conservative value	Reference value and uncertainty (95/95)
Power and Flow Rate		
Material Properties		
Plant Parameters (Pressure Loss, Halving Time during Coastdown, etc.)		

Notes:
1. AOO and DBA both use the same degree of conservatism.
2. For ATWS, conduct reference condition analysis and statistical analysis to obtain 95 percent probability at a 95 percent confidence level.

Table 4.1.1-2
Plant Thermal-Hydraulic Parameters

Item		Design Value	Unit
Main Cooling System at Rated Power	Reactor Thermal Power	30	MWt
	Primary Coolant Outlet/Inlet Temperature	510/355	°C
	Primary Coolant Flow	5.47×10^5	kg/h
	Intermediate Coolant Outlet/Inlet Temperature	485/310	°C
	Intermediate Coolant Flow	4.82×10^5	kg/h
	Feedwater/Steam Temperature	210/453	°C
	Steam Generator Water/Steam Flow	4.42×10^4	kg/h
	Steam Pressure	10.5	MPa

Table 4.1.1-3
Reference Values of Doppler, Density, and Geometry Reactivity Coefficients

		Region			
Attribute	**Definition**	**Core**	**Shutdown Rod**	**Inside Core Barrel**	**Reflector**
Doppler Reactivity Coefficient	Tdk/dT	-3.8×10^{-3}	-	-	-
Fuel Density Coefficient	dk/kk'/dro/ro	3.6×10^{-1}	-	-	-
Coolant Density Coefficient	dk/kk'/dro/ro	2.8×10^{-4}	-1.2×10^{-3}	1.5×10^{-2}	3.3×10^{-3}
Structure Density Coefficient	dk/kk'/dro/ro	-1.5×10^{-2}	-4.9×10^{-4}	-	-
Core Support Structure Expansion	dk/kk'/°C	-5.9×10^{-6}			

Note:
ro: density

Table 4.1.1-4
Doppler, Density, and Geometry Reactivity Coefficient to be Used for Analysis

Attribute	Definition	Condition	Core
Doppler Reactivity Coefficient	Tdk/dT	At temperature rise	-3.2×10^{-3}
		At temperature fall	-4.3×10^{-3}
Fuel Density Coefficient	dk/kk'/dro/ro	At temperature rise	3.2×10^{-1}
		At temperature fall	4.0×10^{-1}
Coolant Density Coefficient	dk/kk'/dro/ro	At temperature rise	-1.5×10^{-3}
		At temperature fall	-1.0×10^{-4}
Structure Density Coefficient	dk/kk'/dko/ro	At temperature rise	-2.2×10^{-2}
		At temperature fall	-8.4×10^{-3}
Core Support Structure Expansion	dk/kk'/°C	At temperature rise	-4.1×10^{-6}
		At temperature fall	-7.6×10^{-6}

Table 4.1.1-5
Set Values and Response Times of the Safety Protection Systems Used for Analysis

Signal	Analytical Value	Response Time(1, 2)	
		t1(sec)	t2(sec)
Main reactor shutdown system signal	-	-	-
Reactor power-range monitor at flux level low (low setting)	50%	0.5	-
Reactor power-range monitor at flux level high (high setting)	120%	0.5	-
Intermediate heat exchanger primary outlet temperature high	390°C	1.0	30.0
Intermediate heat exchanger primary outlet temperature low	325°C	1.0	30.0
Voltage of power line	65%	1.0	-
Backup reactor shutdown system signal	-	-	-
Wide-range monitor at flux level low (low setting)	50%	0.6	-
Wide-range monitor at flux level high (high setting)	120%	0.6	-
Primary main circulation EM pump voltage low	80%	1.0	-

Notes:
1. Response time is time from when variable reaches trip set value to when trip breaker opening action is completed.
2. The meanings of t1 and t2 are as follows:
   t1: Time delay component (relay operation delay, etc.)
   t2: The first-order delay component (thermometer well response delay, etc.)

Table 4.1.1-6
Sequence of the Reactor Trip

State	Time	Notes
Transient State Starts	0	-
Reach Signal Set Point	t_o	-
Finish Trip to the Scram Circuit Breaker	$t_o + dt_1$	dt_1: response time of the signal (Table 4.1.1-5)
Start Trip to the Pumps	$t_o + dt_1 + dt_2$	dt_2: delay time of the pump trip (= 0s)
Start to Lower the Reflector	$t_o + dt_1 + dt_3$	dt_3: delay time of the response of latch of the reflector drive mechanism (= 0.5s)

Table 4.1.1-7 Start Sequence of IRACS		
State	**Time**	**Notes**
Transient state start Closed state of the AC damper of IRACS (on standby)	0	-
Transmittal of the reactor trip signal	t_0	-
Start opening the AC damper of IRACS	$t_0 + dt_1$	dt_1: delay time of opening the damper (1.0s)
Finish opening the AC damper of IRACS (full close to open state)	$t_0 + dt_1 + dt_2$	dt_2: required time to finish opening the damper (30.0s)

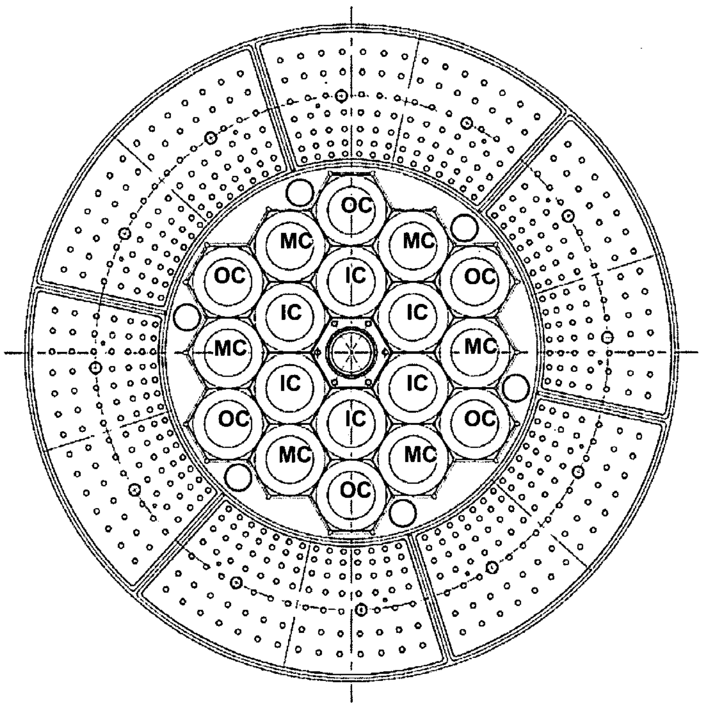
Table 4.1.1-8
Single Failures to be Considered at Safety Analysis and Assumed at Each Event Analysis

Event		Single Failure: Sensor of Reactor Protection System	Single Failure: Sensor of Systems Except for Reactor Protection System	Single Failure: Lowering of the Reflector	Single Failure: Primary Flow Coastdown System	Single Failure: Residual Heat Removal System	Single Failure: Containment
AOO	Loss of offsite power	Single failure does not result in degradation and/or loss of the performance of the safety function because of channel redundancy and diversity.	Does not rely on the actuation of the detection system. (N/A)	One stuck reflector segment is considered	Single failure does not result in degradation and/or loss of the safety performance because active components have redundancy.	Loss of offsite power and failure of the emergency power generator, → failure of the intermediate pump to start, failure of IRACS blower to start, Either one of the two intakes or two exhaust dampers fail to open.	Loss of offsite power and failure of the emergency power generator → failure of the isolation valve (main feedwater/ steam isolation valve). Single failure does not result in the loss of the leak tightness of the containment because of redundancy of isolation valves.
	Decrease of primary coolant flow						
	Reactivity insertion by uncontrolled motion of segments of reflector at startup						
	Inner or outer tube failure of steam generator						
DBA	Failure of a cavity can	Single failure does not result in degradation and/or loss of the performance of the safety function because of channel redundancy and diversity.	Does not rely on the actuation of the detection signal. (N/A)				
	Sodium leakage from intermediate piping	N/A	Leak detection signal[1]			N/A	

Note:
1. Single failure does not result in degradation and/or loss of the performance of the safety signal because of channel redundancy.

Table 4.1.1-9
Release Fractions from Core into Cover Gas

	(a) Release Fraction - Core to Primary Sodium	(b) Release Fraction - Primary Sodium to Cover Gas Region	Net Release Fraction (Core to Cover Gas Region = (a)*(b))	LWR Case (PWR Core Inventory Fraction Released into Containment, RG 1.183)
Assumed Fuel Failure Fraction	1%	-	1%	-
Noble Gases	0.01	1.0	0.01	1.0
Halogens	0.01	$5x10^{-6}$	$5x10^{-8}$	0.4
Alkali Metals	0.01	$1x10^{-4}$	$1x10^{-6}$	0.3
Te Group	0.01	$3x10^{-6}$	$3x10^{-8}$	0.05
Ba, Sr	0.01	$3x10^{-6}$	$3x10^{-8}$	0.02
Noble Metals	0.001	$3x10^{-6}$	$3x10^{-9}$	0.0025
Ce Group	0.00001	$3x10^{-6}$	$3x10^{-9}$	0.0005
Lanthanides	0.00001	$3x10^{-6}$	$3x10^{-9}$	0.0002
Coolant	-	$3x10^{-6}$	-	-



IC : Inner Core

MC: Middle Core

OC: Outer Core

Figure 4.1.1-1 Cross-Section Shape and Area of the Core

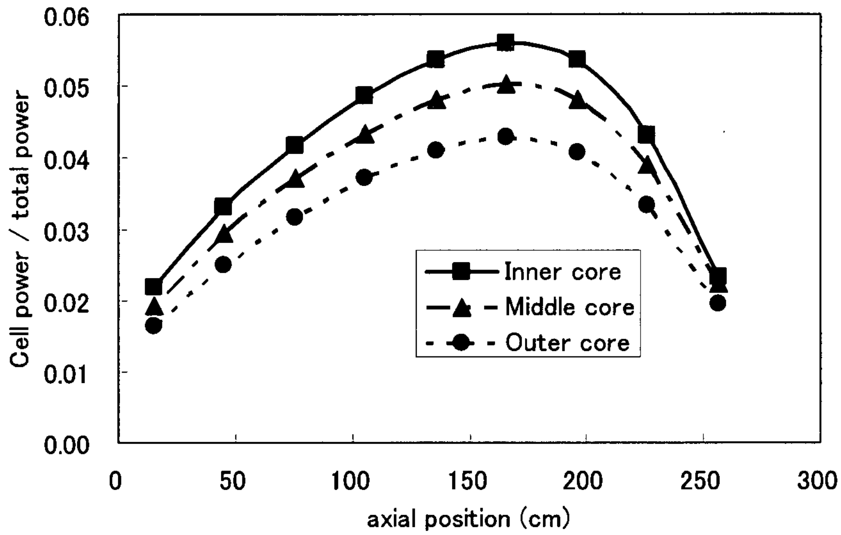


Figure 4.1.1-2 Axial Power Distribution

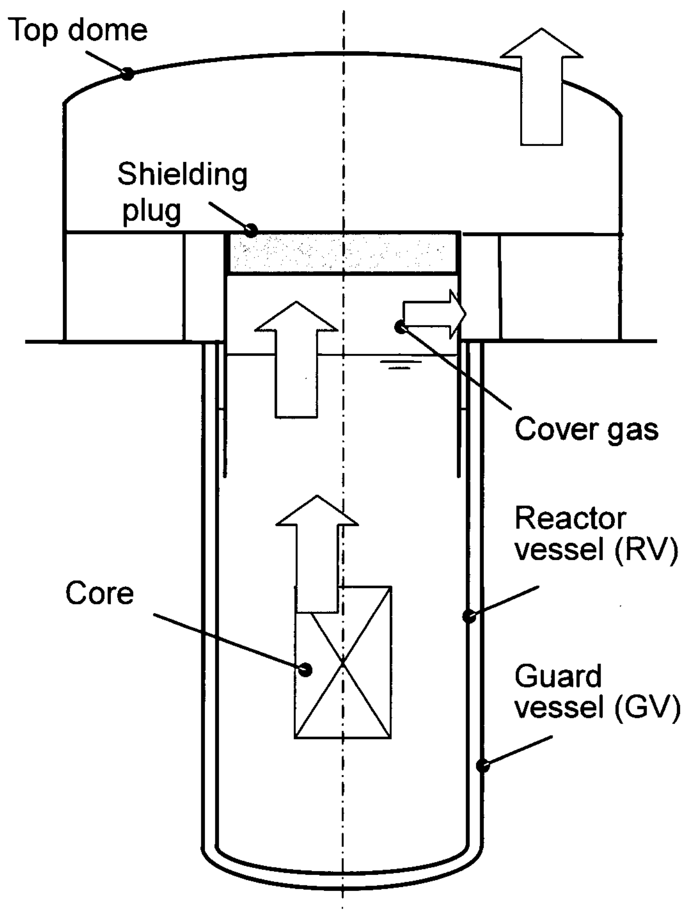


Figure 4.1.1-3 Leak Path for Source Term Analysis

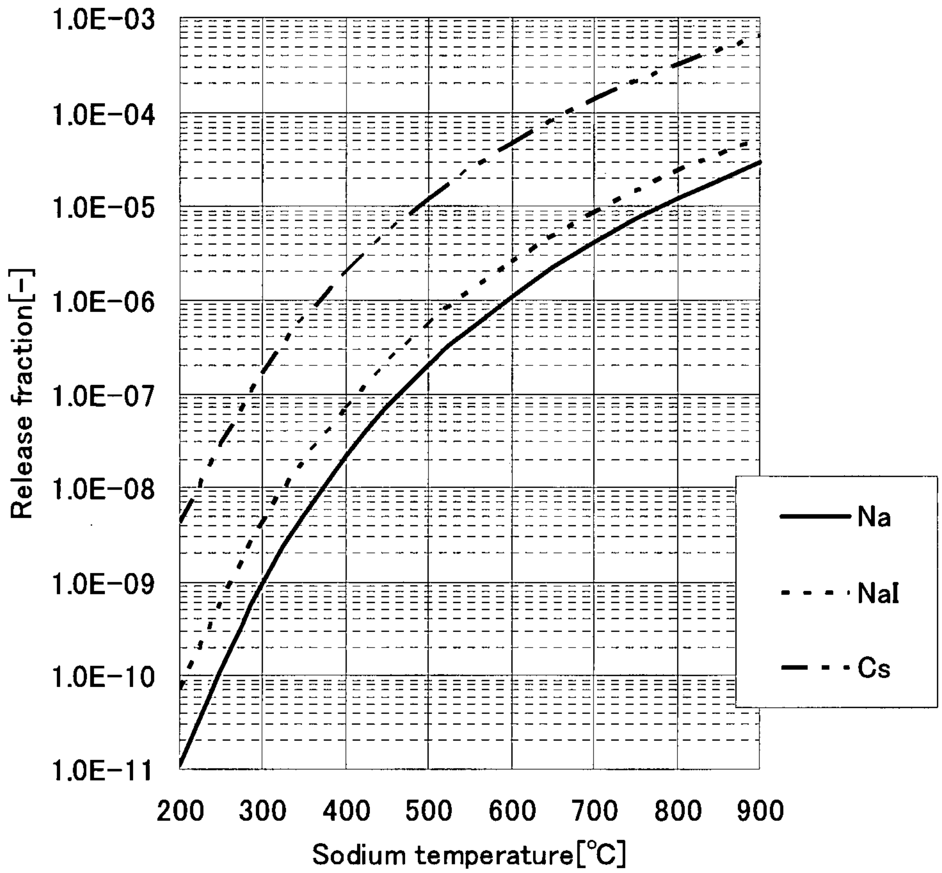


Figure 4.1.1-4 Release Fraction of Sodium, NaI, and Cs from Coolant to Cover Gas

4.1.2 Conditions Assumed for ATWS

The 4S reactor has two types of independent and diverse reactor shutdown systems, so ATWS has a very low occurrence frequency. It is considered a beyond-DBE, so the analytical conditions and evaluation procedures differ from those used for DBAs and AOOs. The basic analysis procedure for ATWS was derived from Ref. [36].

4.1.2.1 Items to be Considered in the Analysis

1. Time in Reactor Lifetime to be Considered during Analysis

   The entire 30-year operational term from completion of the performance tests to just before decommissioning the reactor is usually considered when analyzing the assumed event.

   A given transient calculation is continued until the reactor become the steady state.

2. Assumptions for the Safety Functions

   a. The safety functions of all safety-related systems, structures, and components, except scram functions, were considered in the analysis conditions. The functions of non-safety-related systems are not considered except for systems maintained at the same quality level as a safety system.

   b. A single failure is not assumed.

   c. Appropriate time is considered for performance of any operator actions required to cope with the event.

   d. The plant protection systems are assumed not to operate.

   e. Cases in which offsite power cannot be used are considered when anticipating the operation of engineered safety components.

4.1.2.2 Plant Conditions

1. Plant Heat Balance

   During analysis of ATWS, the rated power (100 percent), based on the design specifications, and the initial value of the flow rate in the cooling systems (primary, intermediate loop, and water/steam systems) is prescribed as the rated flow (100 percent).

4.1.2.3 Core Conditions

1. Core State

The EOL - at which internal pin pressure is the highest and cladding corrosion is maximized - presents the most severe conditions for the fuel cladding as compared to the acceptance criteria of the fuel cladding integrity described in Chapter 3. Thus, EOL is selected as the time in core life to be analyzed.

2. Power Distribution

The core is divided into three zones: inner core (IC), middle core (MC), and outer core (OC). Three types of the representative fuel subassemblies and fuel pins were modeled in the transient analyses.

The radial heat transportation between assemblies and the periphery of the core region is considered. The radial heat transfer paths are described below (refer to Figure 4.1.1-1):

- Between IC, MC, and OC
- Between IC and shutdown rod region
- Between MC, OC, and the gap between the reflector and core
- Between gap between the reflector and core and the reflector region

3. Reactivity Coefficients

a. Reactivity Feedbacks to be Considered

The analysis accounts for the reactivity changes due to Doppler effect, density change in the fuel, coolant, and structural material. The analysis also accounts for reactivity feedback due to:

- Thermal expansion of the core support structure and fuel assembly middle pad (pad-to-pad mechanical interactions due to thermal expansion leads to radial core expansion).

- Axial core reactivity distribution

- Reflector motion.

- Axial thermal expansion of the in-vessel structures (reactor vessel, reflector driveline, fuel subassembly element, and core support structures).

Each reactivity coefficient value for evaluating these reactivity changes was defined in consideration of the following description.

b. Regions in which the Reactivity Feedback Effects Occur

As shown in the Table 4.1.2-1, the coolant density coefficients in reflector and inside core barrel regions are positive. In case of temperature rise in these regions, negative reactivity is inserted. The reactivity in these regions is considered as a condition of reference case. On the other hand, the coolant density coefficient and structure density coefficient in the shutdown rod region are negative. In case of temperature rise in these regions, positive reactivity is inserted. The reactivity in these regions is considered also.

The regions in which the effect of reactivity is considered are as follows:

- Core
- Reactor shutdown rod and fixed absorber area in the center of the core
- Gap between the reflector and core
- Reflector

c. Uncertainty of Reactivity Coefficient

The reference reactivity coefficients such as density coefficient, geometry coefficient, core support expansion reactivity changes, radial core expansion reactivity changes, and axial core and reflector displacement reactivity changes used for the analysis are shown in Tables 4.1.2-1 for each region.

The uncertainty of reactivity changes is treated as follows.

The reactivity coefficients that have the potential for a large impact on safety characteristics are identified. Then, their sensitivity against the safety characteristics is analyzed. As a result through sensitivity analyses, the reactivity coefficients that have a large impact on safety characteristics are used for statistical analysis as a parameter with each uncertainty.

4. Fuel Pin Temperature Conditions

The conditions of the representative fuel pin, i.e., hottest pin, for the evaluation of the fuel cladding integrity are as described below.

The nominal hottest pin is selected as the representative fuel pin without considering the uncertainties of power, flow, and fabrication tolerance.

5. Decay Heat Conditions

The decay heat after 30 years of burnup was obtained from the nuclear calculations. During analysis of the ATWS, a design margin (for example, the shutdown period is different from the schedule) of +10 percent was added to the results of the nuclear calculations.

4.1.2.4 Plant Protection System Conditions

The plant protection systems are assumed not to operate.

4.1.2.5 Residual Heat Removal System Conditions

Modeling of the residual heat removal system:

1. The ATWS event proceeds without a reactor trip and AC can be started even with the temperature high signal. However, it was set without starting conservatively here.

2. RVACS is modeled so that the design heat removal value is accomplished at the specified reactor vessel temperature.

   The RVACS heat removal capacity for the transient is evaluated from the wall temperature of the reactor vessel and the natural convection air flow in the designed heat transfer configuration (reactor vessel, guard vessel, collector, and airflow path).

4.1.2.6 Single Failure Criterion

No single failure is assumed.

4.1.2.7 Definition of the Radiological Consequence Analysis

The radiological consequence analysis is not included in ATWS event.

4.1.2.8 Analysis Procedure for ATWS

The ATWS events were analyzed using reference conditions that were based on the described analysis conditions from subsection 4.1.2.1 to 4.1.2.7. Analysis results were compared with safety criteria described in the Section 3, and the validity of the 4S design was demonstrated. The analysis was performed in accordance with the following procedure:

1. Reference analysis using the reference data.

2. Identify the plausible phenomena using PIRT [37, 38].

3. Determine the important phenomena and parameters for sensitivity analysis.

4. Set the uncertainty width and perform sensitivity analysis.

5. Select dominant parameters.

6. Set the uncertainty width and distribution of the dominant parameters.

7. Perform a statistical evaluation [39].

8. Indicate that upper side of 95 percent probability at a 95 percent confidence level is within the acceptance criteria (see Section 3.4).

Table 4.1.2-1
Reactivity Coefficients to be Used for ATWS Analysis

-		Region			
Attribute	**Definition**	**Core**	**Shutdown Rod Region**	**Inside Core Barrel**	**Reflector**
Doppler Reactivity Coefficient	Tdk/dT	-3.8×10^{-3}	-	-	-
Fuel Density Coefficient	dk/kk'/dro/ro	3.6×10^{-1}	-	-	-
Coolant Density Coefficient	dk/kk'/dro/ro	2.8×10^{-4}	-1.2×10^{-3}	1.5×10^{-2}	3.3×10^{-3}
Structure Density Coefficient	dk/kk'/dro/ro	-1.5×10^{-2}	-4.9×10^{-4}	-	-
Core Support Structure Expansion	dk/kk'/°C	-5.9×10^{-6}			
Radial Core Expansion	dk/kk'/°C	-4.4×10^{-6}			
Axial Core and Reflector Displacement	dk/kk'/m(1)	-9.5×10^{-3}			

Note:
1. dk/kk'/m: reactivity coefficient to be used at the relative displacement of 1m reactor and reflector.

4.1.3 Summary of AOO, DBA, and ATWS

A summary of AOO, DBA, and ATWS analysis conditions is provided in Table 4.1.3-1.

Table 4.1.3-1
Analysis Conditions and Procedures for AOO, DBA, and ATWS

Attribute		Analysis Event: AOO and DBA	Analysis Event: ATWS
Plant Heat Balance	Power	102% of the rated value	100% of the rated value
	Primary Coolant Outlet/Inlet Temperature	510/355°C	
Core	State	EOL	
	Power Distribution	Core region is divided into 3 zones (IC, MC, and OC) and axial power distribution is set per each region.	
	Reactivity Coefficient	The uncertainty of the reactivity feedback from Doppler effect and the density change of each material (fuel, coolant, and structure) are considered.	• The reference value of reactivity feedback from Doppler effect and the density change of each material (fuel, coolant, and structure) are considered. • The reactivity feedback (from core support expansion, radial core expansion, and axial core and reflector displacement) is considered.
	Condition of the Fuel Pin	• The hottest pin (the uncertainties of power and flow distribution and fabrication tolerance are considered) is evaluated. The maximum cladding temperature before transient is 612 °C. • Maximum linear heat generation: 97 W/cm.	• The nominal hottest pin (w/o consideration of the uncertainties) is evaluated. The maximum cladding temperature before transient is 570 °C. • Maximum linear heat generation: 90 W/cm.
	Decay Heat Condition	• 120% of the nuclear calculation result value.	• 110% of the nuclear calculation result value.
Safety System	Set Value and Response Time	Ref. Table 4.1.1-5.	N/A
	Reactivity Insertion at Scram	Reactivity value under the condition of one stuck reflector segment.	
	Delay Time of Pump Trip	0s	

<table>
<tr><th colspan="4">Table 4.1.3-1
Analysis Conditions and Procedures for AOO, DBA, and ATWS (cont.)</th></tr>
<tr><th colspan="2" rowspan="2">Attribute</th><th colspan="2">Analysis Event</th></tr>
<tr><th>AOO and DBA</th><th>ATWS</th></tr>
<tr><td rowspan="3">Decay Heat Removal System</td><td>Steam Generator System</td><td>Insulated at transient initiation.</td><td>Insulated at transient initiation.</td></tr>
<tr><td>IRACS</td><td>• Flow coastdown of the intermediate pump is not considered.
• Natural circulation of the intermediate-loop system and air is considered.</td><td>• Not started.</td></tr>
<tr><td>RVACS</td><td colspan="2">Same condition as rated operation.</td></tr>
<tr><td colspan="2">Single Failure Condition</td><td>Single failure of active components is assumed.</td><td>Single failure not assumed.</td></tr>
<tr><td colspan="2">Radiation Exposure</td><td>Reflecting the characteristics of the 4S reactor, conditions of the source term and environmental release are set.</td><td>N/A</td></tr>
<tr><td colspan="2">Additional Conditions</td><td>None.</td><td>• All functions of the safety-related component are considered.
• The functions of the non-safety-related components are not considered unless maintained with the same quality level of safety system.
• The radial heat transportation in the core and the periphery of the core region are considered.</td></tr>
<tr><td colspan="2">Analysis Procedure</td><td>Analyses under above conditions are performed.</td><td>Analyses under reference condition and uncertainty analysis (95/95)[1] are performed.</td></tr>
</table>

[1] Upper side of 95 percent probability at a 95 percent confidence level

4.2 Analysis Method

The list of the analyzed events described in Chapter 4, codes used for the analyses, and the procedure is shown in Table 4.2.1-1. The summary of the analysis codes and each procedure are as follows.

4.2.1 Plant Dynamics Analysis Code (ARGO)

The purpose of the analysis using the ARGO code [40-43] is to calculate the physical quantities that show the integrity of the cladding and primary coolant boundary at transient state and/or accident conditions.

The analysis functions that the ARGO code includes are as follows. The detail of the functions, construction of the code, and the system configuration diagram are shown in Appendix A.

- Plant thermal dynamics analysis
- Variation of reactor power and decay heat
- Variation of the reactor fuel pin temperature
- The calculation formula for CDF of the cladding
- Variation of the primary coolant boundary temperature
- Thermal-hydraulic characteristics of the intermediate heat exchanger and the steam generator
- Hydraulic characteristics of the circulation pump
- Heat removal by decay heat removal systems (e.g., IRACS, RVACS)
- Actuation of the safety and interlock systems

The ARGO code is currently undergoing V&V following RG 1.203 [44] and NUREG-1737 [45]. The document summarizing the results of the validity of the analysis results for the ARGO code will be submitted as part of the design approval application.

4.2.2 Method of Radiation Exposure Analysis

The following metrics are calculated to show compliance with the criteria for allowable radiation exposure at the site boundary.

1. EAB, 2 hours, total equivalent dose equivalent[1]

2. LPZ boundary, 30 day, total equivalent dose equivalent[1]

3. Although 10 CFR 50.34 and 10 CFR 52.47 set the evaluation period of dose at the LPZ boundary as "during the entire period of its passage," referring to the RG 4.7 [46], the time period for evaluation of dose at the LPZ boundary is set in this report as the postulated accident duration of 30 days.

External dose equivalent (EDE) and committed effective dose equivalent (CEDE) are evaluated according to the time characteristics of the radioactive release and transport of each radionuclide and the coefficient of the radiation exposure analysis, e.g., dose conversion coefficient, atmospheric dispersion factor, absorption rate.

The details of the procedure and the system configuration diagram are shown in Appendix A.2.

1. TEDE= Effective dose equivalent (EDE; External absorbed dose) + committed effective dose equivalent (CEDE; Internal absorbed dose)

Table 4.2.1-1
Events Selected for Analysis, Analysis Code, and Analysis Method

Event		Phenomena	Analysis Code and Method
AOO	Loss of Offsite Power	• Thermal-hydraulics	• ARGO
	Decrease of Primary Coolant Flow	• Thermal-hydraulics	• ARGO
	Reactivity Insertion by Uncontrolled Motion of Segments of Reflector at Startup	• Thermal-hydraulics	• ARGO
	Inner or Outer Tube Failure of Steam Generator	• Thermal-hydraulics	• ARGO
DBA	Failure of a Cavity Can	• Thermal-hydraulics • Radiological consequences	• ARGO • Theoretical analysis
	Sodium Leakage from Intermediate Piping	• Thermal-hydraulics • Radiological consequences	• ARGO • Theoretical analysis
	Primary Cover Gas Boundary Failure	• Radiological consequences	• Theoretical analysis
ATWS	Loss of Offsite Power without Scram	• Thermal-hydraulics	• ARGO

5 REPRESENTATIVE ANALYSIS RESULTS

This section presents representative analysis cases that cover the spectrum of design basis events and beyond DBEs. The DBEs include AOO and postulated design basis accidents. For these DBEs, the analysis used conservative input parameters and assumed a single failure in the systems responding to the event. Beyond DBEs were analyzed using reference values. In all cases, the analysis led to the following findings:

- Insignificant impact on the fuel clad integrity or the fuel 30-year design life as measured by the CDF
- Insignificant impact on the primary coolant boundary as measured by the primary sodium temperature and intact core
- Insignificant radionuclide release

5.1 Anticipated Operational Occurrences (AOOs)

The following four events have been selected (from those listed in Table 3.3-1) as the representative (most severe) events for the 4S transients in light of the safety criteria described in Chapter 3:

- Loss of offsite power (Section 5.1.1)
- Decrease of primary coolant flow (Section 5.1.2)
- Reactivity insertion by uncontrolled motion of segments of reflector at startup (Section 5.1.3)
- Inner or outer tube failure of the SG (Section 5.1.4)

These events are described further in the following sections.

5.1.1 Loss of Offsite Power (LOSP)

5.1.1.1 Identification of Causes and Frequency Classification

This event may result from the whole or partial loss of the station power supply due to failure in the transmission system or failure of the station electrical facilities during the power operation of the reactor. The event is assumed to occur one or more times during the reactor lifetime. Consequently, this event is classified as an AOO.

5.1.1.2 Sequence of Events and System Performance

The loss of offsite power leads to simultaneous trip of the primary circulation pumps, the intermediate-loop circulation pump, and the feedwater pump. The power supply for the primary and intermediate-loop circulation pumps is switched to the coastdown power supply from the individually independent motor-generator (MG) sets. The flow rates of the primary and intermediate coolant will then coast down in response to the reduction of the circulation pump head.

A reactor scram is caused by a low signal in the normal bus voltage signal of the reactor protection system. With the occurrence of a scram signal, the segments of reflector lower, the reactor shutdown rod is inserted, and the reactor power drops rapidly. A signal of the reactor protection system also triggers the IRACS residual heat removal startup, whereby the air-side damper of the AC installed in the intermediate loop is opened and residual heat removal commences. The residual heat of the reactor core is removed by natural circulation in both the primary and intermediate-loop coolant, using both the IRACS and the RVACS. The blower associated with the AC and the intermediate-loop circulation pump is then started up by the emergency power supply shortly after the occurrence of the event. This results in the forced circulation of both air flowing into the AC and intermediate-loop coolant, rapidly cooling the reactor to the cold standby state.

5.1.1.3 Core and System Performance

1. Evaluation Model

   The reactor core and system response is analyzed using the ARGO code (see Section 4.2).

   The main evaluation model characteristics are as follows:

   - The nuclear dynamic characteristics are evaluated by a one-point kinetic model.

   - The reactivity feedbacks from both the Doppler effect and the density change effect (fuel, coolant, structure) are considered in the calculation. The axial and radial distributions of reactivity feedback are also considered.

   - The coolant flow in the core and the primary and intermediate coolant systems are simulated by using a flow network model. The flow network is modeled by using a one-dimensional dynamic equation that accounts for pump head, natural circulation head, and pressure loss of the system.

   - With respect to the intermediate heat exchanger, heat transfer is modeled between the primary-side fluid and a heat transfer tube, and between the heat transfer tube and secondary-side fluid.

- With respect to the steam generator, the heat transfer from the intermediate-loop coolant to the steam system is considered.

- The detection system is modeled so that a signal can be transmitted at a predetermined delay time when the detected variable exceeds or falls below the predetermined trip setpoint. Transmitting the signal results in actuation of the reactor protection system and pump trip at their specified delay times.

2. Input Parameters and Initial Conditions

The main initial conditions are as follows:

- The initial power is defined as 102 percent of the rated power.

- For the inlet and outlet temperature of primary and intermediate coolant, rated temperature is set for each as an initial condition.

- The representative pin for evaluating fuel cladding integrity is the hottest pin. The cladding temperature of the hottest pin is the maximum temperature accounting for the temperature uncertainty assigned to the peak temperature.

3. Analysis Conditions

The conditions of system response and single failure are as follows:

- The intermediate-loop flow coastdown system is not credited because the coastdown system of the intermediate pump is not a safety system.

- The steam/water system is assumed to undergo instantaneous loss of the heat removal function following the event.

- The single failure of the reactor scram system is assumed as one reflector is stuck and the single failure of IRACS is the failure to open either one of the two intakes or two exhaust dampers.

4. Analysis Results (Response Behavior)

The sequence and start time of each phenomenon and action are shown in Table 5.1.1-1 for the analytical results.

The reactor power and primary flow rate are shown in Figure 5.1.1-1. The fuel temperature of the hottest pin and the change in the cladding temperature are shown in Figure 5.1.1-2. The changes to CDF and peak temperature at the core outlet are shown in Figures 5.1.1-3 and 5.1.1-4, respectively.

When the event commences, the primary circulation pumps, intermediate-loop circulation pump, and feedwater pump are tripped. The bus voltage low signal is transmitted, resulting in a scram signal at 1.0 second, and the reflector starts to descend at 1.5 seconds. The relative value of the reactor power exceeds the relative value of the primary coolant flow rate momentarily until the reflector begins to descend. However, this imbalance is slight and the core temperature is hardly changed due to the extremely short time involved and the low power density of the core. The fuel peak temperature and the fuel cladding peak temperature of the hottest pin do not exceed the initial temperatures of 639°C and 613°C, respectively.

After the segments of reflector descend, the relative flow rate becomes larger than the relative power rate; hence, the core temperature decreases (see Figures 5.1.1-1 and Figure 5.1.1-2). At 60 seconds, the power generated by the MG set is lost and the primary coolant flow makes the transition to natural circulation. Subsequently, both the primary and intermediate coolant flows are by natural circulation, and the residual heat is removed by the RVACS and IRACS (see Figure 5.1.1-4).

The fuel peak temperature of the hottest pin is 639°C, meaning there is a sufficient margin to the fuel melting point, thus meeting the "no fuel melting" safety acceptance criterion (see Section 3.4). The CDF value of the hottest pin is 5.7×10^{-9}, which means there is sufficient margin for the safety acceptance criteria specified by the inequality $\Sigma CDF < 0.1$, with multiple occurrences considered. Therefore, the 4S design complies with the safety acceptance criterion of cladding integrity.

The core outlet coolant temperature does not exceed the temperature for rated power; therefore, the 4S design maintains the integrity of the coolant boundary with sufficient margin.

Table 5.1.1-1 Sequence of Events for LOSP	
Time (s)	**Events**
0.0	Loss of the power supply for the station
0.0	Trip of the primary- and intermediate-loop and feedwater pumps
0.0	Switch of status of the primary pumps from normal operation to flow coastdown
1.0	Transmittal of the signals: Bus voltage "Low" signal Scram initiation signal Residual heat removal system initiation signal
1.5	Start lowering of the reflector
2.0	Start opening of the AC damper
9.5	Finish lowering of the reflector
32.0	Finish opening of the AC damper
60.0	Finish of the flow coastdown state of the primary pumps, start of natural circulation state of the primary coolant flow

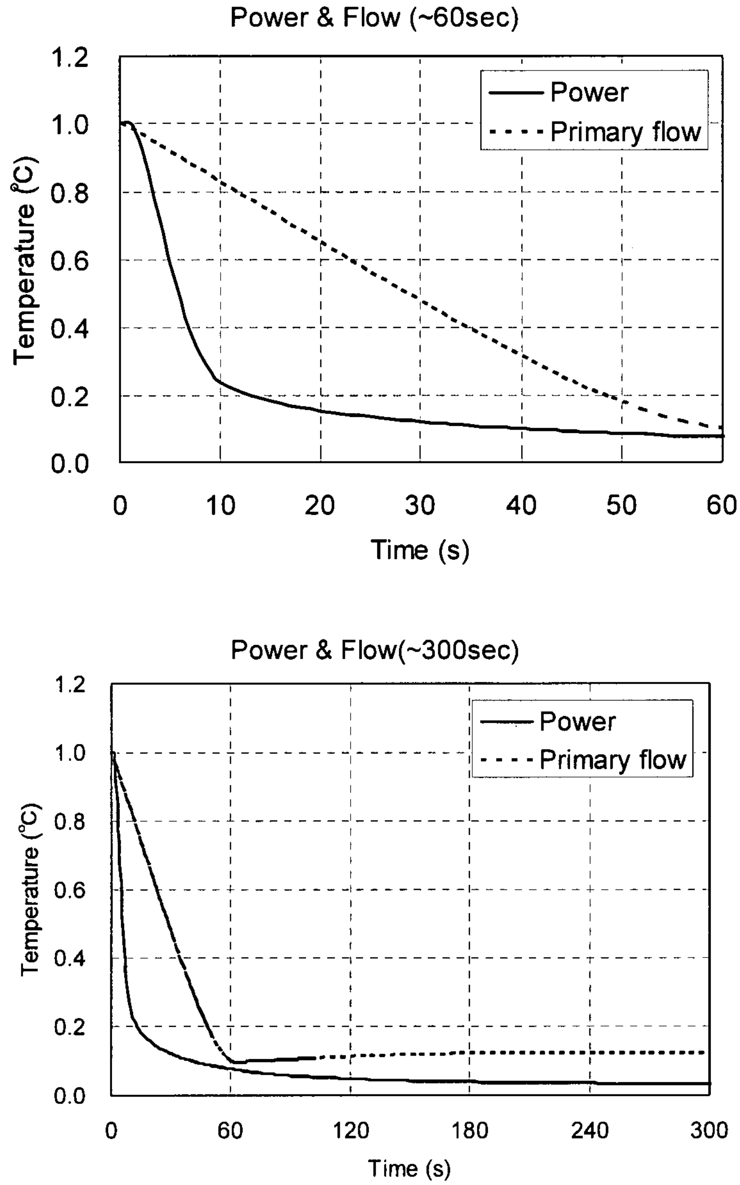


Figure 5.1.1-1 Power and Primary Flow (LOSP)

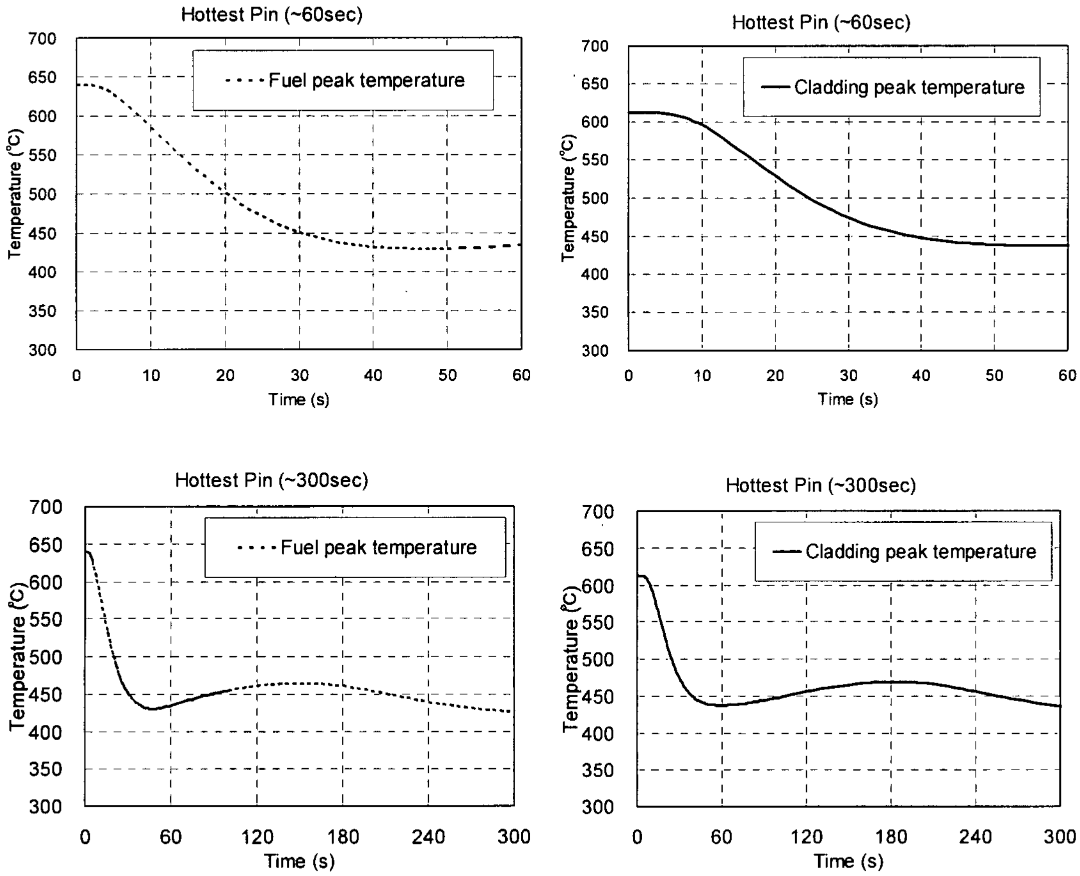
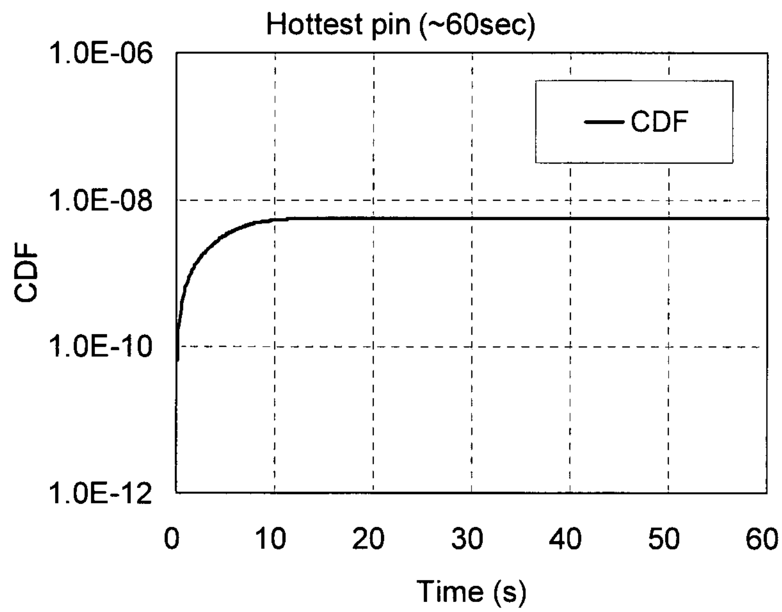


Figure 5.1.1-2 Fuel and Cladding Peak Temperature (LOSP)



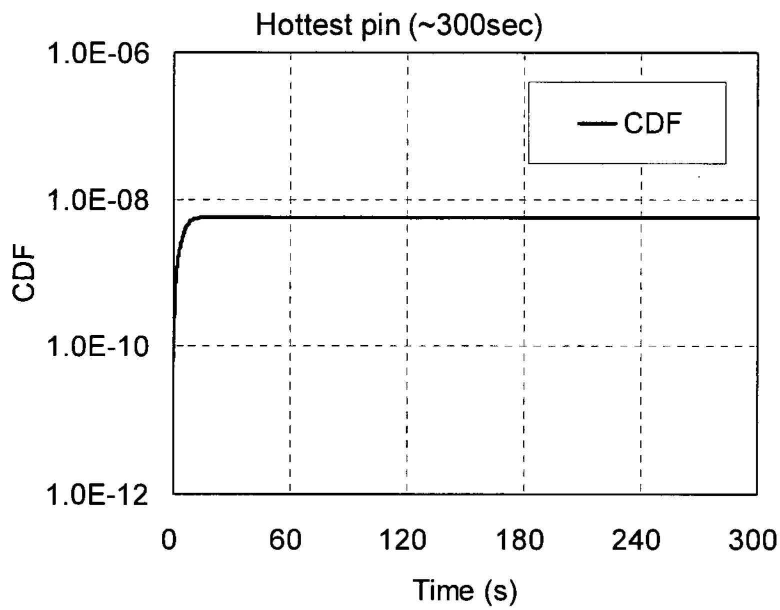


Figure 5.1.1-3 Cumulative Damage Fraction (LOSP)

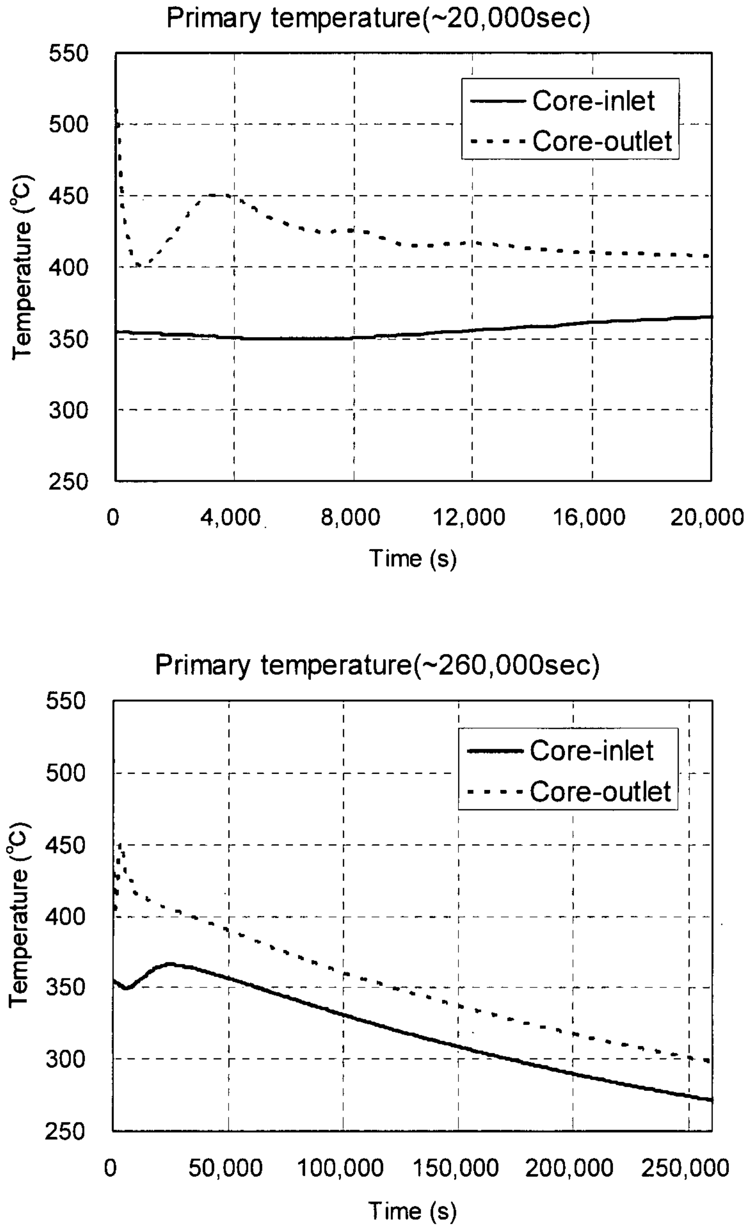


Figure 5.1.1-4 Core Inlet and Outlet Temperature (LOSP)

5.1.2 Decrease of Primary Coolant Flow

5.1.2.1 Identification of Causes and Frequency Classification

A decrease of primary coolant flow occurs due to a postulated power converter failure of one of the two primary circulation electromagnetic pumps. The postulated failure causes the pump head loss during rated power operation. This event is classified as LOF event.

The occurrence frequency of the decrease of primary coolant flow event is assumed to be once or more during the life of the reactor. Consequently this event is classified as an AOO.

5.1.2.2 Sequence of Events and System Performance

When either of the primary EM pumps fails at rated operation, the flow rate of the primary coolant decreases. The other pump of the two primary EM pumps trips due to generation of a pump protection signal and/or influence of the failed pump. A low voltage/current signal from the failed EM pump is transmitted to the plant protection system, whereupon reactor scram is initiated. As a result, the shutdown rod is inserted, the reflector lowers, and the reactor power rapidly decreases.

The scram signal causes the other primary circulation EM pump and the intermediate-loop EM pump to trip, whereupon the power supply for the primary and intermediate-loop circulation pumps switches to the power generated by the MG set. The flow rates of the primary and intermediate-loop coolant coastdown in response to the change in the circulation pump head.

The flow rates of the primary- and intermediate-loop coolant coast down before making the transition to natural circulation. A signal from the reactor protection system also initiates residual heat removal startup, whereupon the air-side damper of the AC is opened and IRACS residual heat removal commences. The remaining event behavior is the same as the loss of offsite power described in Section 5.1.1.

5.1.2.3 Core and System Performance

1. Evaluation Model

   The core and the system response are analyzed by using the ARGO code (see Section 4.2). The evaluation model characteristics are the same as in the loss of offsite power described in subsection 5.1.1.3(1).

2. Input Parameters and Initial Conditions

   The major initial conditions are the same as for the loss of offsite power event described in subsection 5.1.1.3(2).

3. Analysis Conditions

The conditions of the system response and single failure are the same as in the loss of offsite power event described in subsection 5.1.1.3(3) with one exception.

The difference is as follows:

- The steam/water system is assumed to undergo the instantaneous loss of the heat removal function due to loss of the feedwater pump by the plant protection system signal.

4. Analysis Results (Response Behavior)

The sequence and start time of each phenomenon and action are shown in Table 5.1.2-1.

The change in reactor power and the change in primary flow rate are shown in Figure 5.1.2.-1. The fuel temperature of the hottest pin and the change in cladding temperature are shown in Figure 5.1.2-2. The change in CDF is shown in Figure 5.1.2-3. The temperature change at the core outlet is shown in Figures 5.1.2-4 from the analytical results using the ARGO code.

After 6.6 seconds have elapsed, the low voltage/current trip signal of the primary EM pump is initiated and, at 8.1 seconds, the shutdown rod begins to insert. The relative value of the reactor power exceeds the relative value of the primary coolant flow rate momentarily until the start of shutdown rod descent; however, this rise is slight and the core temperature is hardly changed due to the extremely short time involved and the low power density of the core. At this time, the cladding and fuel temperatures peak, but the fuel peak temperature and the cladding peak temperature of hottest pin do not exceed 644°C and 621°C, respectively.

After the shutdown rod descends, the flow rate decrease exceeds the power decrease; hence, the reactor core temperature decreases (see Figure 5.1.2-1, Figure 5.1.2-2). At 60 seconds, the pump head of the primary pump coasting down by the power generated at the MG set is lost and the flow makes the transition to natural circulation. Subsequently, both the flow rates of the primary- and intermediate-loop coolant are in a state of natural circulation and residual heat is removed by the RVACS and IRACS (Figure 5.1.2-4).

The fuel peak temperature of hottest pin is 644°C, meaning there is sufficient margin to the fuel melting point, thus satisfying the "no fuel melting" safety acceptance criterion. The CDF value of the hottest pin is 1.7×10^{-8}, which means there is sufficient margin for the safety acceptance criterion indicated by the inequality $\Sigma CDF < 0.1$, for any conceivable frequency of occurrence.

The 4S design complies with the safety acceptance criteria for cladding integrity.

The core outlet coolant temperature does not exceed the temperature for the rated power; therefore, the 4S design maintains the integrity of the coolant boundary with sufficient margin.

According to the results, the design of the 4S reactor complies with the safety acceptance criteria for the integrity of the fuel, the integrity of the cladding and the integrity of the coolant boundary with sufficient margin.

Table 5.1.2-1
Sequence of Events for LOF

Time (s)	Events
0.0	Trip of one primary pump due to failure and trip of other primary pump due to generation of a pump protection signal and/or influence of failed pump
6.6	Transmittal of the signals: EM pumps voltage/current low signal Scram initiation signal Residual heat removal system initiation signal
7.6	Trip of the other primary- and intermediate-loop and feedwater pumps
8.1	Start inserting of the shutdown rod
8.6	Start opening of the AC damper
16.1	Finish inserting of the shutdown rod
38.6	Finish opening of the AC damper
60.0	Finish of the flow coastdown state of the primary pump, start of the natural circulation state of the primary coolant flow

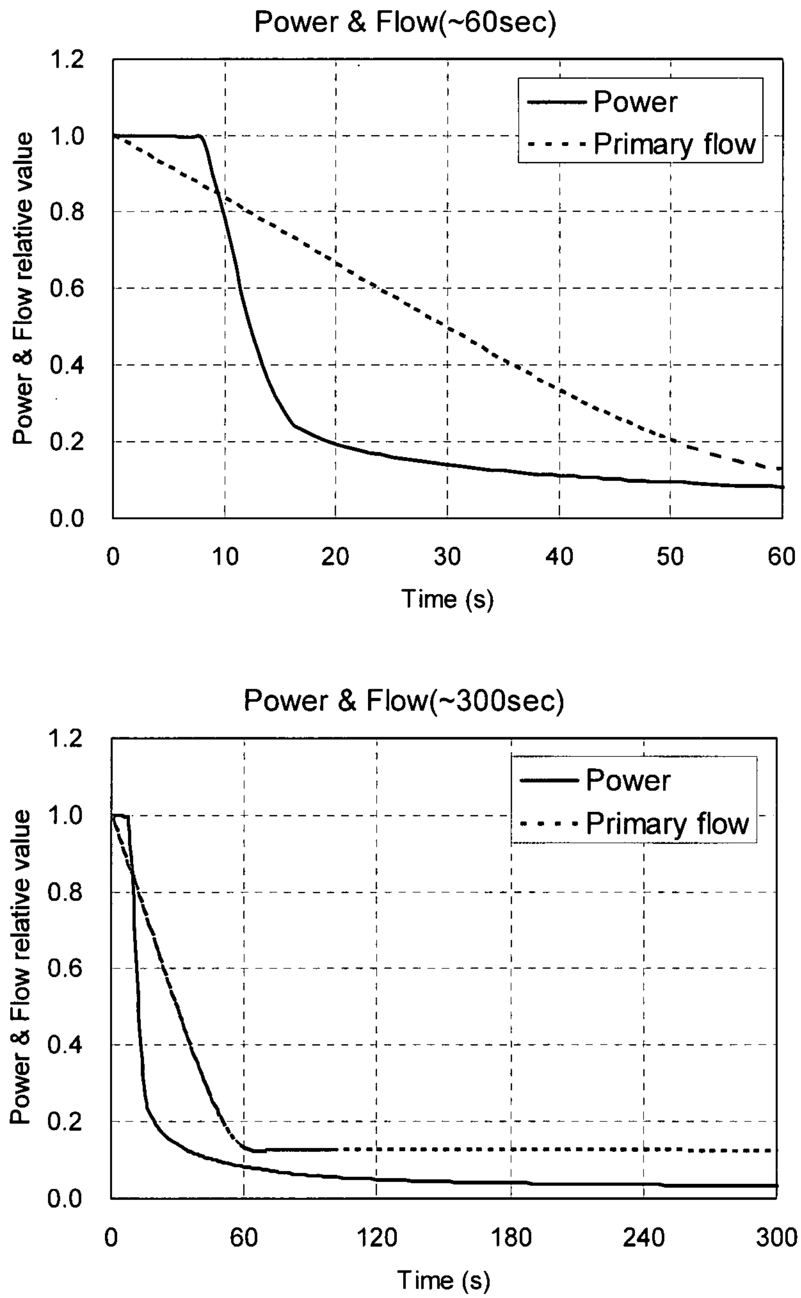


Figure 5.1.2-1 Power and Primary Flow (LOF)

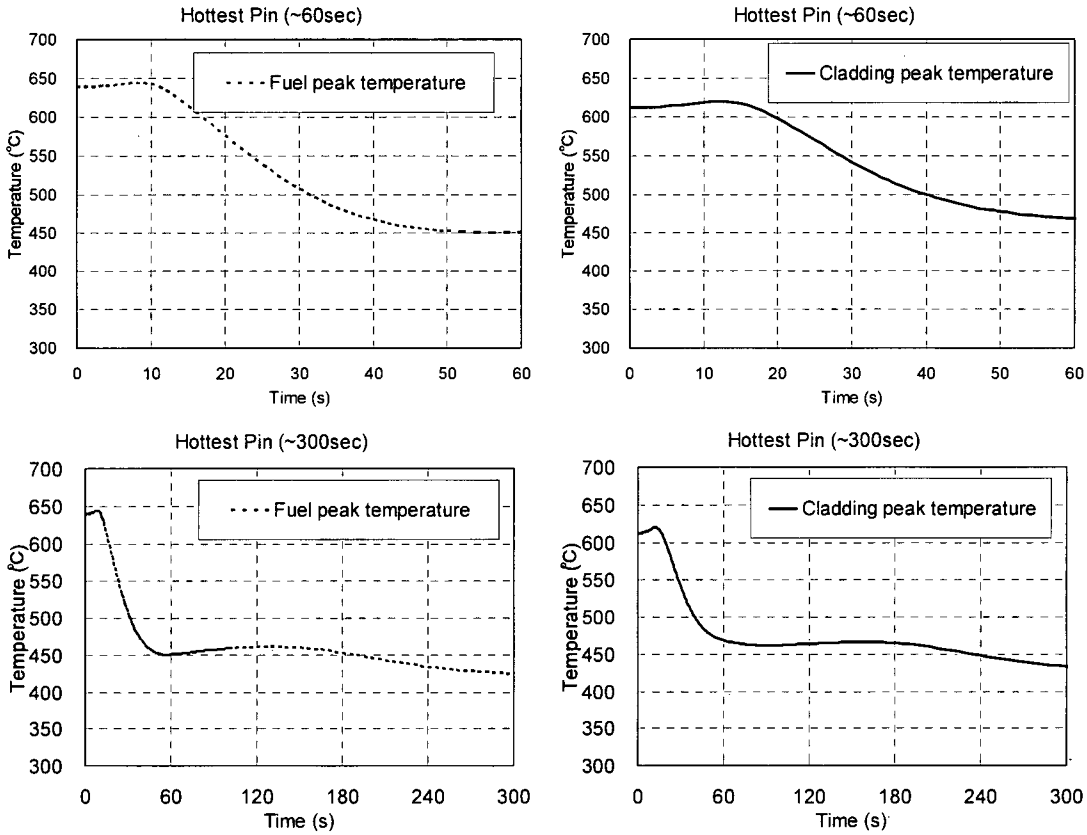


Figure 5.1.2-2 Fuel and Cladding Peak Temperature (LOF)

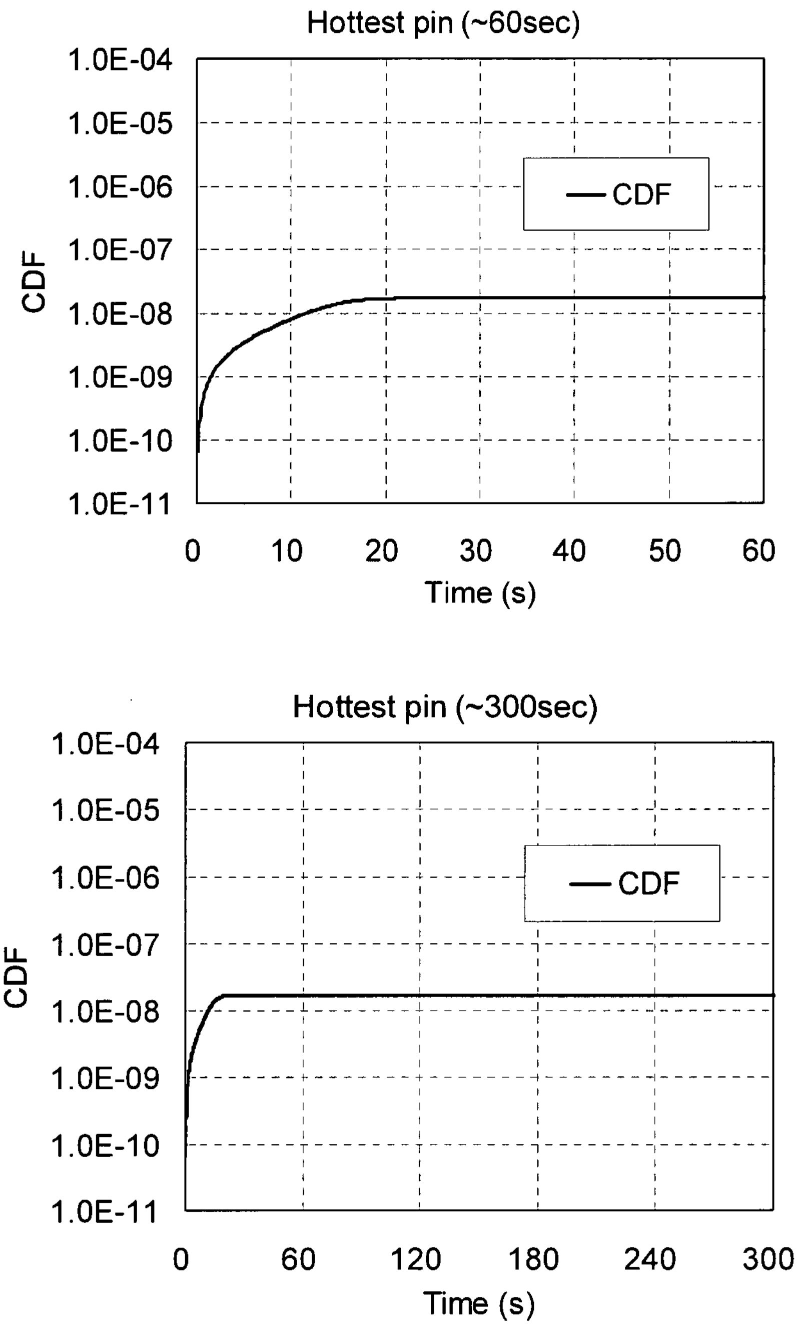


Figure 5.1.2-3 Cumulative Damage Fraction (LOF)

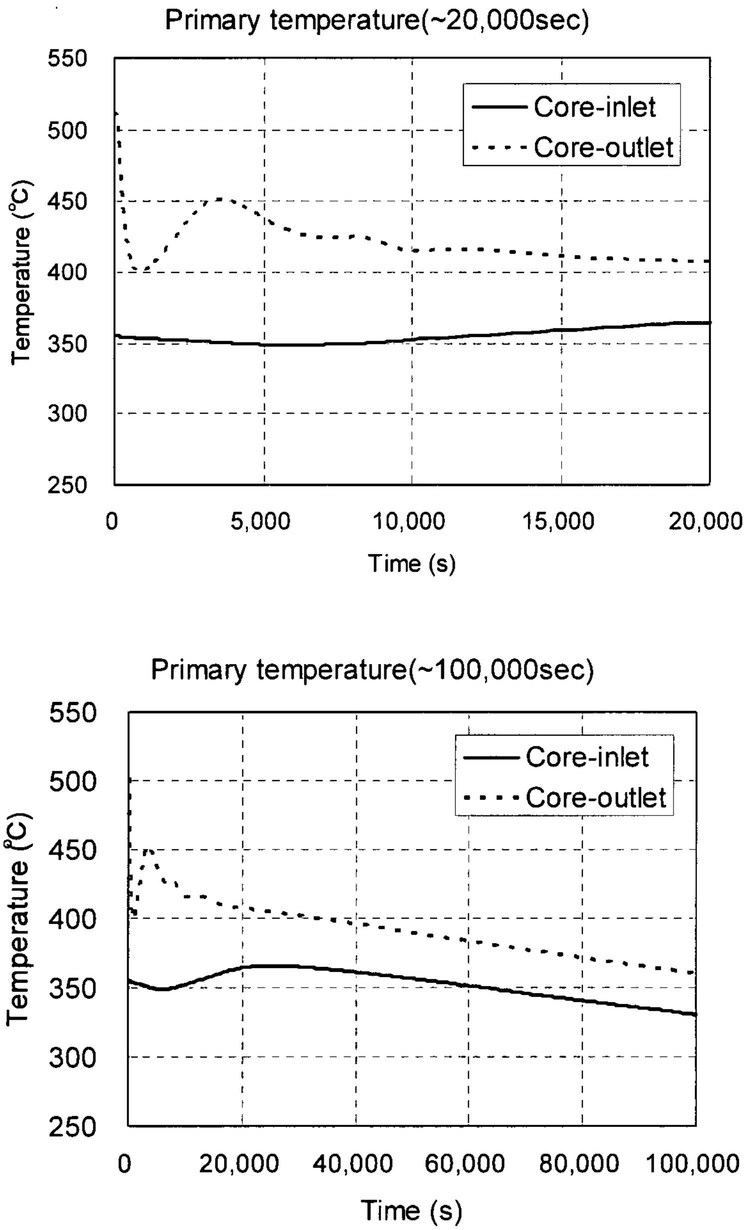


Figure 5.1.2-4 Core Inlet and Outlet Temperature (LOF)

5.1.3 Reactivity Insertion by Uncontrolled Motion of Segments of Reflector at Startup (RIR)

5.1.3.1 Identification Causes and Frequency Classification

The event results from uncontrolled motion of reflector drive system on reactor startup cause insertion of positive reactivity into the core, and the reactor power will eventually increase.

The event assumed is one that could potentially occur when the reactor power is increased from partial power (30 percent) to rated power (100 percent). When an attempt is made to increase the reactor power from partial power (30 percent) to rated power (100 percent), the reflector is raised, whereupon the reactor power is increased simultaneously with the flow rate of the coolant by increasing the discharge pressure of the EM pumps, meaning the reactor power and flow rate are respectively increased at a ratio of approximately 1:1 [13]. When the power increase rate exceeds the flow increase rate due to loss of reflector drive system control, the system temperature rises.

The occurrence frequency of this event is assumed to be once or more during the life of the reactor. Consequently, this event is classified as an AOO.

5.1.3.2 Sequence of Events and System Performance

When the increase in flow rate does not match the increase in reactor power during power-up of reactor power from 30 to 100 percent, the power-to-flow mismatch results in initiation of a high-power trip signal. This causes the primary- and intermediate-loop circulation EM pumps and reflectors to trip and the insertion of the reactor shutdown rod. The remaining event behavior is the same as that described in Section 5.1.1, loss of offsite power.

5.1.3.3 Core and System Performance

1. Evaluation Model

   Analysis code: ARGO

   Evaluation model: Same as described in subsection 5.1.1.3(1)

2. Input Parameters and Initial Conditions

   - The initial power is defined as 30.6 percent of the rated power.
   - Set temperature of inlet/outlet of primary and intermediate system by the heat balance under the status of operation with power at 30 percent.
   - The representative pin for evaluating fuel cladding integrity is the hottest pin.

- The hottest temperature at partial power is calculated by applying the conservative condition with rated power. The cladding temperature of the hottest pin is the maximum temperature accounting for the temperature uncertainty assigned to the temperature.

- The reactivity insertion rate is 0.0070 cent/s (this insertion rate can make reactor power 100 percent in 3 hours of normal operation).

3. Analysis Conditions

The conditions of the system response and single failure are the same as described in subsection 5.1.1.3(3) with one exception:

- The steam/water system is assumed to undergo the instantaneous loss of the heat removal function after the occurrence of the scram.

4. Analysis Results (Response Behavior)

The sequence and start time of each phenomenon and action is shown in Table 5.1.3-1 for the analytical results.

The change in reactor power and the change in primary flow rate are shown in Figures 5.1.3-1. The change of fuel temperature of the hottest pin and the change of cladding temperature are shown in Figure 5.1.3-2. The change in CDF is shown in Figure 5.1.3-3. The temperature change at the core outlet and inlet is shown in Figures 5.1.3-4 from the analytical results using the ARGO code.

After 514 seconds from the start of power runup, various event profiles including that the high neutron flux trip signal is transmitted and the reflector started to lower with 1.0-second delay were shown. The relative value of the reactor power exceeds the relative value of the primary coolant flow rate until the reflector starts to lower, which causes the core temperature to increase. The fuel and cladding temperatures respectively peak just before the scram, but since the initial temperature is low, the fuel peak temperature and the cladding peak temperature are not more than 613°C and 607°C, respectively.

Subsequently, both the flow rates of the primary and intermediate-loop coolant are in a state of natural circulation and then the residual heat is removed by the RVACS and IRACS (see Figure 5.1.3-4.).

With respect to the integrity of the fuel, the fuel peak temperature is 613°C, meaning there is sufficient margin to the fuel melting point specified as the safety acceptance criterion. With respect to the integrity of the cladding, the CDF value of the evaluated pin is 6.9×10^{-8} whereupon there is sufficient margin to the safety acceptance criteria indicated by the inequality ΣCDF < 0.1, even taking into in consideration the possibility of multiple the occurrences.

With respect to integrity of the coolant boundary, the core outlet coolant temperature does not exceed the temperature for rated power; therefore, the 4S design maintains the integrity of the coolant boundary with sufficient margin to the criteria.

According to these results, the design of the 4S reactor complies with the safety acceptance criteria for the integrity of the fuel, the integrity of the cladding, and the integrity of the coolant boundary with sufficient margin to the criteria.

Table 5.1.3-1
Sequence of Events for RIR

Time (s)	Events
0	Abnormal withdrawal of the reflector
514	Transmittal of the signals: Neutron flux level "High" signal Scram initiation signal Residual heat removal system initiation signal
515	Trip of the primary- and intermediate-loop and feedwater pumps
515	Start lowering of the reflector
516	Start opening of the AC damper
523	Finish insertion of the reflector
546	Finish opening of the AC damper

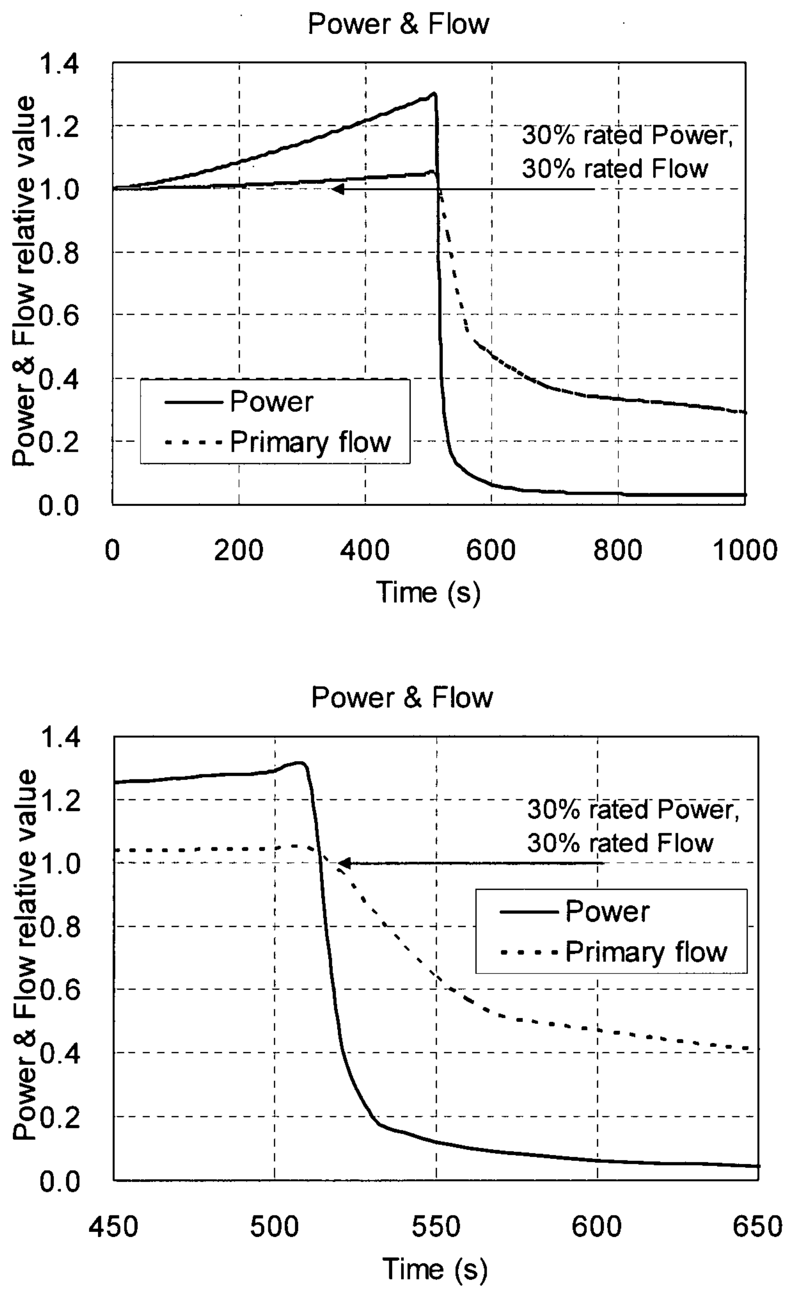


Figure 5.1.3-1 Power and Primary Flow (RIR)

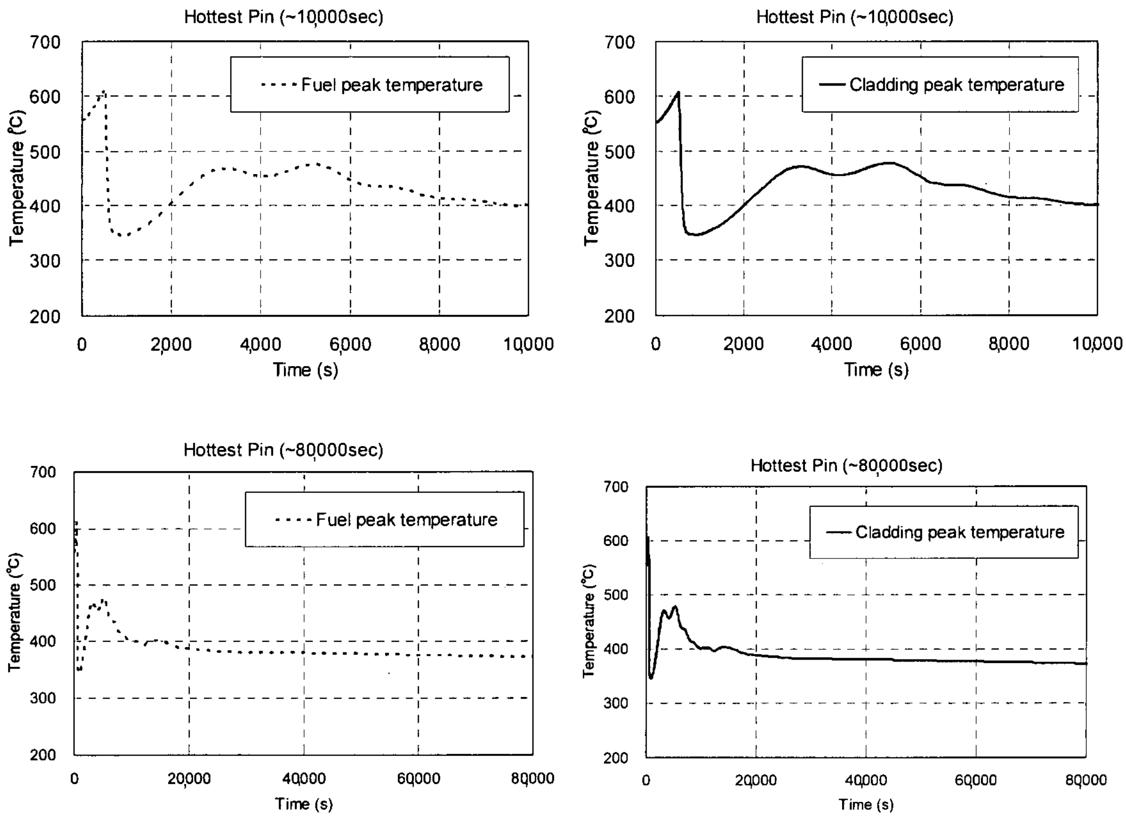
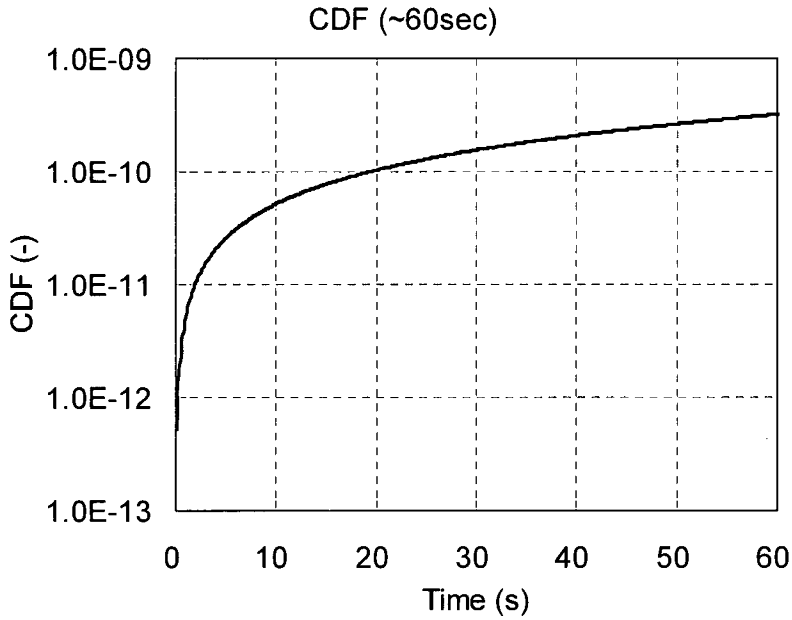


Figure 5.1.3-2 Fuel and Cladding Peak Temperature (RIR)



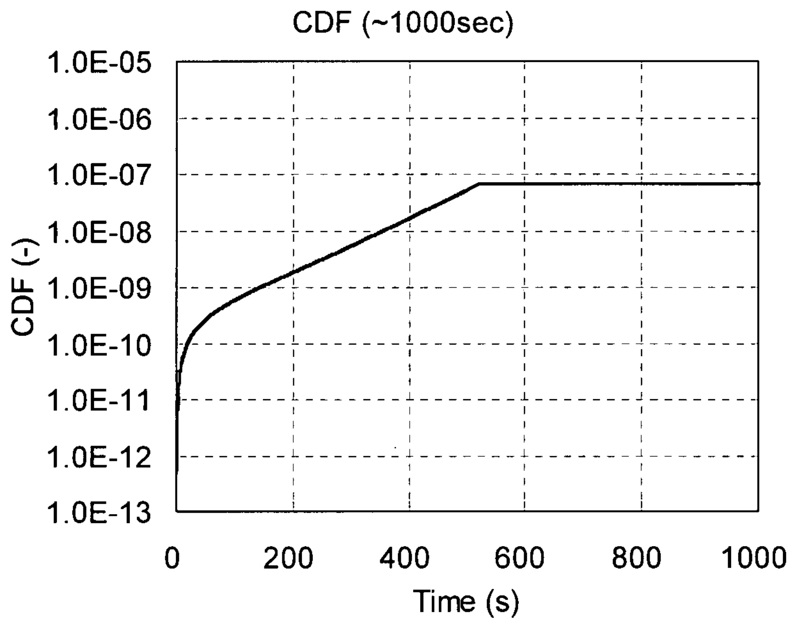
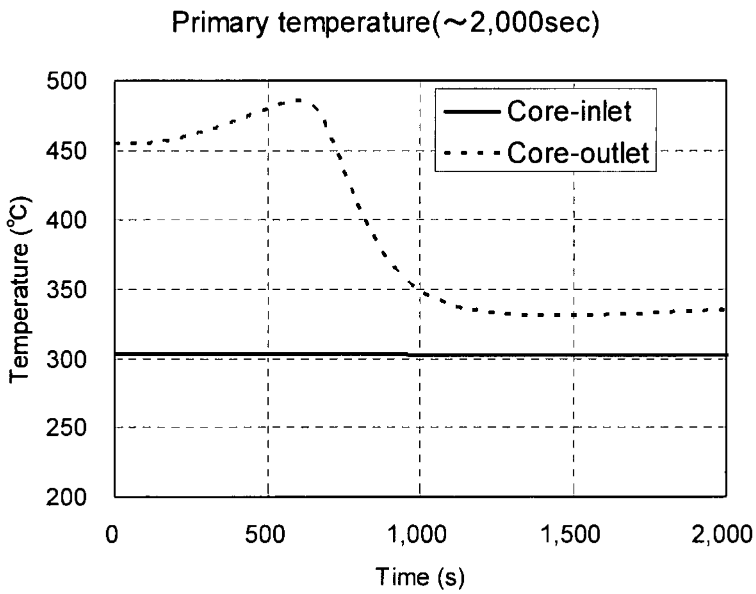


Figure 5.1.3-3 Cumulative Damage Fraction (RIR)



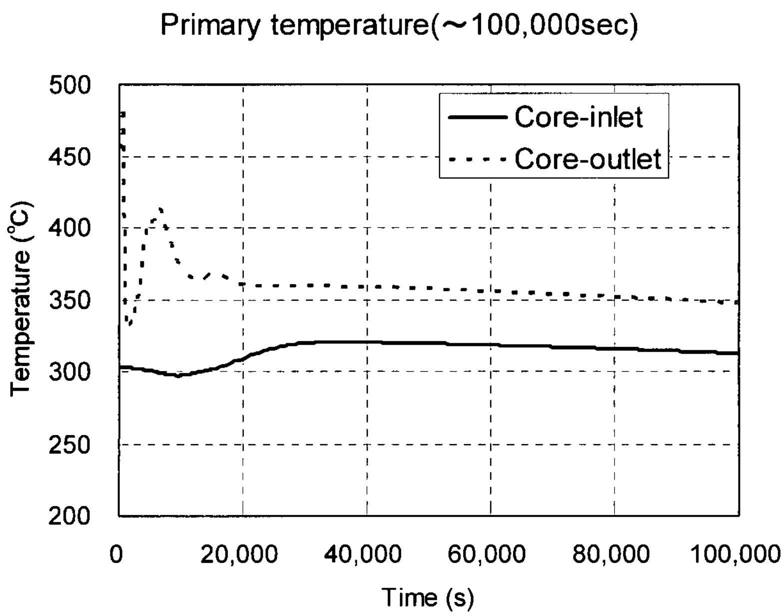


Figure 5.1.3-4 Core Inlet and Outlet Temperatures (RIR)

5.1.4 Inner or Outer Tube Failure of the Steam Generator

With respect to the SG the 4S reactor, the occurrence frequency of a sodium water reaction associated with damage to a heat transfer tube is negligibly low, based on the presumption that the heat transfer tube is a double wall tube. In this section, the system for detecting failure to either the inner or outer tubes is explained and the successive event sequence is described.

5.1.4.1 Configuration and Performance of the Detection System

The double wall tube for the 4S SG has an approximately 0.4 mm wire mesh layer between the inner and outer tubes. The wire mesh layer structurally separates the inner and outer tube; hence a crack in one heat transfer tube does not directly propagate to the other, substantially reducing the probability of failure both of the heat transfer tubes with the same fault. In addition, the detection system includes a function for constantly detecting leaks from a failed portion of the double wall tube by filling the porous layer of wire mesh with a third fluid, such as helium. The probability of an accident caused by the reaction of sodium and water is reduced to a small value for the entire plant.

In this section, the configured devices and the detecting features thereof in the leak detection system are described for the inner and outer tubes.

The layout of measurement devices making up the leak detection system for the SG is shown in Figure 5.1.4-1. Under normal circumstances, the pressure in the helium plenum is constantly maintained at an intermediate value that is lower than the pressure on the steam side and also higher than the pressure on the sodium side. One sampling system configured with a cooler, humidity instrumentation, a circulator, and a heater is installed on the feedwater side as well as the steam side of the helium each plenums. The system constantly monitors for any increase in moisture density in the helium plenum caused by failure to the inner tube, as well as monitoring pressure fluctuation in the helium plenum. This system is configured with the aim of immediately detecting occurrence of failure on one side of the heat transfer tube, such as micro-scale through medium-scale failure to the inner tube, and allowing shutdown of the plant and thereby preventing the failure from propagating to both tube boundaries simultaneously and leading to a reaction of sodium and water.

The leak detection system for the inner tube is configured with the aim of detecting the occurrence of penetrating failure independently on the inner tube. When the inner tube is damaged, steam is leaked into the helium in wire mesh layer. The leaked steam is moved to the helium plenum on the feedwater side or the steam header side, while the moisture moved to each helium plenum is detected by the increase in the moisture density of the humidity instrumentation installed in the sampling system. The abnormal state of the moisture is similarly detected by an increase of pressure in the helium plenum.

The leak detection system for the outer tube is configured with the aim of detecting the occurrence of penetrating failure independently on the outer tube. When the outer tube is failed, helium in wire mesh layer leaks into the intermediate-loop sodium and the pressure drop in the plenum caused by the outflow of helium is detected to indicate failure to the outer tube. Some of the helium leaking into the sodium dissolves; the remaining undissolved helium accumulates in the cover gas plenum of the SG. The leak detection system for the outer tube is configured with the aim of monitoring the pressure of the cover gas in the SG and the helium density in the cover gas to rapidly detect any helium leakage from initial micro-scale to medium-scale leakage.

5.1.4.2 Identification of Causes and Frequency Classification

This event results when the inner or outer tube of a heat transfer tube for the SG is damaged during reactor operation. The occurrence frequency is assumed to be once or more during the life of the reactor. Consequently, this event is classified as an AOO.

5.1.4.3 Sequence of Events and System Performance

When the inner or outer tube of the heat transfer tube for SG is damaged during reactor operation, damage to the heat transfer tube is detected by the detection systems for the inner or outer tubes, whereupon the reactor is manually shut down. In the event of a delay in the manual shutdown or if multiple detection systems detect damage, the feedwater pump and the intermediate EM pump are tripped, causing primary EM pumps trip. A low primary circulation voltage/current trip signal is transmitted, and the reactor is scrammed. Upon the scram, the reflectors descend and the reactor shutdown rod is inserted, and reactor power rapidly drops.

5.1.4.4 Core and System Performance

1. Evaluation Model

   See subsection 5.1.1.3(1).

2. Input Parameters and Initial Conditions

   See subsection 5.1.1.3(2).

3. Analysis Conditions

   See subsection 5.1.1.3(3).

4. Analysis Results

   The flow rates of the primary and intermediate-loop coolant coast down before making the transition to natural circulation. The remaining event behavior is the same as described in Section 5.1.1 for a loss of offsite power.

   With respect to the integrity of the fuel, there is sufficient margin to the fuel melting point specified as the safety acceptance criteria. With respect to the integrity of the cladding, there is sufficient margin to the safety acceptance criteria indicated by the inequality $\Sigma CDF < 0.1$, even in considering the possibility of multiple occurrences.

With respect to the integrity of the coolant boundary, the core outlet coolant temperature does not exceed the temperature for rated power; therefore, the 4S design maintains the integrity of the coolant boundary with sufficient margin to the criteria.

According to these results, the design of the 4S reactor complies with the safety acceptance criteria for integrity of the fuel, of the cladding, and of the coolant boundary with sufficient margin to the criteria.

Table 5.1.4-1
Sequence of Events for Inner or Outer Tube Failure of the Steam Generator

Time (s)	Events
0.0	Manual trip
0.0	Trip of the primary- and intermediate-loop and feedwater pumps
0.0	Switch of status of the primary pumps from normal operation to flow coastdown
0.5	Start lowering of the reflector
1.0	Start opening of the AC damper
8.5	Finish lowering of the reflector
31.0	Finish opening of the AC damper
60.0	Finish of the flow coastdown state of the primary pumps, start of the natural circulation state of the primary coolant flow

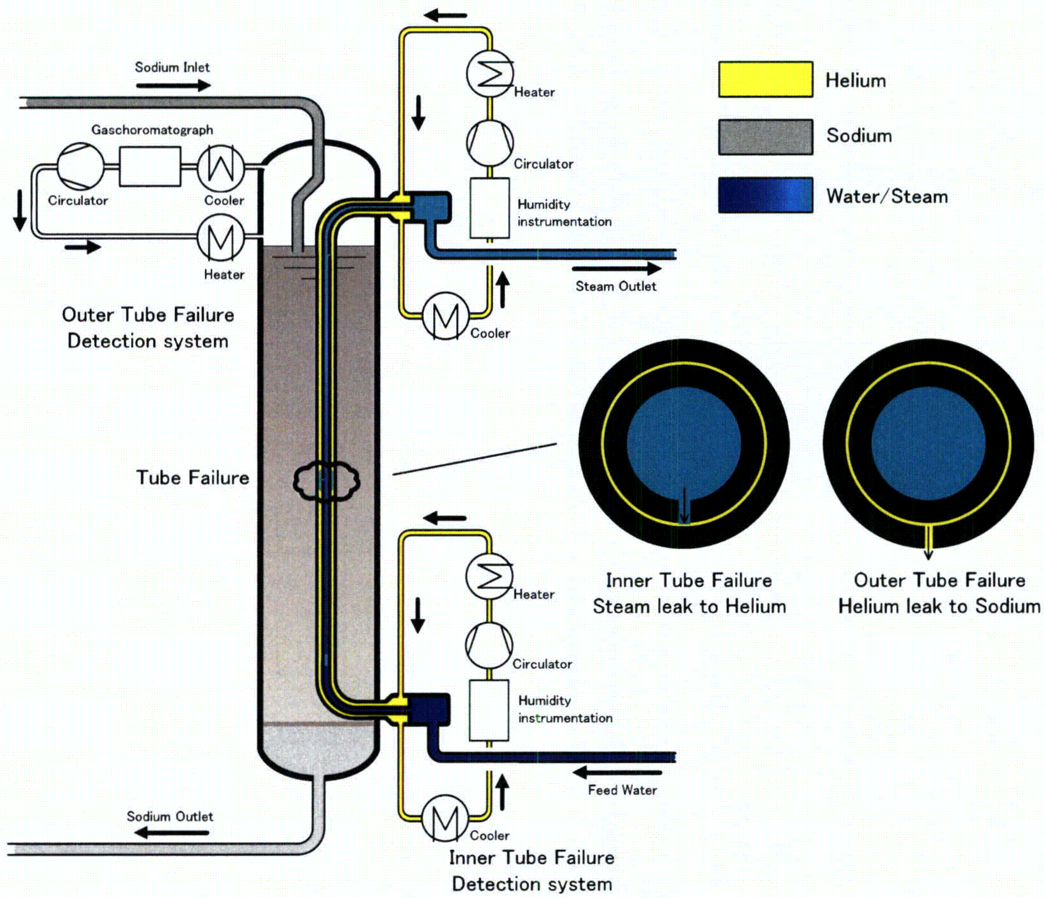


Figure 5.1.4-1 Leak Detection System Base Configuration

5.2 Design Basis Accidents (DBAs)

5.2.1 Failure of a Cavity Can (FCC)

5.2.1.1 Identification of Causes and Frequency Classification

The reflector is divided into six segments, and six cavity cans are included on each reflector segment.

The failure of a cavity can event results in increasing power from a positive reactivity insertion. The reactivity insertion is caused by a reduction in neutron leakage due to replacement of the inert gas in the failed cavity cans with sodium. (Six cavity cans are provided in each of the six reflector segments; one of these cans is assumed to be damaged.) The gas rises by the flow of sodium and buoyancy, and reaches to the cover gas region. Minute bubbles might enter the core by being circulated in the reactor. However, the bubbles would not affect reactor power because they are minute and the reactor has already shut down. This event is not expected to occur during the reactor lifetime. This event causes the most rapid increase in power among the PAs that involve reactivity addition, and represents a DBA for the reactor shutdown system.

5.2.1.2 Sequence of Events and System Performance

If a cavity can is damaged during reactor operation, positive reactivity is inserted and increases reactor power, resulting in a reactor power high signal from the power-range monitor to the safety protection system.

This results in a scram signal and causes the power supplies of the primary- and intermediate-loop circulation electromagnetic pumps to switch to the MG-set coastdown power supply. The reflectors descend and the shutdown rod are inserted, resulting in a rapid decrease in reactor power. The subsequent plant response is identical to that of the loss of offsite power described in Section 5.1.1.

5.2.1.3 Core and System Performance

1. Evaluation Model

   Analysis code: ARGO

   Evaluation model: same as described in subsection 5.1.1.3(1).

2. Input Parameters and Initial Conditions

   The reactivity insertion is assumed to be 30 cents, which is the maximum reactivity caused by failure of a cavity can. The insertion rate is set to 30 cent/s.

The initial conditions are identical to those for the loss of offsite power described in subsection 5.1.1.3(2).

3. Analysis Conditions

System response conditions and single failure conditions are identical to those for the loss of offsite power described in subsection 5.1.1.3(3) with the following exception:

- For the steam system, instantaneous loss of the heat removal function is assumed upon the trip of the feedwater pump.

4. Analysis Results (Response Behavior)

The sequence and start time of each phenomenon and action are shown in Table 5.2.1-1.

The change in reactor power and the change in primary flow rate are shown in Figure 5.2.1-1. The change of fuel temperature of the hottest pin and cladding temperature are shown in Figure 5.2.1-2. The change in CDF is shown in Figure 5.2.1-3. The temperature change at the core outlet is shown in Figure 5.2.1-4 from the analysis results using the ARGO code.

The neutron flux high signal is transmitted in 0.6 second after the event occurrence, the primary and intermediate-loop pumps trip 1.1 seconds later, and the reflectors begin lowering 1.6 second later. The reactor power momentarily increases to up to 1.58 times the rated power, due to the reactivity insertion, until the reflectors start lowering, which causes the reactor temperature to increase. However, the power increase occurs for an extremely short time and since the core power density is low, the increase in core temperature is small. The fuel peak temperature and the cladding peak temperature of the hottest pin do not exceed 664°C and 629°C.

The flow of the primary- and intermediate-loop coolant is in natural circulation, and residual heat is removed by the RVACS and IRACS (see Figure 5.2.1-4).

The fuel peak temperature of the hottest pin is 664°C, which provides a sufficient margin to the fuel melting point for the safety acceptance criteria. The CDF value for the hottest pin is 2.8×10^{-8} in this event, which provides a sufficient margin to $\Sigma CDF < 0.1$ of the acceptance criterion, even with one occurrence time considered. Thus, the 4S design meets the acceptance criterion for core coolable geometry.

The coolant temperature of the core outlet does not exceed the temperature for the rated power and, therefore, the 4S design meets the acceptance criteria for the coolant boundary integrity.

The system design and performance are adequate to protect the fuel and primary system boundary.

Table 5.2.1-1 Sequence of Events for FCC	
Time (s)	**Events**
0.0	Failure of one cavity can, insertion of positive reactivity
0.6	Transmittal of the signals: Neutron flux level "High" signal Scram initiation signal Residual heat removal system initiation signal
1.1	Trip of the primary- and intermediate-loop and feedwater pumps
1.6	Start lowering of the reflector
2.1	Start opening of the AC damper
9.6	Finish lowering of the reflector
32.1	Finish opening of the AC damper
61.1	Finish of the flow coastdown state of the primary pumps, start of the natural circulation state of the primary coolant flow

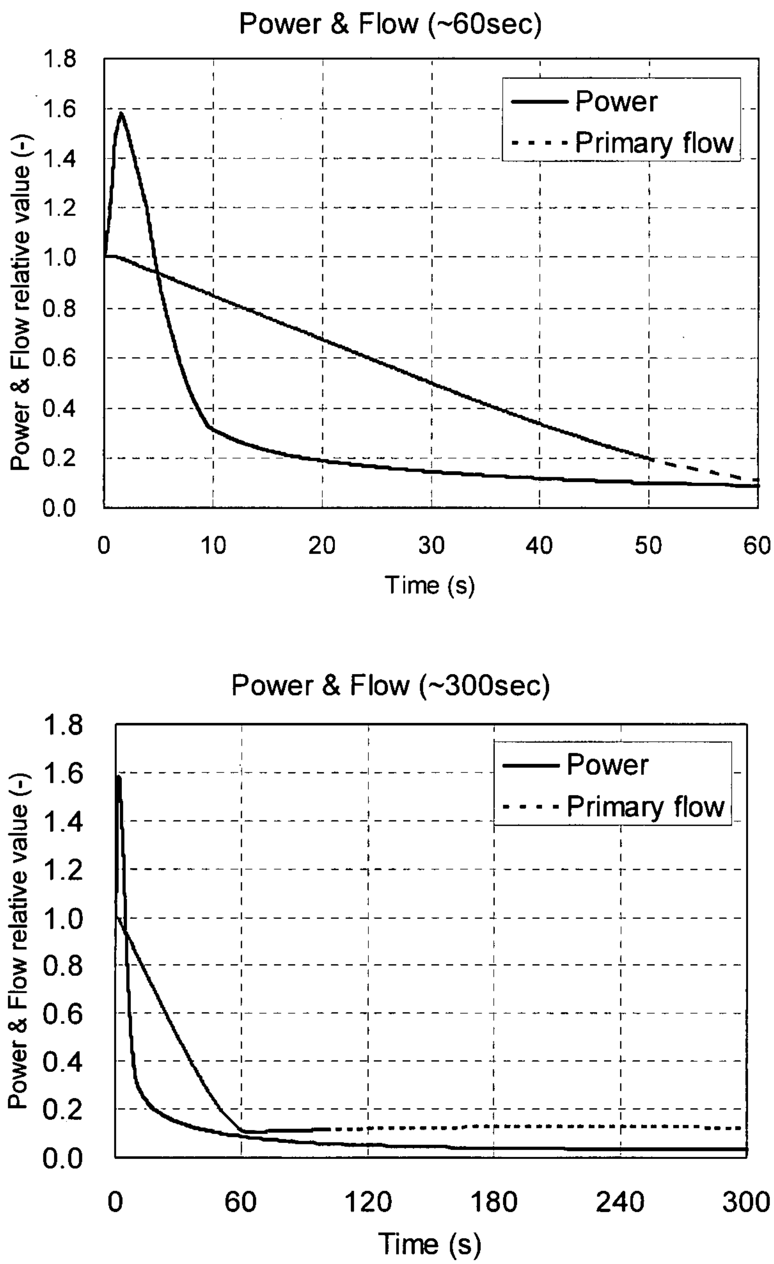


Figure 5.2.1-1 Power and Primary Flow (FCC)

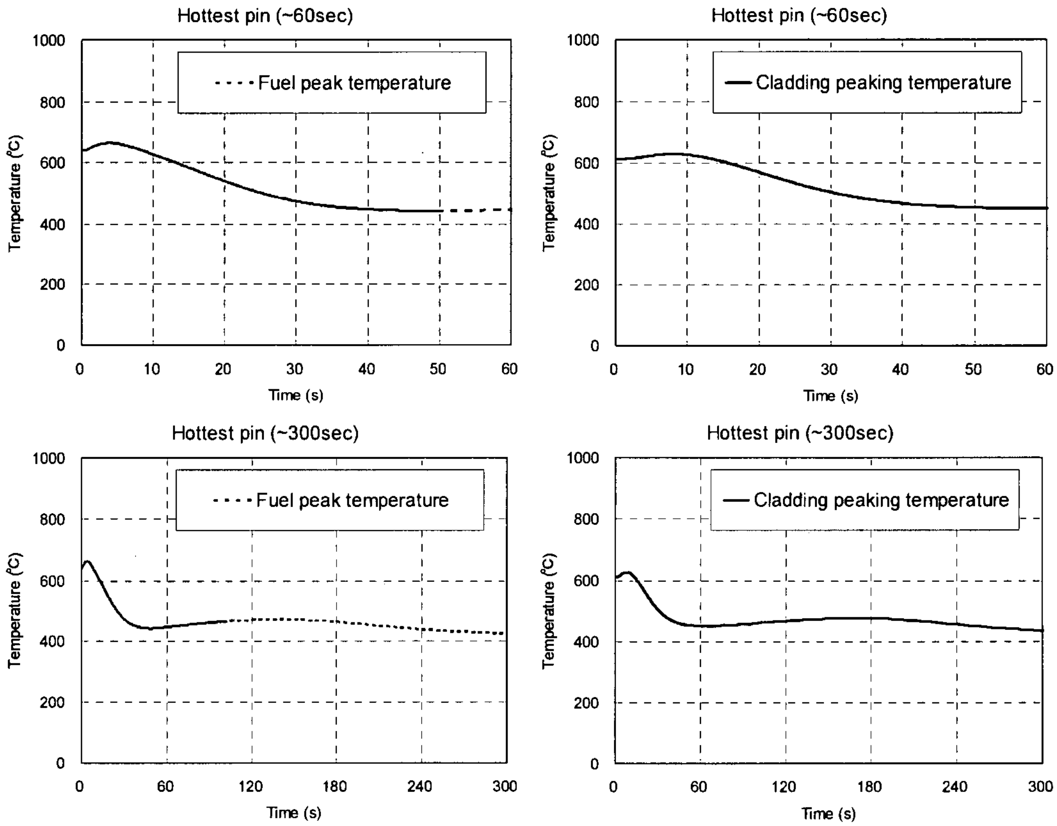


Figure 5.2.1-2 Fuel and Cladding Peak Temperature (FCC)

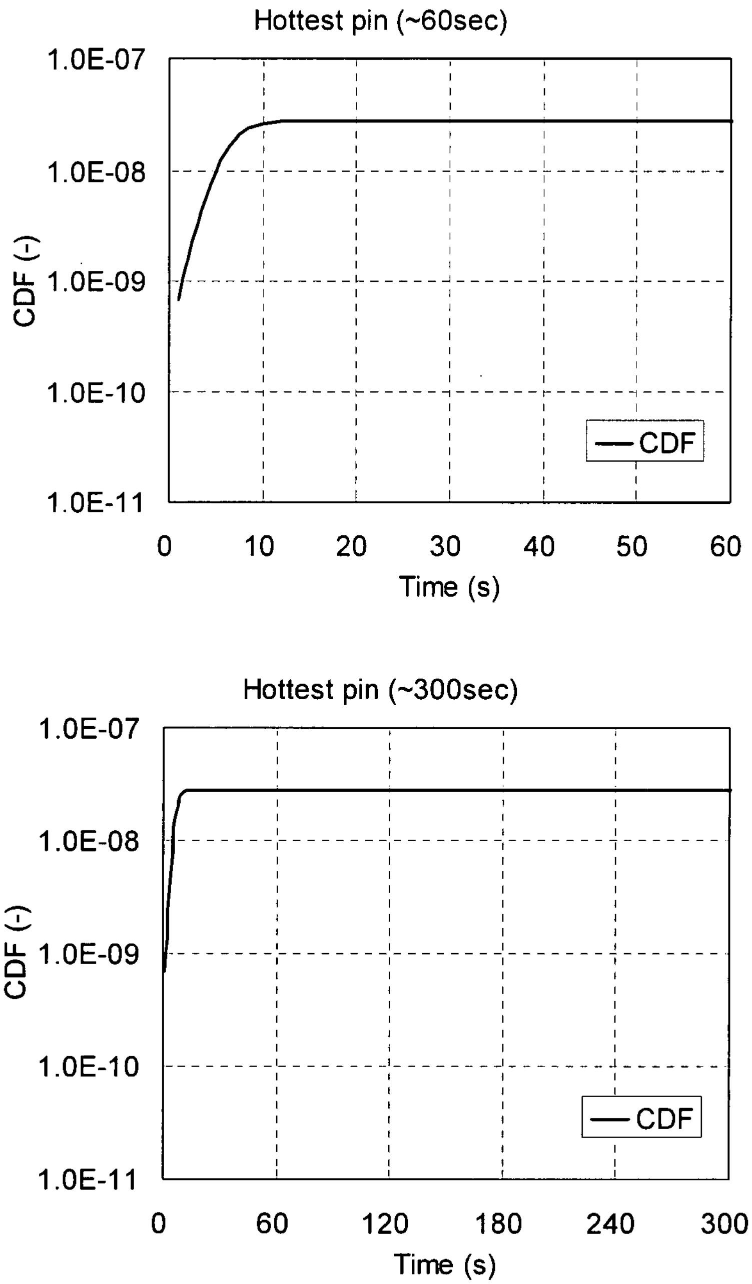
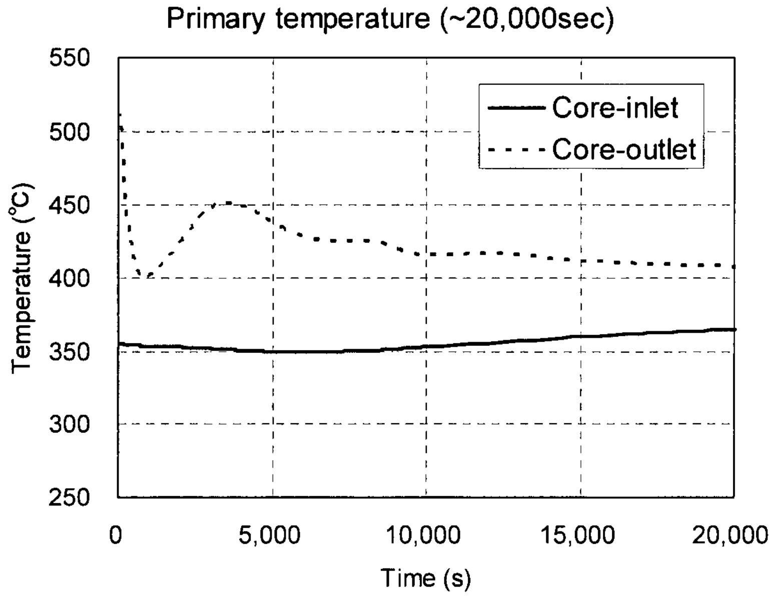


Figure 5.1.3-3 Cumulative Damage Fraction (FCC)



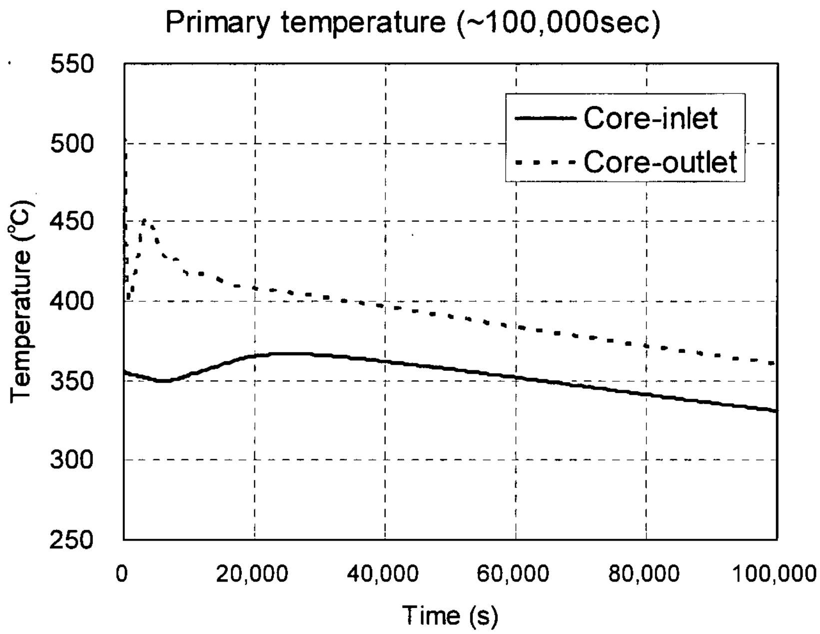


Figure 5.2.1-4 Core Inlet and Outlet Temperature (FCC)

5.2.2 Sodium Leakage from Intermediate Piping (SLIP)

5.2.2.1 Identification Causes and Frequency Classification

This event is defined as damage to intermediate-loop piping, resulting in sodium leakage and burning of sodium. This event is not expected to occur within the reactor lifetime. In addition, only the RVACS is available for residual heat removal because the use of the AC installed in the intermediate loop is not possible due to drainage of the sodium coolant from the intermediate-loop piping. This event is a DBA for RVACS.

5.2.2.2 Sequence of Events and System Performance

Figure 5.2.2-1 shows the diagram of the event sequence. If the intermediate-loop piping is damaged, the sodium coolant in the system piping leaks and burns. Sodium leakage is detected by a leakage detector, prompting the operator to immediately shut down the reactor. During the event, the remaining intermediate-loop sodium is drained into the drain tank after stoppage of the circulation pump, whereupon the residual heat of the core must be removed by the RVACS only.

Sodium that flows down onto the liner installed on the floor then flows into a reservoir, located in the tank room on a lower floor, via connecting piping. At the top of this reservoir, a combustion restricting plate with a limited vent rate is provided to extinguish the burning sodium by depriving it of oxygen. In addition, a heat sink material (AL_2O_3) with superior anti-sodium properties is installed in the reservoir to lower the temperature of the stored sodium, hence ensuring the temperature of the concrete in the lower portion of the reservoir is restricted to within the limited value.

The pressure in the affected area rises due to the sensible and combustion heat of the leaked sodium. For the rise in pressure, the inner pressure is restricted to below the building design pressure by using the pressure relief damper installed on the steam generator enclosure.

5.2.2.3 Core and System Performance

1. Evaluation Model

   The major initial conditions for the core and system response are identical to those for the loss of offsite power described in subsection 5.1.1.3(1).

2. Input Parameters and Initial Conditions

   The initial conditions are identical to those for the loss of offsite power described in subsection 5.1.1.3(2).

3. Analysis Conditions

The analysis conditions, system response conditions, and single failure conditions are described below.

- The core is tripped by a manual trip.
- Heat removal from the intermediate heat exchanger is assumed to be lost at an early stage. (Loss of heat removal of the intermediate-loop system is conservatively set.)
- The single failure assumed is one stuck reflector.
- Due to the absence of active components in RVACS, no single failure is taken into account for the residual heat removal system.

Other conditions are identical to those for the loss of offsite power described in subsection 5.1.1.3(3).

4. Analysis Results (Response Behavior)

The sequence and start time of each phenomenon and action is shown in Table 5.2.2-1 for the analytical results.

The changes in reactor power and primary flow rate are shown in Figure 5.2.2-2. The changes in fuel temperature of the hottest pin and cladding temperature are shown in Figure 5.2.2-3. The change in CDF is shown in Figure 5.2.2-4. The temperature changes at the core outlet and inlet are shown in Figures 5.2.2-5 from the analytical results using the ARGO code.

After the event occurs, the primary- and intermediate-loop and feedwater pumps trip, and 0.5 second later, the reflector begins to descend. The initial behavior is relatively similar to that for the loss of offsite power described in Section 5.1.1. The fuel peak temperature and the cladding peak temperature of the hottest pin do not exceed 639°C and 613°C, respectively.

Regarding long-term behavior, the flow of the primary coolant is in a natural circulation state, and residual heat is removed by the RVACS (see Figure 5.2.2-5).

Regarding fuel integrity, the fuel peak temperature is 639°C, which provides sufficient margin to fuel melt for the safety acceptance criterion. Regarding the cladding integrity, the CDF value for the evaluation pin is less than $3\text{x } 10^{-8}$ in this event, which provides a sufficient margin for $\Sigma CDF < 0.1$ as the acceptance criterion, even with one occurrence time considered. Thus, the 4S design meets the acceptance criteria for core coolable geometry.

The coolant temperature of the core outlet does not exceed the temperature for the rated power; therefore, the 4S design meets the acceptance criterion for coolant boundary integrity with sufficient margin.

Table 5.2.2-1
Sequence of Events for SLIP

Time (s)	Events
0	Sodium leakage from intermediate loop
0	Reactor manual trip Loss of heat removal from IHX (immediate loss of heat removal from intermediate-loop system is conservatively set considering draining of sodium from intermediate loop)
60	Residual heat removal by RVACS Natural circulation state of the primary coolant flow

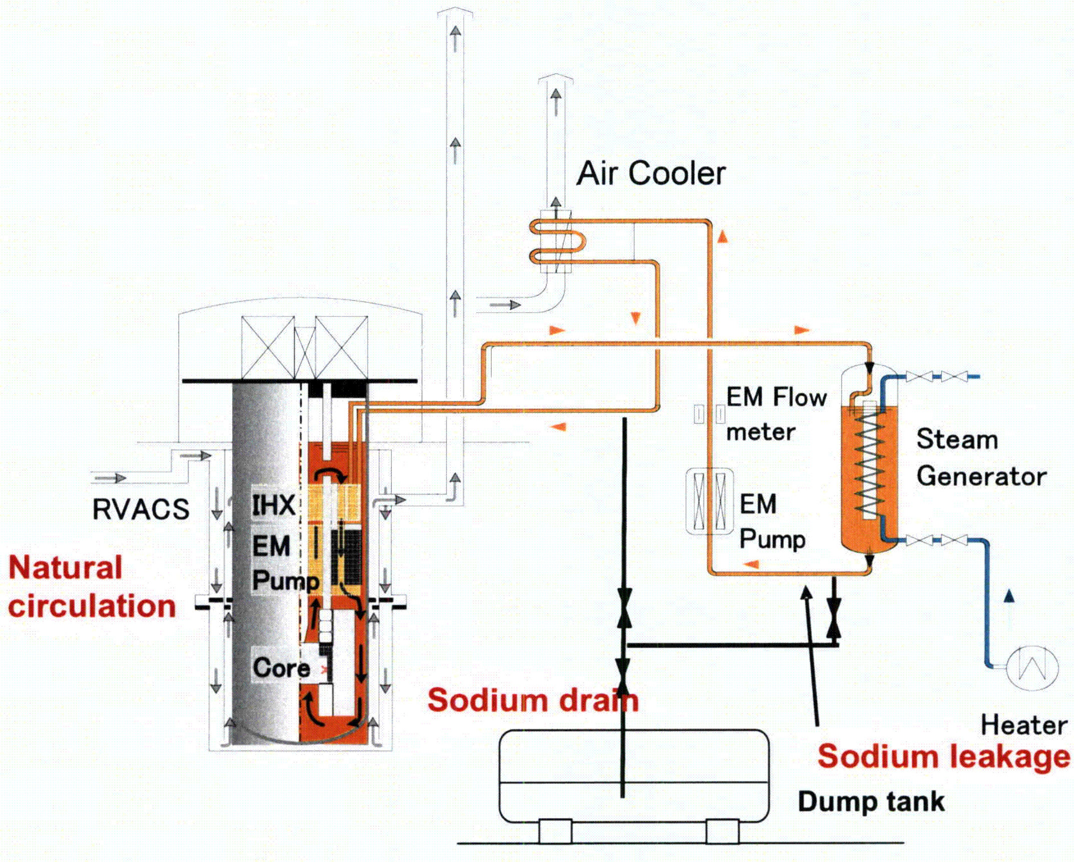


Figure 5.2.2-1 Diagram of Sodium Leak from Intermediate Piping

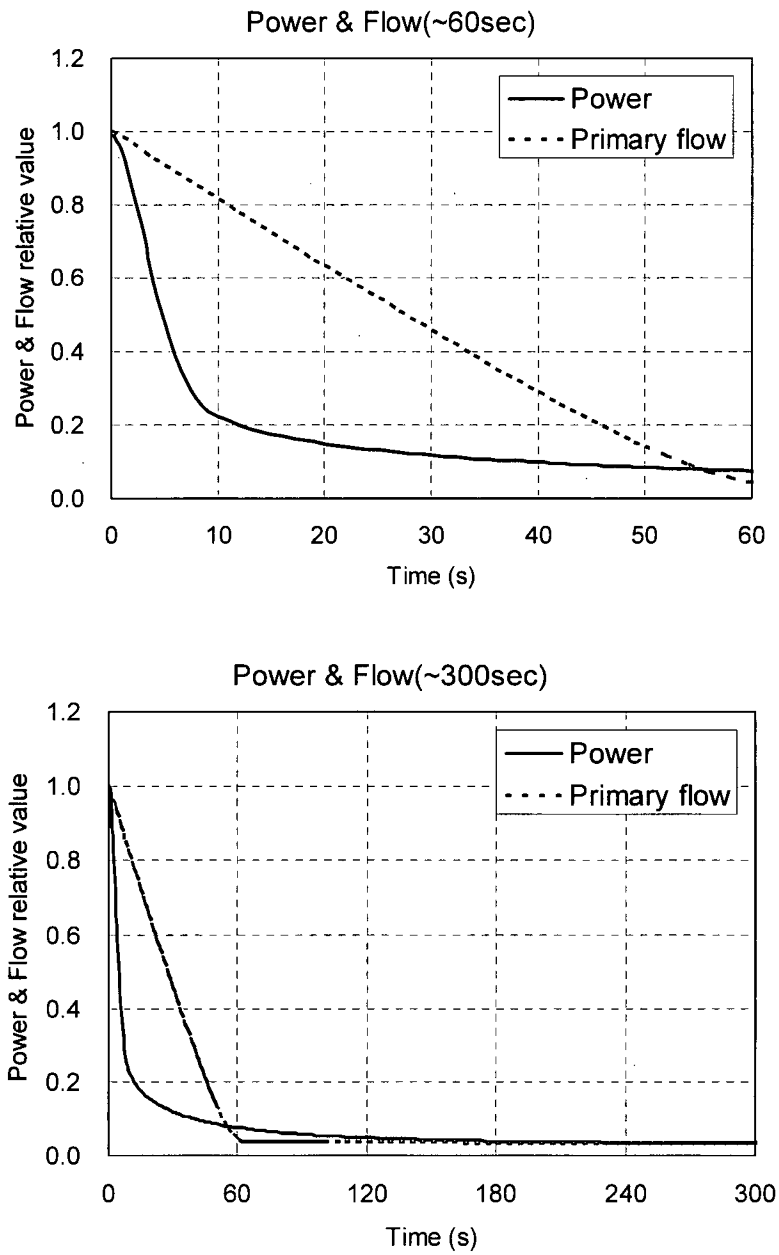


Figure 5.2.2-2 Power and Flow Short and Long Term (SLIP)

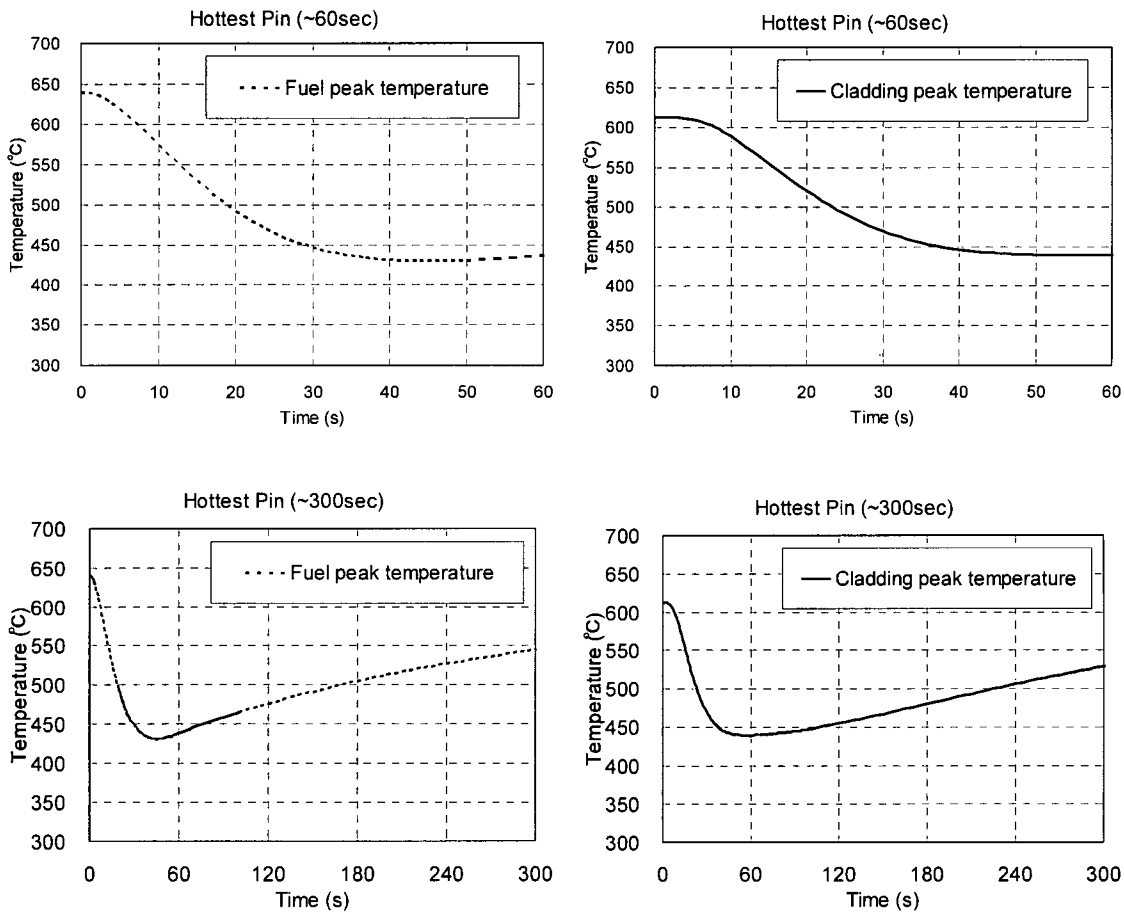
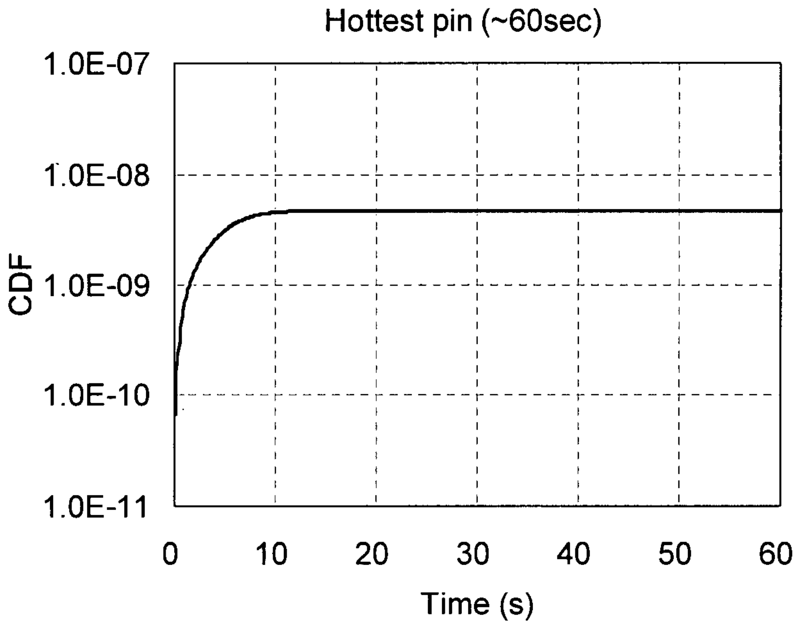


Figure 5.2.2-3 Fuel and Cladding Peak Temperature (SLIP)



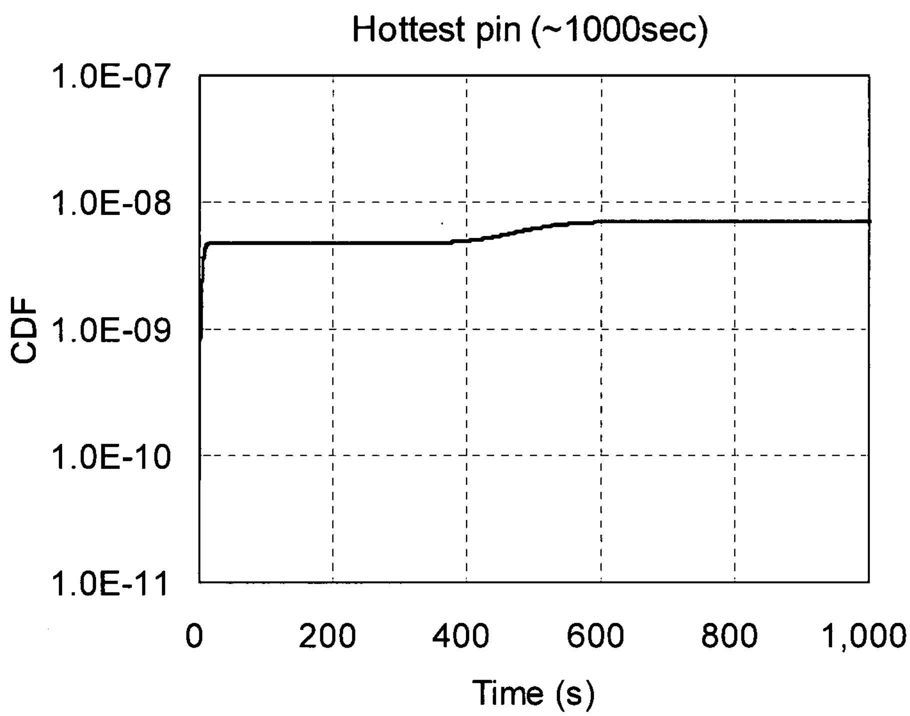
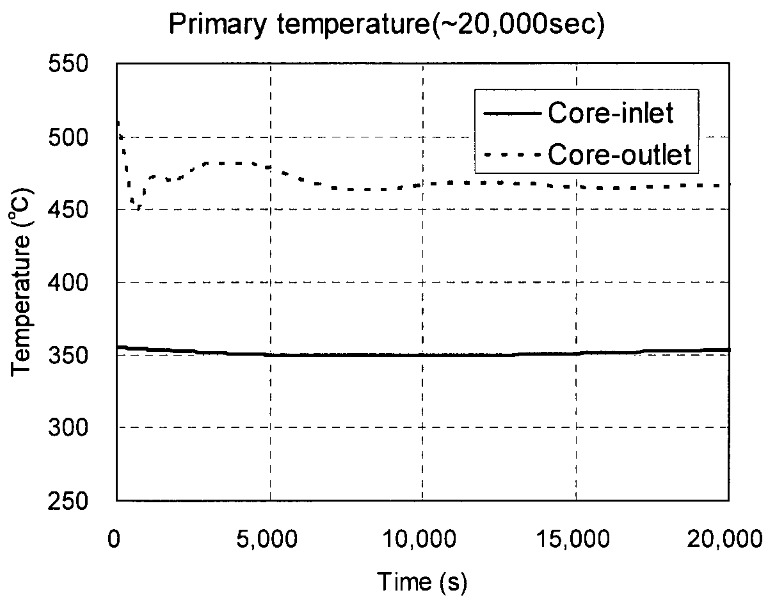


Figure 5.2.2-4 Cumulative Damage Fraction (SLIP)



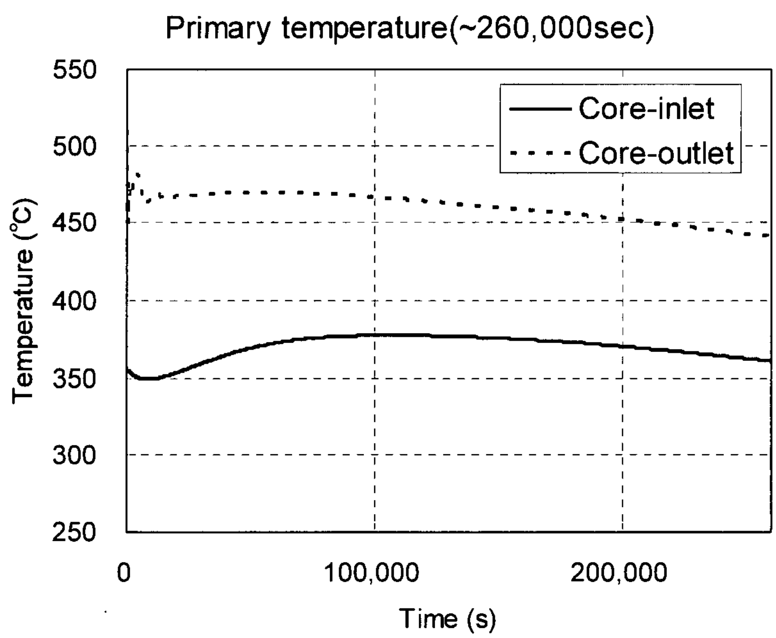


Figure 5.2.2-5 Core Inlet and Outlet Temperature (SLIP)

5.2.3 Primary Cover Gas Boundary Failure

5.2.3.1 Identification of Causes and Frequency Classification

This event is defined as the failure of the primary argon cover gas boundary, a portion of which is assumed to be damaged during normal operation, causing an inflow of the primary cover gas into the containment vessel. This event is not expected to occur within the reactor lifetime. In addition, this event is a DBA for the containment vessel.

5.2.3.2 Sequence of Events and System Performance

When the boundary containing the primary argon gas is damaged, the primary argon gas, as a radioactive material, and FPs, which are possibly contained in the primary argon gas, leak and flow into the containment top dome and subsequently leak out to the environment.

When the containment facility radioactive concentration high signal or the primary argon gas pressure low signal is generated, the operator shuts down the reactor. Subsequently, the decay heat is removed by the residual heat removal systems and the containment facility is isolated after the reactor shutdown.

5.2.3.3 Core and System Performance

The system response of the reactor is the same as that for the normal reactor shutdown. The radiation dose due to FPs such as noble gases is evaluated in the condition of 1 percent fuel pin failure.

1. Evaluation Model

   The main evaluation model is described in subsection 4.1.1.7 and Appendix A.2.

2. Input Parameters and Initial Conditions

   Main initial conditions are as shown in subsection 4.1.1.7 and Appendix A.2.

3. Analysis Conditions

   Two layers of boundaries are modeled for the shielding plug and its related equipment, located between the primary cover gas space and the inside of the top dome. The assumption for this evaluation is that either of the two boundaries is damaged and the other retains integrity.

   The leakage rate from the cover gas to the top dome, namely, that of the shielding plug and its related equipment, is assumed to be 1 percent per day for analysis. Actually, owing to the two layers of boundaries, the leakage rate at one boundary damage is designed to be less than 1 percent per day. For the leakage rate from the top dome to the environment, the same value is assumed.

4. Analysis Results (Response Behavior)

Table 5.2.3-1 lists the radiation dose assuming 1 percent fuel damage. The distance from the reactor for both the EAB and LPZ is set to 50 meters based on the plot plan, as shown in Figure 5.2.3-1. From Table 5.2.3-1, the total radiation dose for both the EAB and LPZ has a large margin relative to the limit dose of 25 rem.

Table 5.2.3-1
Radiation Dose in the Environment Under Conditions of 1% Fuel Failure

Distance (m)	50
EAB (rem)	0.0003
LPZ (rem)	0.01
Acceptance Dose Criterion (rem)	25

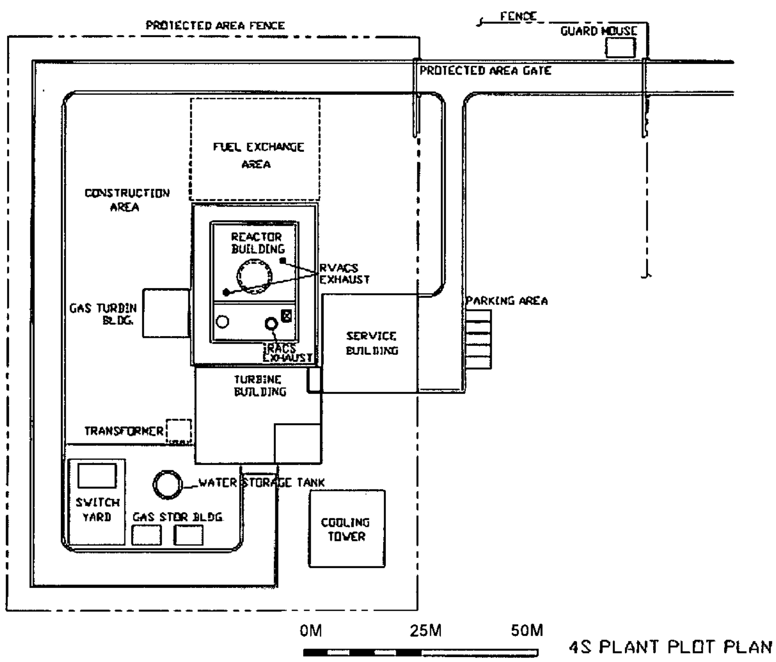


Figure 5.2.3-1 Plot Plan

5.3 Anticipated Transient Without Scram (ATWS)

5.3.1 Introduction

The ATWS event described in this chapter is a loss of offsite power without a scram. With respect to the ATWS event for metal fuel, experiments for the unprotected loss of flow (ULOF) and unprotected loss of heat sink were conducted in EBR-II from 100 percent power [47]. The experiments proved that the metal fuel core can be statically stabilized without fuel failure during these ATWS events. The 4S reactor has the same features relating to fuel configuration and reactivity as EBR-II, as well as passive safety.

For the ATWS event, when upper side of 95 percent probability at a 95 percent confidence level is within the acceptance criterion, the 4S design is validated. The analysis conditions and the procedure are described in Section 4.1.2.

According to the analysis results, the design of the 4S reactor complies with the safety criteria required to ensure the core coolable geometry.

5.3.2 Loss of Offsite Power without Scram

5.3.2.1 Identification of Causes and Frequency Classification

This event results when part or the whole of the power supply of a station is lost during power operation of the reactor. The reactor is normally scrammed by a signal of low power voltage, but in this event there is a failure to scram.

5.3.2.2 Sequence of Events and System Performance

The reactor is automatically shut down under normal conditions by the operation of the plant protection system and the decay heat is removed by the residual heat removal system after the reactor is tripped. However, if there is a failure to scram the reactor, the flow rate of the primary coolant decreases and the temperature of the reactor core increases due to a mismatch between the flow rate and power. The increase in reactor core temperature causes reactivity feedback from the Doppler effect, coolant density change, and fuel density change, whereupon the reactor power decreases. The temperature of the core continuously increases, due to the mismatch between the flow and power, but with established natural circulation, the temperature starts to decrease. Subsequently, the reactor temperature decreases gradually due to residual heat removal performed with natural circulation by the RVACS.

5.3.2.3 Core and System Performance

1. Evaluation Model

   Analysis code: ARGO

   Evaluation model: Same as described in subsection 5.1.1.3(1) with the following exception:

   - Reactivity coefficient for reference case analysis is defined as the reference condition (see Tables 4.1.1-3 and 4.1.2-1).

2. Input Parameters and Initial Conditions

   The initial conditions are the same as those described under loss of offsite power in Section 5.1.1 with the following exceptions:

   - The initial power is defined as 100 percent of the rated power.
   - Set temperature of primary and intermediate coolant inlet/outlet as plant heat balance temperature.
   - The evaluated pin is that with the cladding peak temperature (i.e., nominal hottest pin).

3. Analysis Conditions

   The reference case analysis conditions are as follows:

   - Single failure is not considered.

   The uncertainty case analysis procedure and conditions are follows:

   - Identify the plausible phenomena using phenomena identification and ranking table (PIRT) [37, 38].
   - Determine the important phenomena and parameters for sensitivity analysis following the PIRT process.

     The ranking table of the relative importance of the plausible phenomena is made by five experts. Seventeen phenomena and parameters are selected from 88 plausible phenomena according to the table. The selected phenomena are as follows:

     1. Pressure loss in the core region
     2. Doppler reactivity

3. Coolant density reactivity
4. Fuel density reactivity
5. Radial core expansion reactivity
6. Cladding density reactivity
7. Axial core and reflector displacement reactivity
8. Flow distribution of the intra- and inter-core assembly
9. Distribution of core power
10. Temperature to start eutectic reaction between fuel and cladding
11. Maldistribution of the core flow
12. Position of the IHX inlet
13. Flow coastdown time of the primary EM pumps
14. Flow coastdown time of the intermediate EM pump
15. Pressure loss coefficient of the air side of the IRACS
16. Heat transfer between GV and air
17 Radiation between RV and GV, GV and collector

- Perform sensitivity analysis and determine the dominant parameters for statistical evaluation.

  Based on existing knowledge, the width of the uncertainty band is set for the selected 17 parameters, and sensitivity analysis is performed. As a result of the sensitivity analysis, the parameters having an effect of less than 3°C on the cladding temperature rise, which strongly influences the CDF, are excluded from the parameters for statistical analysis. Finally, the following 10 dominant parameters are selected for statistical analysis:

  1. Pressure loss coefficient in the core region
  2. Doppler reactivity
  3. Coolant density reactivity
  4. Fuel density reactivity
  5. Radial core expansion reactivity
  6. Flow distribution of the intra- and inter-core assembly
  7. Distribution of core power
  8. Maldistribution of the core flow
  9. Flow coastdown time of the primary EM pumps
  10. Flow coastdown time of the intermediate EM pump

- Perform statistical analysis.

  Based on existing knowledge, the width of the uncertainty band is set for the selected 10 parameters and a total of 60 cases of statistical analysis are performed. The 60 calculation cases show the results with a 95 percent probability at a 95 percent confidence level [48].

The remaining conditions are the same as described for the loss of offsite power in Section 5.1.1.

4. Analysis Results (Response Behavior)

The sequence and start time of each phenomenon are shown in Table 5.3.2-1.

a. Reference Case

- The changes in reactor power and primary flow rate are shown Figure 5.3.2-1.
- Reactivity components are shown Figure 5.3.2-2.
- Fuel peak temperature of the evaluated pin and the cladding peak temperature are shown Figure 5.3.2-3.
- The change in CDF is shown Figure 5.3.2-4.
- The temperature changes at the core outlet and inlet are shown in Figure 5.3.2-5.

In a loss of offsite power, the reactor is normally scrammed automatically by the plant protection system and the decay heat is then removed by the residual heat removal systems. For this event, however, there is a failure to scram the reactor, so the flow rate of the primary coolant decreases and the temperature of the core rise, due to a mismatch between the flow rate and power. The rise in core temperature causes net negative reactivity feedback from Doppler, coolant density, fuel density, and radial core expansion, whereupon reactor power decreases. Core temperature increases continuously due to a mismatch between flow rate and power, but with established natural circulation, the flow rate of the primary coolant reaches a virtually constant condition, and the temperature starts to decrease. At this time, the fuel and cladding temperatures peak at 757°C and 743°C, respectively. Subsequently, reactor temperature decreases gradually due to residual heat removal performed with natural circulation by the RVACS. Steady cooling can be confirmed in Figure 5.3.2-5.

With respect to maintaining the coolable geometry, the CDF value of the evaluated pin is 4.4×10^{-4} for this event, meaning there is sufficient margin to the transient design criterion indicated by the inequality $\Sigma CDF < 0.1$. According to the results, the design of the 4S complies with the acceptance criterion required to secure the coolable geometry, and the integrity of the coolant boundary of the coolant with sufficient margin. In addition, the 4S has no effect on the containment facility and also complies with the acceptance criteria for the containment facility.

b. Statistical Analysis

A total of 60 cases of the statistical analysis are performed. Figures 5.3.2-6 and 5.3.2-7 show the 60 calculation run results of peak cladding temperature and CDF, respectively. A total of 60 cases of each calculation run indicate a normal distribution for peak cladding temperature and CDF. From the distribution, cladding peak temperature and CDF with 95 percent probability is found. The average value of 60 cases and the 95 percent probability value are as follows:

- Cladding peak temperature
  - Average value: 745°C
  - 95 Percent probability value: 798°C

- CDF
  - Average value: 4.6×10^{-4}
  - 95 Percent probability value: 1.2×10^{-2}

Thus, 95 percent probability at a 95 percent confidence level section in CDF is 0.1 or less for this event. According to the results, the design of the 4S complies with the acceptance criterion required to secure the coolable geometry with sufficient margin. In addition, the 4S has no effect on the containment facility and also complies with the acceptance criteria for the containment facility [49].

Table 5.3.2-1
Sequence of Events for Loss of Offsite Power without Scram

Time (s)	Events
0.0	Loss of the power supply of the station
0.0	Trip of the primary- and intermediate-loop and feedwater pumps
0.0	Switch of the state of the primary pumps from normal operation to flow coastdown
60.0	Finish of the flow coastdown state of the primary- and intermediate-loop pumps
60.0	Start of the natural circulation state of the primary- and intermediate-loop coolant flow

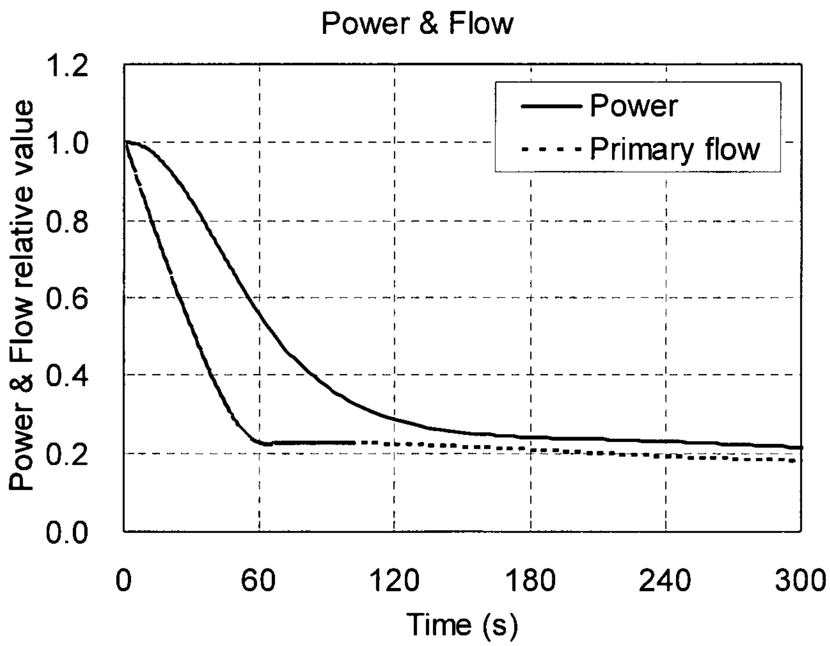


Figure 5.3.2-1 Power and Primary Flow (ATWS Reference)

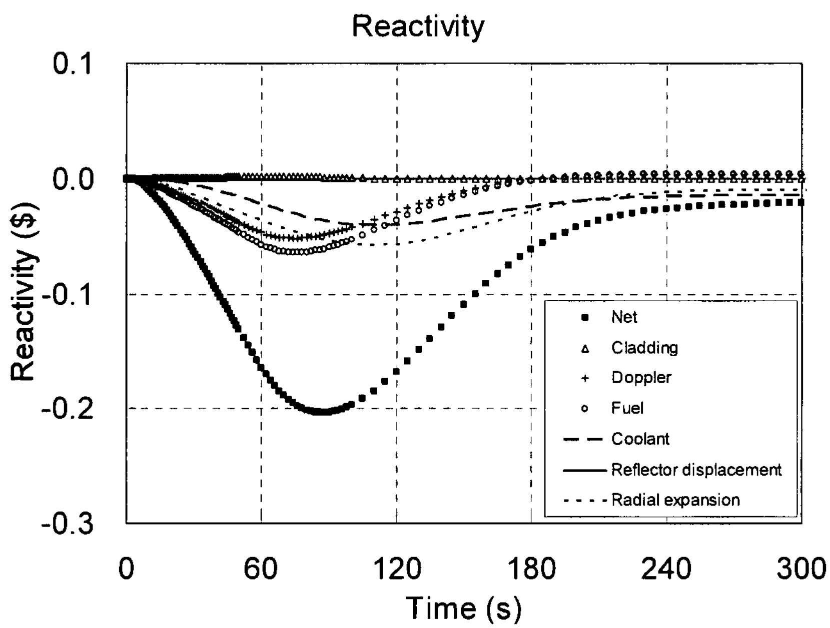


Figure 5.3.2-2 Reactivity (ATWS Reference)

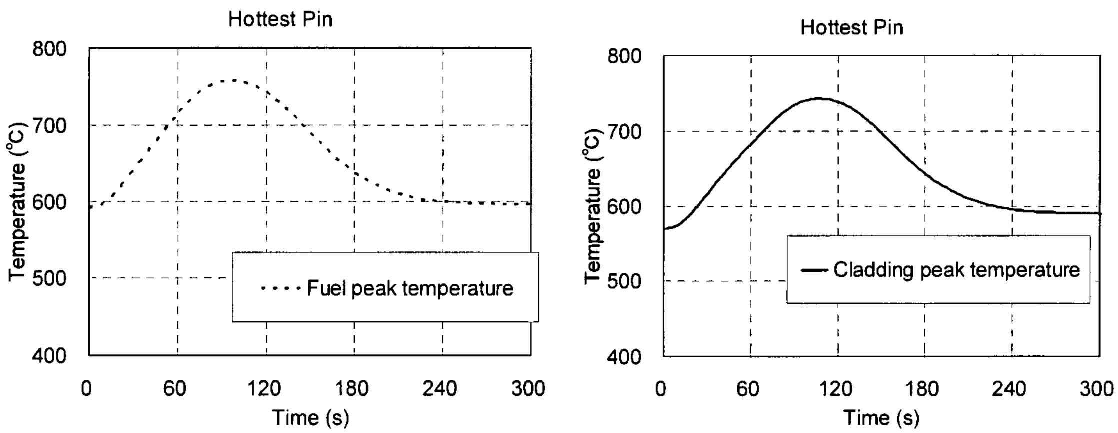


Figure 5.3.2-3 Fuel and Cladding Peak Temperature (ATWS Reference)

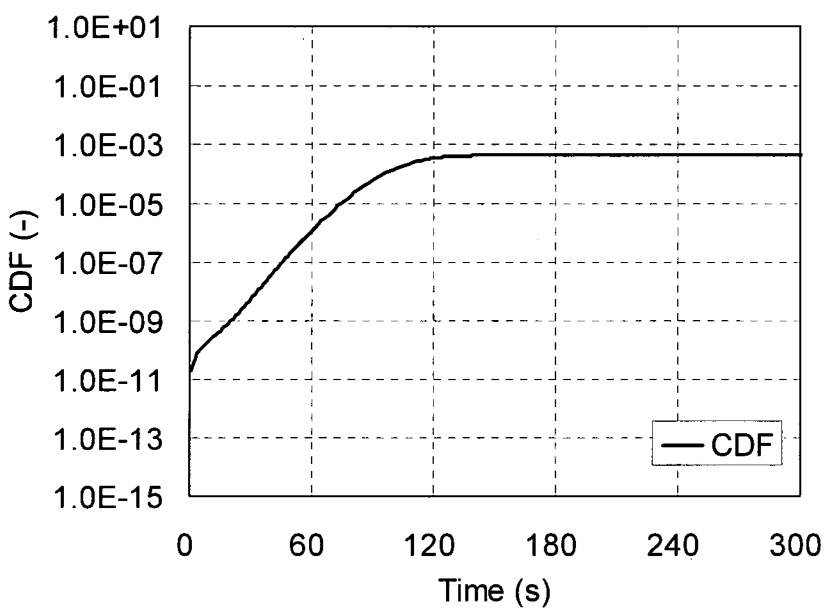


Figure 5.3.2-4 Cumulative Damage Fraction (ATWS Reference)

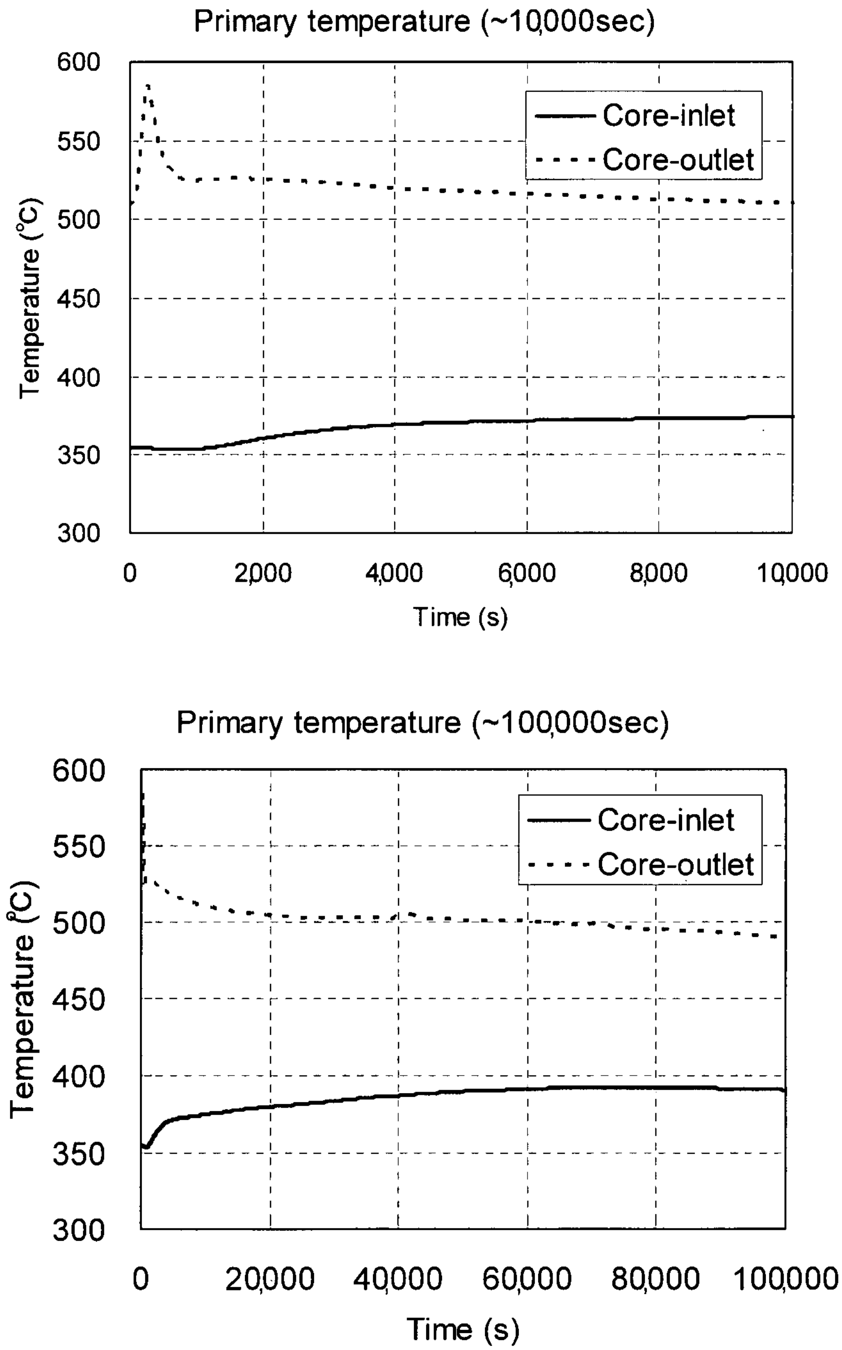


Figure 5.3.2-5 Core Inlet and Outlet Temperature (ATWS Reference)

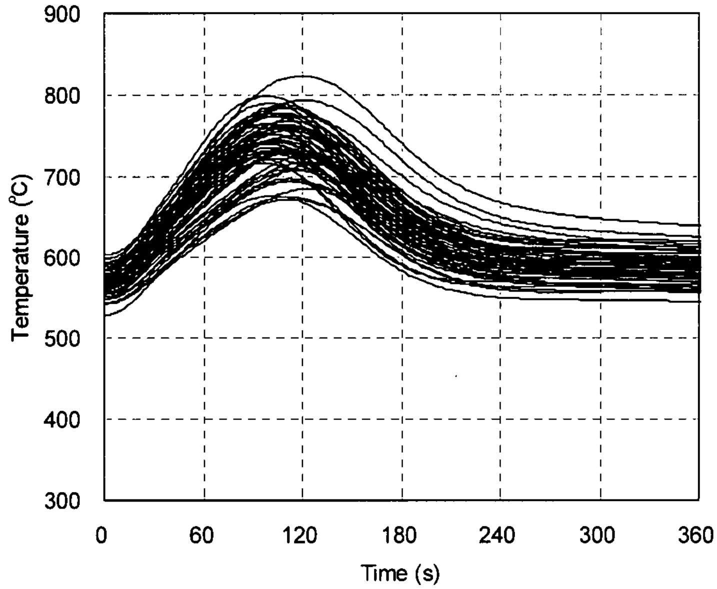


Figure 5.3.2-6 Cladding Peak Temperature (ATWS Statistical Analysis)

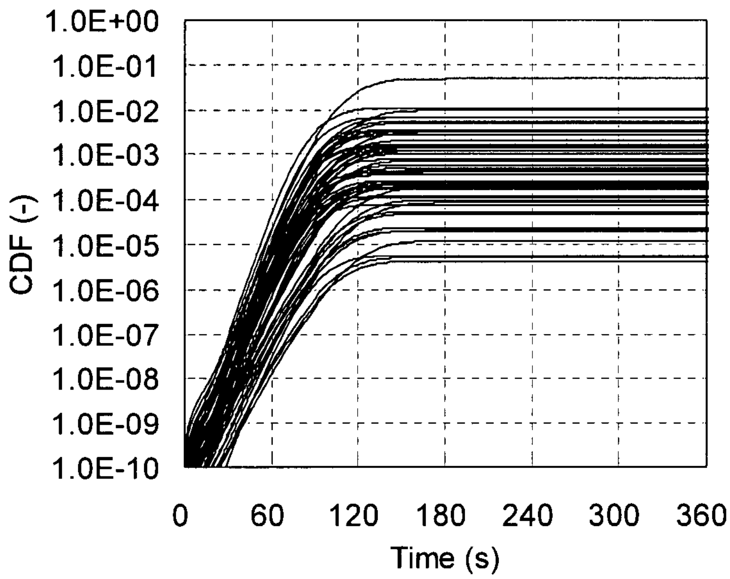


Figure 5.3.2-7 Cumulative Damage Fraction (ATWS Statistical Analysis)

6 CONCLUSIONS

6.1 Summary of Event Selection and Acceptance Criteria

Chapter 15.0 of NUREG-0800 [11] and Chapter C.I.15 of RG 1.206 [6] contain the events to be evaluated and their acceptance criteria in the safety analysis for light water reactors.

In the safety analysis for the 4S, the events to be evaluated and their acceptance criteria have been modified and replaced as follows using the basic philosophy and contents of the aforementioned documents:

- The events to be evaluated have been categorized into anticipated operational occurrences (AOOs) and design basis accidents (DBAs), as well as anticipated transients without scram (ATWS).
- Failure modes and effects analysis (FMEA) has been performed for the systems and components to select the events to be evaluated based on the frequency of occurrence and characteristics of event progression.
- The analysis acceptance criteria for the AOO, DBA, and ATWS have been determined.

6.2 Summary of the Analysis Methodology

As matters to be considered in conducting the analysis, the plant conditions, nuclear conditions and configuration conditions for the plant protection system, and the assumed conditions for single failure and radioactive materials release have been determined, as have the codes and analysis methods to be used.

6.3 Summary of Analysis Results

- Analysis Results for AOO

  The following four typical events have been chosen from those that are categorized as AOO:

  - Loss of offsite power
  - Decrease of primary coolant flow
  - Reactivity insertion by uncontrolled motion of reflector segments at startup
  - Inner or outer tube failure of the steam generator

  For each event, the acceptance criteria for the fuel temperature, fuel cladding, and primary coolant boundary are satisfied.

- Analysis Results for DBA

  Of all the events that establish the design conditions and sizing for the systems and equipment of the safety systems for reactor shutdown, decay heat removal, and the containment of radioactive materials, the following events have been determined to be typical examples to which the most severe criteria are applied to establish the accident consequences:

  - Reactivity insertion due to failure of a cavity can (FCC)
  - Sodium leakage from the intermediate piping (SLIP)
  - Primary cover gas boundary failure

  For events (FCC) and (SLIP), the acceptance criteria of the reactor core coolable geometry and coolant boundary integrity are satisfied, and no radioactive materials are released.

  For Primary cover gas boundary failure event, the exposure dose meets the requirements for the limit values of 10 CFR 50.34, even if the EAB or LPZ is set at 50 meters.

- Analysis Results for ATWS

  The event in which failure to shut down the reactor is superimposed on the loss of offsite power has been regarded as a typical event for ATWS consideration. Based on the statistical analysis, the results have also met the acceptance criteria for the reactor core coolable geometry and primary coolant boundary integrity, while the soundness of the containment system is also demonstrated.

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APPENDIX A
ANALYSIS METHODS AND ANALYTICAL CODES

A.1 Analysis Methods and Analytical Codes

This section discusses the main structures and models concerning the plant safety characteristics (dynamic behavior) analysis code ARGO and the radiological analysis method.

To demonstrate the validity of each analytical code, V&V is being performed in compliance with RG 1.203 [46] and NUREG-1737 [47], to confirm the reliability of the results. A topical report summarizing the results will be provided during the DA application period.

A.1.1 ARGO Computer Code

A.1.1.1 Outline of the ARGO Computer Code

The objective of the analysis using the ARGO code is to evaluate the integrity of fuel pins and the temperature of the primary coolant boundary during DBEs and beyond DBEs such as ATWS. The code has the capability to simulate the plant response to accidents and transients involving increase or decrease in heat removal by the balance of plant (BOP), decrease or increase of flow by the primary- or intermediate-loop coolant, reactivity insertion at full-power operation or startup, and loss of electrical power.

The ARGO code models one-dimensional flow of an incompressible fluid using a one-dimensional flow network model that includes the intermediate heat exchanger, the steam generator, and the core. Starting from an initial steady thermal-hydraulic balance, the code analyzes the above accidents and transients.

The core is modeled with multiple channels, with each channel representing one pin enclosed in the associated wrapper duct. The one-dimensional coolant flow in the channel absorbs heat from the fuel pin and flows out from the channel; the flow of each channel joins at the channel exit. The fuel, cladding, and coolant in the core are divided into nodes in the axial direction. At each elevation, the fuel pellet or slug is further divided in the radial direction into cells. The main flow network model is shown in Figure A.1.1-1.

This code describes the fuel pin under a specific thermal condition, such as the hottest pin in the core, and evaluates the integrity of fuel and cladding. The temperature distribution of the specific fuel pin is evaluated by dividing the pin in detail. The initial stress of the cladding is assumed in the detailed pin analysis, and the CDF of the cladding is evaluated using a stress-rupture correlation, i.e., the Larson-Miller parameter (LMP). Because there is a eutectic formation due to interaction between the metal fuel and cladding, an inside temperature of the cladding is calculated semi-statically; the amount of the thinning is calculated according to the inside temperature of the cladding. The eutectic is considered to be strengthless, so the thinning is reflected as an increase in the stress.

Nuclear dynamic characteristics (kinetics) are calculated using a one-point approximation model (point kinetics) and six groups of delayed neutron precursors. The reactivity contributions include the insertion reactivity, scram reactivity, Doppler feedback, and reactivity feedback from the temperature change of the fuel, cladding, and wrapper duct. In addition, reactivity contributions can include the reactivity change due to thermal expansion-caused relative displacement of the core and the reflector drive extension rod or reactor core radial expansion as an option.

The flow network models the equation of motion of the fluid in the one dimension by balancing the pump head, natural convection head, and pressure loss. The friction pressure loss is expressed in a form depending on the Reynolds Number and shape pressure loss is included. The motion equation of the flow network is systematically solved by considering the flow distribution in the reactor core, flow of the heat transport system, and flow in the heat exchanger. The SG models heat exchange of the intermediate-loop fluid (sodium) and the water of the water-steam system. The water-steam system side of the SG treats various phases of water from the liquid single phase flow of water, a liquid-vapor two-phase flow, to superheated steam.

The code models heat exchange as a counterflow between sodium-flowing tubes or between a sodium-flowing tube and a water-flowing tube. Since the Nusselt Number used in the heat exchanger model is modeled in the form depending on the Reynolds Number and the Prandtl Number, both the Lubarsky-Kaufman and Subbotin Numbers also can be modeled. The heat exchanger is divided in the axial direction into cells, and models the heat transfer between the primary side fluid and secondary side fluid through the heat transfer tube. The steel material of the shroud is considered as thermal capacity as an option.

Heat transfer by forced convection, natural convection, and radiation can be modeled. The RVACS is modeled by combining natural convection and the heat transfer models. In this code, the heat transfer can be defined between the structure and fluid, in addition to the heat transfer model. Such heat transfer is selected from a choice of forced convection, natural convection, radiation, or the constant type. RVACS heat removal mechanisms modeled in ARGO include: (1) radiation heat transfer from the RV to the GV and from the GV to the collector, and (2) convective heat transfer from the GV and collector to atmospheric air. ARGO also models air flow through the RVACS, including intake of air from the atmosphere, flow past the heat transfer surfaces by natural circulation, and exhaust of the heated air into the atmosphere.

This code models events such as the initiator and scram signal, e.g., a pump trip, scram, and flow path failure (leakage and opening or closing of a valve) in a transient.

Events that occur in the reactor systems in response to the transient, such as scram, pump trip, opening or closing of valves, etc., are modeled as occurring with a specified delay time after a monitored variable (temperature, flow rate, pump current or voltage, etc.) exceeds its set-point.

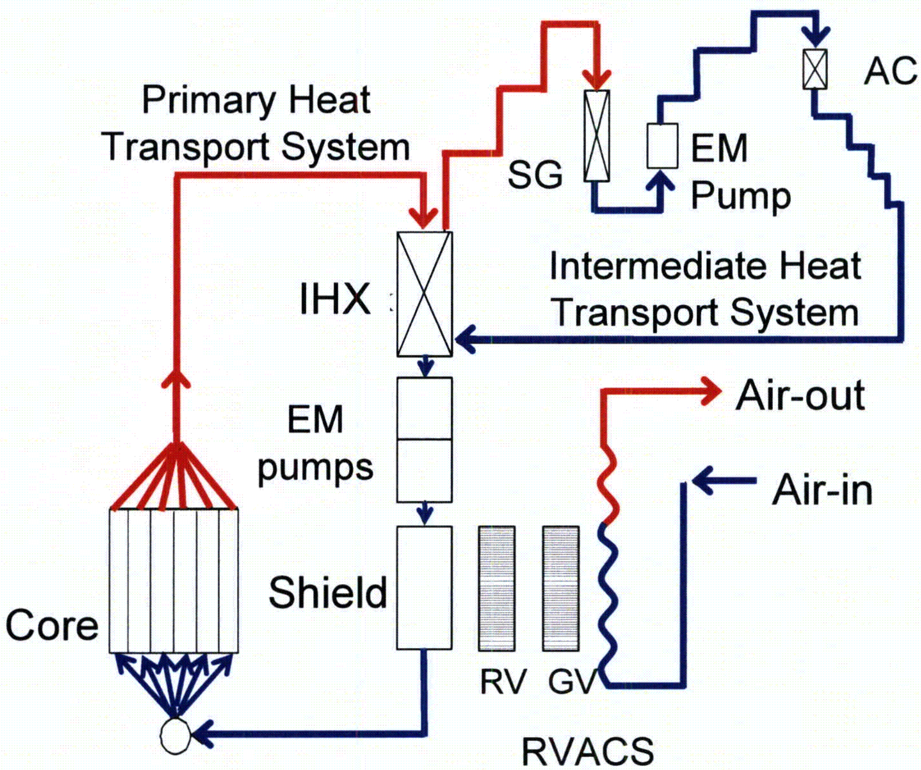

Figure A.1.1-1 Flow Network Model of 4S for ARGO Code Calculations

A.1.2 Verification and Validation of the ARGO Code

This section describes the methods used to verify the ARGO safety analysis code and to confirm its validity.

A.1.2.1 Evaluation Model Development and Assessment Process (EMDAP)

The ARGO flow network code is used in the safety analysis of 4S reactors. Verification and validation of the application methodology of the ARGO code to the safety analysis of the 4S reactor is made in accordance with the EMDAP method described in Regulatory Guide 1.203 issued by the U.S. NRC.

This EMDAP method serves as guidance for those developing or evaluating analytical codes and provides details of the development process used to evaluate transients and accidents involving nuclear power plants. Analytical codes that are developed properly in accordance with the EMDAP method can be considered as acceptable codes by the U.S. NRC.

As shown in Figure A.1.2-1, EMDAP's methodology consists of four elements and 20 steps. The four elements are as follows:

- Element 1: Defines the target event to be analyzed, and the phenomenon or process appearing in such event, as well as the figure of merit (FoM). It also selects the important phenomenon or process through the use of phenomena identification and ranking table (PIRT) and other tools. Furthermore, it clarifies the requirements for an evaluation model that evaluates the selected phenomenon and process.

- Element 2: Constructs a database for verifying a model, which evaluates the important phenomenon and process selected in Element 1. While the test data and plant data are included in this database, the scalability, similarity, and other aspects of such data must be evaluated.

- Element 3: Based on the requirements resulting from Element 1, the methodology produces a model and incorporates it into analytical code.

- Element 4: Verifies the evaluation model developed in Element 3, using the database created in Element 2. Its evaluation process is classified into the following two evaluations:

  — Bottom-up evaluation: This evaluation focuses on evaluation of the individual basic models of analytical code. In this evaluation, the pedigree, applicability, and fidelity to the separate effects test (SET) data and scalability of the expression used in the models are evaluated.

  — Top-down evaluation: This evaluation focuses on the ability and performance evaluation of the analytical code. In this evaluation, the overall performance (governing equation, numerics, applicability, fidelity to integral effects test [IET] data and scalability) are evaluated.

A.1.2.2 Validation of the Safety Analysis Code

As described in subsection A.1.1.1, the verification process of the safety analysis code involves conducting a test analysis, which is described in steps 14 and 18 of EMDAP (Figure A.1.2-1). Validation of the ARGO safety analysis code was also conducted, using data obtained from the SETs, component performance tests (CPTs), and IETs.

SET is a simple test that provides information concerning a basic model and/or one phenomenon, and its test analysis is performed to demonstrate the sufficiency of fundamental and constitutive equations. It needs to be performed for the phenomena selected in PIRT in step 4.

CPT provides performance data concerning the individual components of the 4S reactor, and is performed to demonstrate the sufficiency of the component modeling in the ARGO code. It must be performed for the specific components in actual use.

IET is a test used to simulate the scaled 4S reactor, which is performed to evaluate the integrated system and the interaction between various components and processes in the system. In addition, it is provided to demonstrate accuracy in simulating a reaction inside the integrated system and offer information on the noding of a complicated system.

To examine the need for an experiment to confirm the validity of the 4S reactor's analytical code, a PIRT evaluation was performed with several experts from the U.S. and Japan participating as advisors.

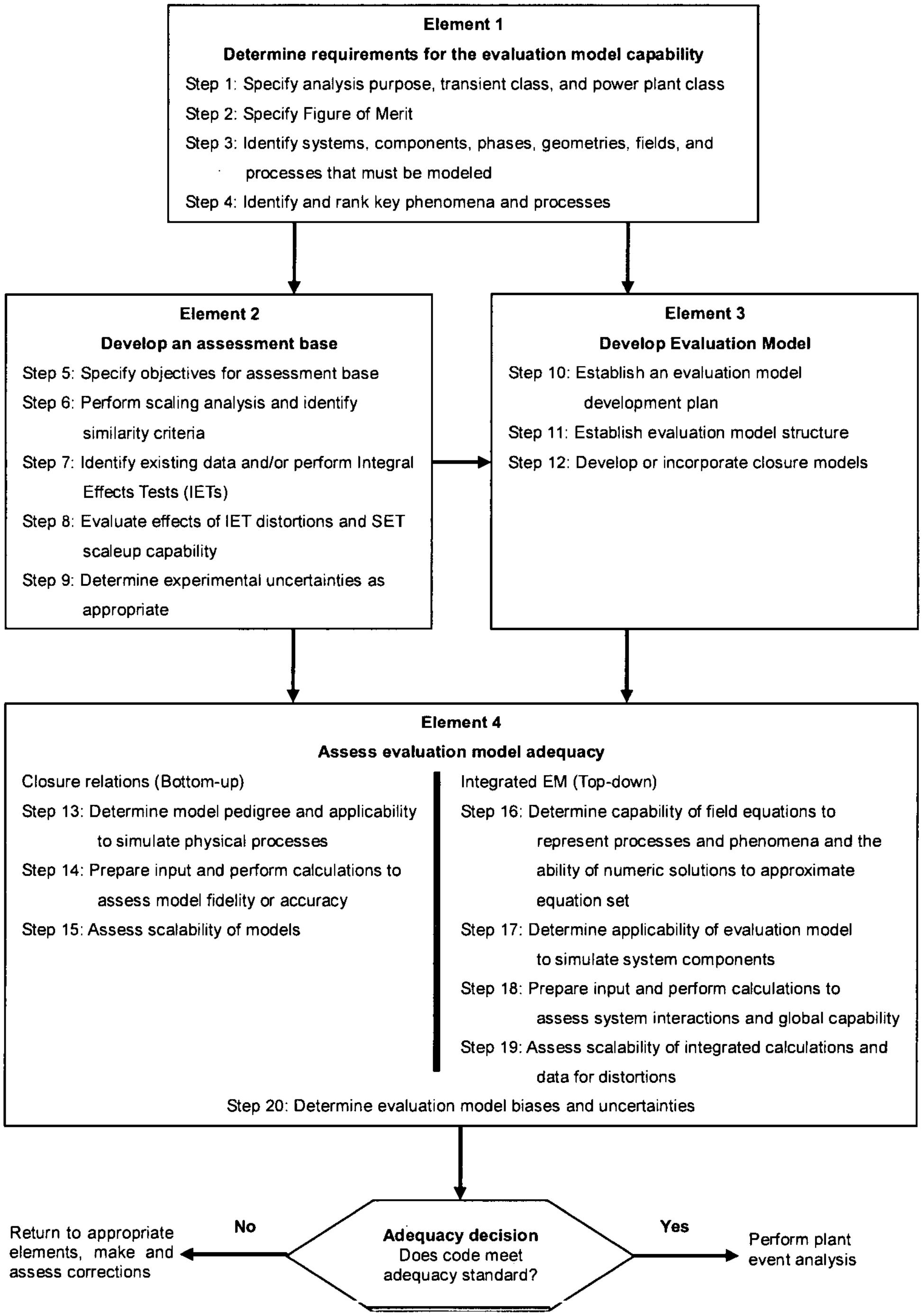

Figure A.1.2-1 Elements of Evaluation Model Development and Assessment Process (EMDAP) [44]

A.2 Radiological Analysis Method

A.2.1 Evaluation Items and Limit Values

In order to clarify the acceptability of the radiation exposure at the site boundary, the acceptance criteria are shown as follows:

1. Exclusion area boundary (EAB), 2 hr, total equivalent dose < 25 rem

2. Low-population zone (LPZ), 30 day, total equivalent dose < 25 rem

   a. Total effective dose equivalent (TEDE) = effective dose equivalent (EDE; immersion exposure) + committed effective dose equivalent (CEDE; inhalation exposure)

      The values shown above have been specified based on 10 CFR 50.34 and 10 CFR 52.47.

   b. Although the evaluation period for the LPZ is regarded as "the entire period of the fission product passage" according to 10 CFR 50.34 and 10 CFR 52.47, the evaluation was carried out in this report based on the condition of an assumed accident period of 30 days with reference to RG 4.7 [46]. RG 1.183 [28] also uses a limit of 30 days.

The calculating formulae for EDE and CEDE are as follows:

- EDE (immersion exposure)

  Evaluated by the following equation, based on the assumption of a semi-infinite cloud:

$$D_{EDE} = \sum_i K1_i \sum_j Q_{ij} \, (\chi / Q)_{ij} \qquad \text{(A.2-1)}$$

where,

$K1_i$ = EDE dose conversion factor for nuclide i (rem-m³/Ci-s)
Q_{ij} = released amount of nuclide i into the environment during time period j (Ci)
$(\chi / Q)_{ij}$ = atmospheric dispersion factor of nuclide i during time period j (s/m³)

- CEDE (inhalation exposure)

  Evaluated based on the following equation:

$$D_{CEDE} = \sum_i K2_i \sum_j Q_{ij} B_j \, (\chi / Q)_{ij} \qquad \text{(A.2-2)}$$

where,

$K2_i$	=	CEDE dose conversion factor for the nuclide i (rem/Ci)
Q_{ij}	=	released amount of nuclide i into the environment during time period j (Ci)
B_j	=	breathing rate during time period j (m^3/s)
$(\chi / Q)_{ij}$	=	atmospheric dispersion factor of nuclide i during time period j (s/m^3)

A.2.2 Establishing Factor Settings for Radiological Analysis

A.2.2.1 Dose Conversion Factor

Immersion exposure factors are established using U.S. Environmental Protection Agency (EPA) Federal Guidance Report No. 12 [50]. Inhalation exposure factors are established using EPA Federal Guidance Report No. 11 [51].

A.2.2.2 (χ / Q) (Atmospheric Dispersion Factor)

This factor was established based on RG 1.183 [28]. For this, the atmospheric diffusion model during the ground level release, which is shown in RG 1.183, was used.

The set of weather conditions for the atmospheric diffusion model during ground level release is as follows:

1. 0–8 hours

   Pasquill Type F, wind speed 1 m/sec, uniform direction

2. 8–24 hours

   Pasquill Type F, wind speed 1 m/sec, variable direction within a 22.5° sector

3. 1–4 days

   40 percent Pasquill Type D, wind speed 3 m/sec, and 60 percent Pasquill Type F, wind speed 2 m/sec, wind direction variable within a 22.5° sector

4. 4–30 days

33.3 percent Pasquill Type C, wind speed 3 m/sec, 33.3 percent Pasquill Type D, wind speed 3 m/sec, and 33.3 percent Pasquill Type F, wind speed 2 m/sec, wind direction 33.3 frequency in a 22.5° sector

Because the factor settings in RG 1.183 [28] are established for distances beyond 100 m and the EAB and LPZ are less than 100 m in the case of 4S, the values of RG 1.183 were extrapolated for this analysis.

A.2.2.3 Breathing Rate

Specified according to RG 1.183 [28].

A.2.3 Evaluation Methodology of the Amount of Radioactive Materials Released into the Environment

This evaluation was carried out using a dual-compartment migration evaluation model, which takes into account the migration and decay of radioactive materials. Figure A.2.3-1 shows the dual-compartment migration evaluation model. In this figure, the first and second compartments refer to the cover gas space and the space inside the top dome, respectively. Moreover, the leak rates of the first and second compartments refer to that of the shield plug and that of the top dome, respectively.

The amount of radioisotope (RI) inside the compartment at time t within time interval $[t_n, t_{n+1}]$ as well as the amount of RI released into the environment within $[t_n, t]$ can be expressed by the following equations.

Amount of RI inside the first compartment:

$$Q_1(t) = Q_1(t_n)e^{-\beta_{1n}t^*} \qquad \text{(A.2-3)}$$

Amount of RI inside the second compartment:

$$Q_2(t) = Q_2(t_n)e^{-\beta_{2n}t^*} + Q_1(t_n)\ell_1(1-R_{B1})\frac{e^{-\beta_{2n}t^*} - e^{-\beta_{1n}t^*}}{\beta_{2n} - \beta_{1n}} \qquad \text{(A.2-4)}$$

Here, considering the time within the interval [0, t], if $Q_1(t_n)$ is the initial amount released Q_0 and $Q_2(t_n)$ is 0, the following equation will hold:

$$Q_3(t)=\ell_2\left\{Q_0\frac{\ell_1}{\ell_2-\ell_1}\left(\frac{1-e^{-(\lambda+\ell_1)t}}{\lambda+\ell_1}-\frac{1-e^{-(\lambda+\ell_2)t}}{\lambda+\ell_2}\right)\right\} \qquad \text{(A.2-5)}$$

where,

β_{1n}	=	$\lambda + \ell_1$ (hr^{-1})
β_{2n}	=	$\lambda + \ell_2$ (hr^{-1})
t^*	=	$t - t_n$ (hr)
λ	=	decay constant (hr^{-1})
ℓ_1	=	leak rate of the 1st compartment (hr^{-1})
ℓ_2	=	leak rate of the 2nd compartment (hr^{-1})
$Q(t_n)$	=	amount of RI at time t_n (Bq)
R_{BI}	=	leak rate from 1st compartment to outside directly (hr^{-1})

In this equation, direct leakage from the cover gas space, which is the first compartment, to the environment is not considered because there is no path via which direct leakage from the cover gas space to the environment takes place. Also, because the evaluation is carried out conservatively, the plateout in the top dome, which is the second compartment, is not considered.

(Direct leakage from the first compartment to the environment is not taken into account.)

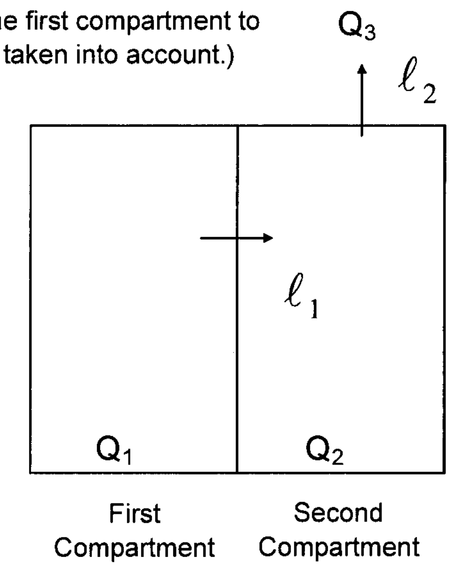

Figure A.2.3-1 Dual-Compartment Mitigation Evaluation Model