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Prevention of Severe Accidents for 4S

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TOSHIBA CORPORATION

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LIST OF ACRONYMS AND ABBREVIATIONS

4S	Super-Safe, Small and Simple
AC	air cooler
BDBA	beyond design basis accident
CDF	cumulative damage fraction
CFR	Code of Federal Regulations
CRBR	Clinch River Breeder Reactor
EMP	electromagnetic pump
GV	guard vessel
IHTS	intermediate heat transfer system
IHX	intermediate heat exchanger
IRACS	intermediate reactor auxiliary cooling system
NA	not applicable
NEI	Nuclear Energy Institute
NRC	The United States Nuclear Regulatory Commission
PL	penetration leakage
PLOHS	protected loss of heat sink
PRA	probabilistic risk assessment
PRISM	Power Reactor Innovative Small Module
RD	rupture disk
RVACS	reactor vessel auxiliary cooling system
SBI	seismic base isolation
SBO	station blackout
SG	steam generator
SR	shutdown rod
SWRPS	sodium-water reaction pressure release system
ULOF	unprotected loss of flow
ULOHS	unprotected loss of heat sink
UTOP	unprotected transient overpower

1 INTRODUCTION

A severe accident is a very low frequency event caused by multiple failures, which in worst case accident scenarios the core is damaged. Several discussions on severe accidents had been raised upon past safety evaluation of fast reactors such as Clinch River Breeder Reactor (CRBR) [1] and General Electric's Power Reactor Innovative Small Module (PRISM) [2].

During the third pre-application meeting on the Super-Safe, Small and Simple (4S) liquid metal fast reactor [3] and in the technical report subsequently submitted to the United States Nuclear Regulatory Commission (NRC) [4], Toshiba presented to the NRC staff what measures the 4S reactor has to prevent the severe accidents historically identified for previous fast reactors.

In addition, the aftermath of the event at the Fukushima Dai-ichi nuclear power plant on March 11, 2011, the NRC staff provided SECY-12-0025 "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami" [5], including external hazards to be evaluated. Subsequently, Nuclear Energy Institute (NEI) submitted NEI 12-06 [6], which presents the limited set of beyond design basis external events.

The purpose of this report is to provide 4S design characteristics and philosophy against prevention of severe accident. This report focuses on internal and external events. For the former case, to identify the more important accident scenarios comprehensively, the scenarios are selected based on the 4S preliminary probabilistic risk assessment (PRA). For the latter case, SECY-12-0025 [5] and NEI 12-06 [6] are referred to for the event selection. The representative events that have a potential to cause core damage accident and plant responses under occurrence of such an event are presented in this report.

Section 2 describes the purpose and scope of the report, and section 3 depicts the plant overview. Then, section 4 and 5 present the preventive measures against the more important severe accident scenarios initiated by internal and external events, respectively, followed by scenario selection. Finally, section 6 summarizes the main conclusions of the report.

2 PURPOSE AND SCOPE

2.1 Purpose

The purpose of this report is twofold:

1. To document the information pertinent to prevention of severe accidents.

2. To obtain feedback from the NRC staff on the presented material either in writing or in a meeting at the staff's convenience. Such feedback will be utilized by the 4S project in confirming and/or completing the plant design.

2.2 Scope

This report presents the 4S preventive measures against severe accident which is defined as a type of accident that may challenge safety systems at a level much higher than expected. The report focuses on the scenarios initiated by internal and external events. For the former case, the initiating events are selected based on the 4S preliminary PRA and the results of the safety evaluation for CRBR and PRISM [1][2]. For the latter case, the initiating events are selected according to the Commission Paper "SECY-11-0025" [5]. External human induced event such as an aircraft impact is ruled out from external event since assessment under such event was presented independently in other technical report [7]. Besides, this report focuses on the preventive measures, so the mitigative measures are not included. As presented in the Title 10, Code of Federal Regulation (CFR) part 52.137 [8], however, mitigative measures are required for light water reactors. For fast reactors, following mitigative measures can be taken into consideration; measures against energetics by recriticality, large scale sodium fire, vapor explosion, ambient pressure increase, containment vessel temperature increase, excessive heating of containment vessel by fuel debris, etc. The information on such mitigative measures shall be presented in the future upon design approval application.

3 OVERVIEW OF PLANT DESIGN

The 4S is a small size sodium cooled fast reactor which is designed for use as a power source in remote locations, and intended to operate for 30 years without refueling. A pool-type fast neutron reactor, the 4S, when coupled to power generation equipment, has an electrical output of 10MWe (30 MWt) [9].

Figure 3-1 is a schematic drawing of the overall 4S plant depicting its major components. The nuclear island is installed below grade (this includes the steam generator as well as the reactor vessel and all vital equipment).

The major components making up the reactor assembly are the reactor vessel, shielding plug, guard vessel (GV), and top dome. Structures internal to the reactor vessel include the core support structures, upper vertical baffle, two electromagnetic pumps (EMPs), and intermediate heat exchanger (IHX).

Reactor power is controlled by a movable reflector. The reactor is scrammed by lowering the (withdrawing) the reflector to the bottom of reactor via gravity. There is a cavity area filled with argon gas called “cavity can” above the reflector which enhances the increase of neutron leakage to the surrounding coolant. The shutdown rod is also inserted for a scram at the core center position to increase neutron absorption. Figure 3-2 shows the core layout, reflector, and shutdown rod.

As shown in Figure 3-3, sodium from the reactor enters and flows through the IHX where it is cooled as it heats the intermediate sodium. The primary EMPs discharge the sodium down into the bottom of the reactor. The sodium is then heated as it flows up through the core and back through the IHX.

The intermediate heat transport system (IHTS) transports heat from the primary system to the steam generator system. An EMP, located separately from the steam generator, circulates intermediate sodium through the IHX and steam generator. The double-wall tubes with a wire mesh layer between the inner and outer tube are adapted to the steam generator, which provides high reliability and significantly reduces the probability of sodium-water interaction.

The residual heat removal systems consist of the reactor vessel auxiliary cooling system (RVACS), which removes heat directly from the GV; the intermediate reactor axially cooling system (IRACS), which uses an air cooler (AC) in the intermediate sodium loop; and the main condenser cooling system. Figure 3-4 shows the schematic diagram of residual heat removal systems.

In the event that normal condenser cooling is not available, as with a loss of power supply, decay heat is removed by the RVACS and IRACS. The RVACS is a passive system. The system transports heat to the atmosphere by natural circulation of air. The flow of sodium within the reactor is aided by natural convection caused by heating in the core and cooling along the reactor vessel wall caused by the RVACS. Airflow in the RVACS is maintained by natural draft of air.

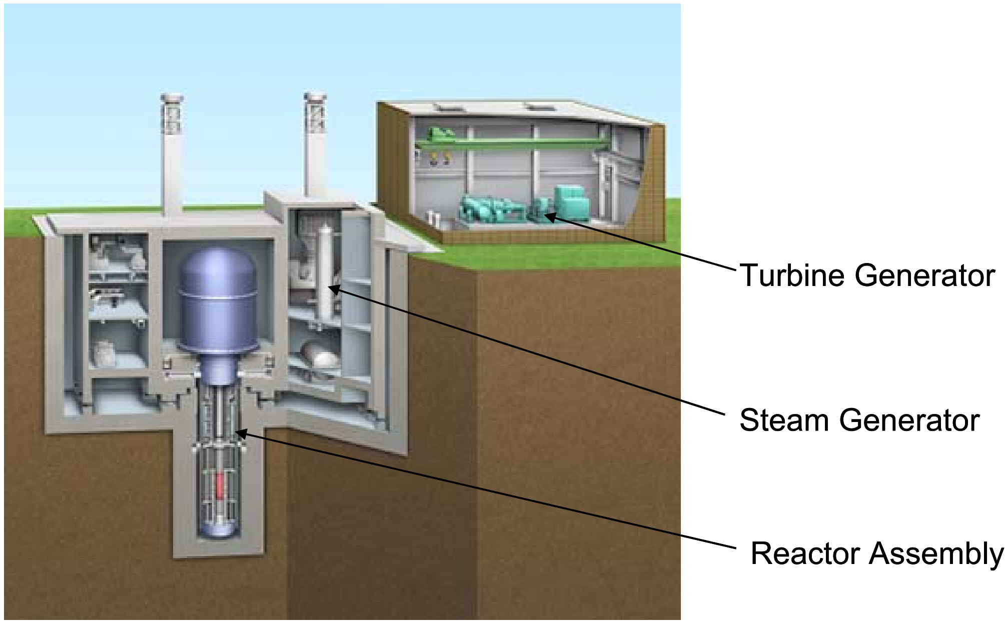


Figure 3-1. A Schematic Drawing of the Overall 4S Plant

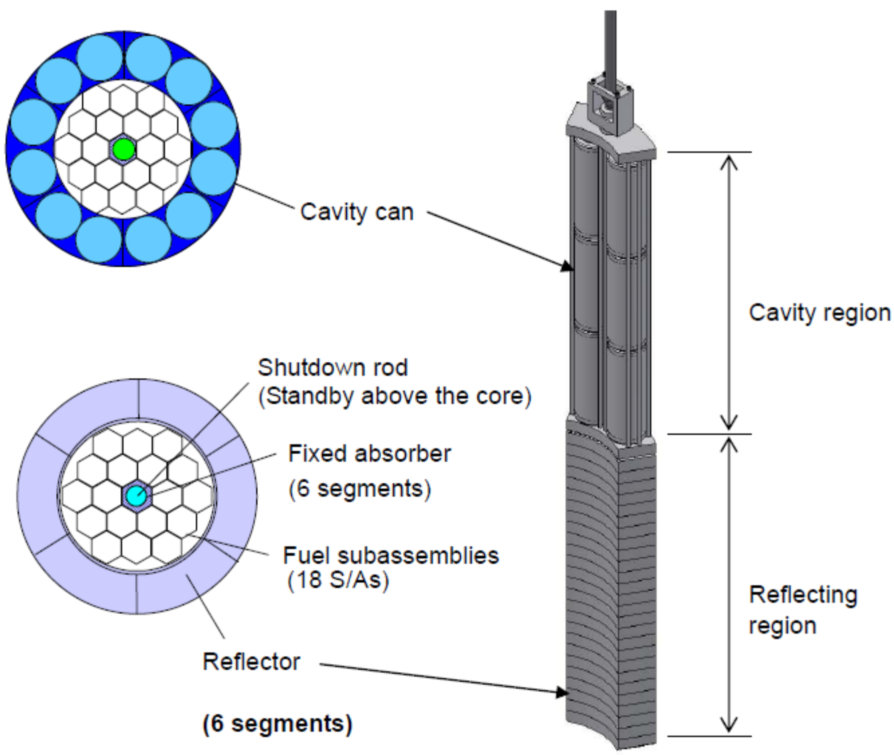


Figure 3-2. Core Layout and a Segment of Reflector

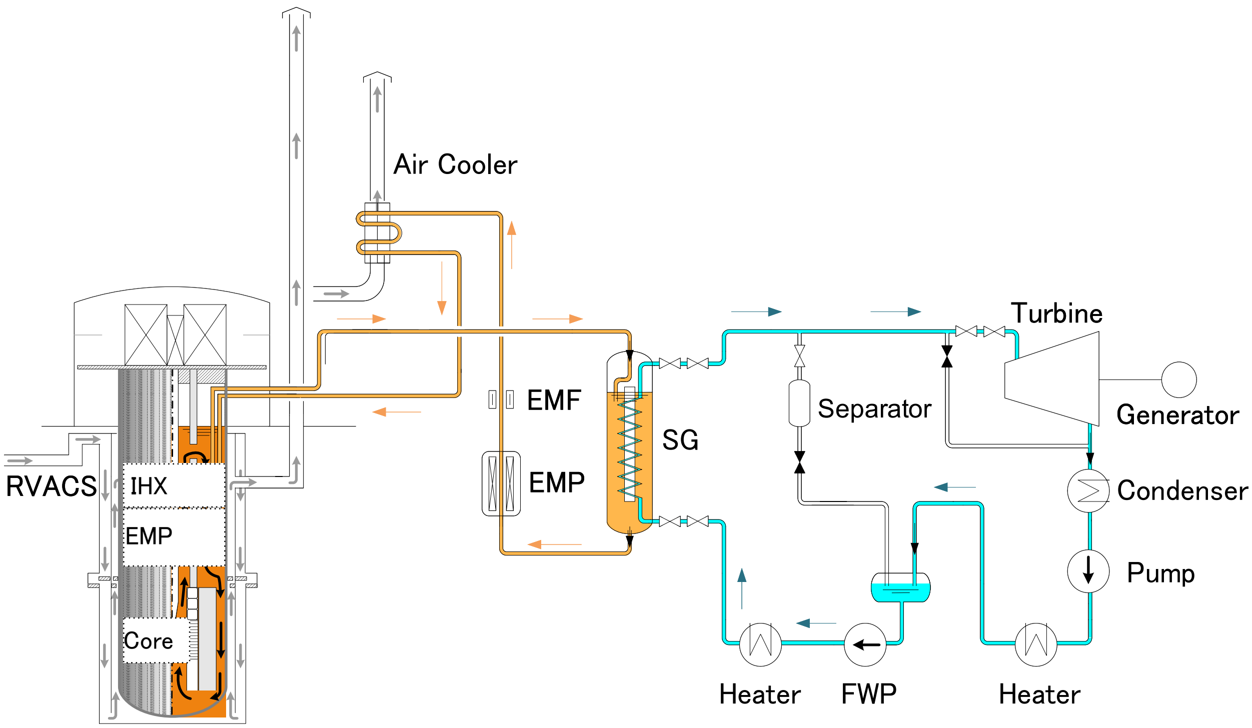


Figure 3-3. Heat Transport System Flow Diagram

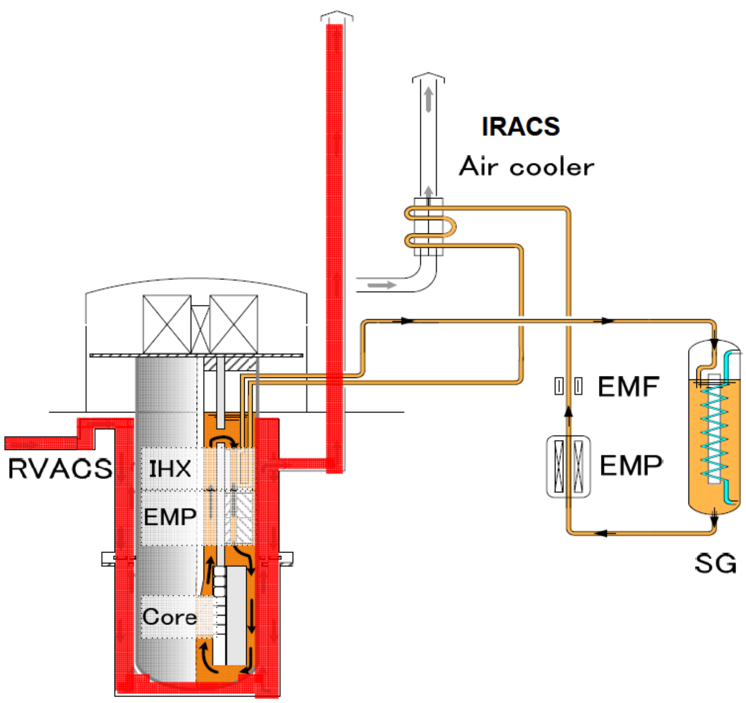


Figure 3-4. Residual Heat Removal Systems

4 PREVENTION OF SEVERE ACCIDENTS INITIATED BY INTERNAL EVENTS

4.1 Scenario Selelction (Internal Events)

This section describes the process of event selection and the results. The initiating internal events which have a potential to cause severe accident are selected based on the 4S preliminary level 1 PRA.

In the selection of important sequences, sequences with an occurrence frequency greater than 10^{-8} per reactor-year were elected to analyze. According to an NRC document [10], for instance, a core damage frequency greater than 10^{-7} per reactor-year was used as screening criteria to analyze severe accidents. In this report, the screening criteria one order of magnitude less than 10^{-7} per reactor-year was adopted. Thus, the internal events which have a potential to cause severe accident were selected from beyond design basis accidents (BDBAs) with an occurrence frequency greater than 10^{-8} per reactor-year, i.e. the events with occurrence frequency between 10^{-6} and 10^{-8} per reactor-year.

Further, the sufficiency of the event to be considered was confirmed by comparing the results of the safety evaluation of the CRBR and PRISM reactor [1][2].

As a result of the 4S preliminary level 1 PRA, the initiating events identified were classified into 14 groups to have similar characteristics. The groups are listed in Table 4.1-1, assigned with an index such as IE1, IE2, and so on. The event sequences and the event trees associated with each initiating event group are presented in the next subsection and Figure 4.1-1 through Figure 4.1-14, respectively. As shown in the figures, event sequence probabilities are grouped into four categories similar to those of the 4S technical reports on safety analysis and phenomena identification and ranking table [11][12], namely:

- Probability greater than 10^{-2} per reactor-year,
- Probability between 10^{-2} and 10^{-6} per reactor-year,
- Probability between 10^{-6} and 10^{-8} per reactor-year, and
- Probability less than 10^{-8} per reactor-year.

4.1.1 Positive Reactivity Insertion: IE1

Erroneous reflector withdrawal due to malfunction of the reflector drive mechanism would cause positive reactivity to be inserted continuously.

As shown in Figure 3-2, the cylindrical reflector has a six-segment structure that can be controlled individually and is located outside the core barrel. During normal operation, reactor core power is controlled by a movable reflector. The reflector drives are installed at each segment of reflector and consist of a combination of fine and fast adjustment mechanisms. Figure 4.1-15 shows the reflector drive mechanisms and their movement. Power cylinder (A) initiates the startup and shutdown motion of the reflector. In case of startup, power cylinder (A) raise the reflector until just before criticality is reached. The mechanical rod stop restricts the motion of reflector to limit the excess reactivity insertion. A separate power cylinder (B) for power control continues to raise the reflector from criticality to rated power. Another mechanical rod stop restricts the motion of reflector to limit the total reactivity inserted. A burnup swing compensation drive incrementally raises the reflector during reactor operation to compensate for reactivity through core life. This drive system consists of a ball screw, motor, and reduction gear, which controls the drive speed of the reflector segment and limits the maximum withdrawal rate as low as 1 ¢/day.

In this event, the temperature of the cooling system and cladding temperature increases, while reactor power is maintained almost constant by reactivity feedback. Subsequently, the IHX primary outlet temperature increases and then reaches scram set point, which result in an automatic shutdown.

In case of a failure in scram, there is a potential of occurrence of the core damage due to loss of heat sink without scram. Even if an automatic scram signal is not transmitted, however, there is enough margin to shutdown the reactor manually because the event proceeds very slowly so that abnormality in system temperature can be recognized in a timely manner by operators. After reactor shutdown, the operation mode is then shifted to the decay heat removal process by RVACS and IRACS. In case of failure of decay heat removal, the event results in protected loss of heat sink (PLOHS) event.

The event sequence No.4 of Figure 4.1-1 is categorized in BDBA. This event sequence is ruled out, however, because its consequence is less significant from the viewpoint of severe accident due to success in scram and subsequent decay heat removal.

4.1.2 Rapid Positive Reactivity Insertion: IE2

Rapid positive reactivity insertion would be caused by redundant fault of the reflector drive mechanism and reduction of neutron leakage due to sodium intrusion into the reflector cavity region filled with argon gas upon cavity can failure. In such a case, the reactivity insertion rate is considered to be in a range of 0.01 to 30 ¢/s, and scram signals are transmitted at an early stage in response to excessive neutron flux increase detected by power range and wide range neutron flux monitors.

In case of scram failure, this event results in an unprotected transient overpower (UTOP). When reactor succeeds in scram, decay heat is removed by RVACS and IRACS.

The event sequence No.13 of Figure 4.1-2 is categorized in BDBA. This event sequence is ruled out, however, because it succeeds in scram and subsequent decay heat removal. The event sequence results in either failure of scram or decay heat removal is categorized in NA since the occurrence frequency of the initiator itself is low.

4.1.3 Excessive Reactivity Insertion at Startup: IE3

Excessive positive reactivity could be added due to excessive and erroneous reflector withdrawal at reactor startup. In such a case, when reactivity insertion continues, operation of the reflector is mechanically prevented by rod stop (Figure 4.1-15). When rod stop does not function, reactor power continues to increase, and scram signals are actuated upon detection of excessive neutron flux increase by power range and wide range neutron flux monitors, which results in the same sequence as No.2 “Rapid positive reactivity insertion.”

The event sequence No.2 of Figure 4.1-3 is categorized in BDBA. This event sequence is ruled out, however, because it succeeds in scram and subsequent decay heat removal. The event sequence results in either failure of scram or decay heat removal is categorized in NA since the occurrence frequency of the initiator itself is low.

4.1.4 Local Fault: IE4

If the flow path of a fuel assembly is blocked on a large scale, it leads to boiling of the coolant at the blocked fuel assembly, eutectic failure of the cladding, and fuel melt. In such a case, reactor is manually shut down by operator upon detection of fuel cladding failure by cover gas monitoring system. Then, decay heat is removed by RVACS and IRACS, and core damage would be prevented. In case of failure of decay heat removal, the event results in PLOHS.

In case manual shutdown fails, reactor power fluctuation would occur because of local relocation of molten fuel. However, succeeded in transmission of scram signal triggered by excessive increase in neutron flux via power range neutron flux monitor and subsequent pump trip, reactor can be shutdown followed by reflector descent. Then, decay heat is removed by RVACS and IRACS, and core damage is prevented. In case of failure of decay heat removal, the event results in PLOHS. When reflector fails to descend, while trigger of scram signal and pump trip succeed, the event results in unprotected loss of flow (ULOF). When transmission of scram signal fails, the event results in loss of core coolable geometry because the pumps do not trip and rated core flow is maintained during event progression.

This event is ruled out because there are no event sequences categorized in BDBA (Figure 4.1-4). Further, core melt and rapid eutectic reaction would not be expected to occur upon local faults categorized in BDBA with occurrence frequency greater than 10^{-8} per reactor-year [12].

4.1.5 Loss of Primary Flow: IE5

Loss of primary flow is caused by failure of the primary EMPs. There are two types of failure which result from a problem at (a) the power supply system and (b) the EMP itself. In the former case, the reactor is automatically shutdown by scram signals transmitted upon excessive voltage reduction of the EMP electric power supply, increase in power range neutron flux, and decrease in the IHX primary outlet temperature. In the latter case, scram signals are triggered by excessive increase of neutron flux detected by power range neutron flux monitor and decrease in the IHX primary outlet temperature. In these events, when scram fails, the feed water pump trips by interlock signal triggered by temperature decrease in the intermediate cooling system. Subsequently, reactor power decreases passively due to negative reactivity feedback followed by increase of core inlet temperature. In case of failure of the feed water pump trip, although core power decreases temporarily due to negative reactivity feedback, it returns to the rated condition because primary coolant flow is in natural circulation mode, which could cause a number of fuel pin failure.

Figure 4.1-5(a) and Figure 4.1-5(b) show the event tree of loss of primary flow caused by failure of the power supply system and the EMP itself, respectively.

The event sequence No.4, No.18 and No.25 of Figure 4.1-5(a) and No.4 and No.24 of Figure 4.1.5(b) are categorized in BDBA. This event sequence No.4 and No.18 of Figure 4.1-5(a) are ruled out, however, because it succeeds in scram and subsequent decay heat removal. Other event sequence has similar characteristics, so event No.4 of Figure 4.1-5(b) is selected as representative one.

4.1.6 Sudden Loss of Primary Flow: IE6

Figure 4.1-16 shows a flow coastdown system of EMP. The EMPs are equipped with a motor-generator set with a flywheel as backup power to compensate for the lack of fluid inertia when the normal power supply is stopped. The MG set can supply sufficient power to maintain an adequate flow coastdown during a power loss. The EMP exciter sends an electric current to the rotor of the generator at the time of the flow coastdown and startup.

Sudden loss of primary flow is caused by failure of flow coastdown followed by pump trip. In such a case, scram signals are transmitted due to excessive increase of neutron flux detected by power range and wide range neutron flux monitors and temperature increase in the IHX primary outlet. Subsequently, decay heat is removed by RVACS and IRACS.

When scram fails, feed water pump trips by interlock signal triggered by temperature decrease in the intermediate cooling system. Then, temperature of the entire cooling system increases due to loss of heat removal by SG, and negative reactivity feedback causes reactor power to decrease passively to the decay heat power level. The event results in ULOF when passive shutdown by negative reactivity feedback fails.

The event sequence No.2, No.9, and No.17 of Figure 4.1-6 are categorized in BDBA. This event sequences are ruled out, however, because it succeeds in scram and subsequent decay heat removal.

4.1.7 Loss of One Primary Pump Flow: IE7

Loss of one primary pump flow occurs when one of the two primary EMPs trips, which results in decrease in core flow rate. In such a case, temperature difference between core inlet and outlet would be expected to increase. Core damage is prevented, however, because the fuel cladding is expected to be intact and core integrity is designed to be maintained under such a condition.

Transmission of a scram signal triggered by voltage reduction of the failed EMP depends on the cause of pump trip. Even if scram fails, however, remaining one primary pump keeps operation, and there would be no potential to lead core damage. Operators can manually shut the reactor down after detecting abnormality.

The event sequences No.4 and No.8 of Figure 4.1-7 are categorized in BDBA. They are ruled out, however, because as aforesaid there would be no potential of core damage.

4.1.8 Reactor Vessel Leak: IE8

When reactor vessel leak occurs, scram signal is transmitted due to decrease in coolant level in the reactor vessel, and decay heat is removed by RVACS and IRACS. Even if scram fails, guard vessel outside the reactor vessel keep an acceptable coolant level required for reactor operation. In case leakage from guard vessel occurs, the coolant level in the reactor vessel is lost, which would result in loss of reactor coolant level type core damage.

This event is ruled out because there are no event sequences categorized in BDBA (Figure 4.1-8).

4.1.9 Loss of Intermediate Flow: IE9

Loss of intermediate flow is caused by failure of the intermediate EMP. Decrease in the intermediate system flow rate leads to temperature increase of the IHX primary outlet and cold coolant plenum in the reactor vessel, which results in transmission of scram signal triggered by increase in the IHX primary outlet temperature. Then, decay heat is removed by RVACS and IRACS.

If scram fails, the reactor power decreases due to negative reactivity feedback to 60 to 70 percent of the rated value and is maintained as is. Then, the reactor core temperature is kept at a lower state than that of the rated value and reactor operation continues in quasi-stable mode. Thus, there would be no potential to lead to core damage in a certain period during which operators can manually shut the reactor down after detecting abnormality.

The event sequence No.16 and No.21 of Figure 4.1-9 are categorized in BDBA. They are ruled out, however, because the former succeeds in scram and subsequent decay heat removal, and as aforesaid, the latter has no potential to cause core damage.

4.1.10 Intermediate Heat Transfer System Leak: IE10

When leakage occurs on a large scale in the intermediate heat transfer system, intermediate pump and feed water pump trip, and then scram signal is transmitted due to excessive voltage reduction of the EMP electric power supply followed by primary EMP trip or, otherwise, increase in the IHX primary outlet temperature. The scram signal actuates the shutdown rod descent. Sodium leakage amount is limited by subsequent emergency drain of the intermediate coolant. 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.

In case scram fails, power flow mismatch leads to the increase of the coolant temperature in the reactor vessel, and then reactor power decreases passively to the decay heat power level by negative reactivity feedback.

Although it is extremely unlikely, when RVACS fails to remove decay heat, the event results in ULOF type core damage.

The event sequence No.3 and No.16 of Figure 4.1-10 are categorized in BDBA. The event sequence No.3 is ruled out, however, because it succeeds in scram and subsequent decay heat removal.

4.1.11 Loss of Heat Removal by Water/Steam System: IE11

Loss of heat removal by water/steam system occurs when its flow rate is lost. In such a case, scram signal is transmitted due to excessive reduction in electric power supply voltage of the primary EMP followed by pump trip or, otherwise increase in the IHX primary outlet temperature. Even if automatic scram fails, the reactor power decreases passively to the decay heat power level due to negative reactivity feedback. In any case, decay heat is removed by RVACS. If reactor shutdown fails, the event results in unprotected loss of heat sink (ULOHS) type of core damage.

The event sequence No.16 and No.22 of Figure 4.1-11 are categorized in BDBA. The event sequence No.16 is ruled out, however, because it succeeds in scram and subsequent decay heat removal.

4.1.12 Steam Generator Tube Leak: IE12

The SG tubes are double-wall with a wire mesh layer between the inner and outer tubes. Double-wall tubes are used to prevent a sodium-water reaction during a steam generator tube failure (Figure 4.1-17).

The wire mesh layer, as it is structurally separating the inner and the outer tube, will reduce the failure probability of a tube caused by a common factor because a crack in the tube in one side does not propagate directly to the other tube. In addition, the wire mesh layer has a high permeability and is filled with a third fluid, such as helium, to provide continuous leak detection. This continuous monitoring can detect a failure of one of the double-wall tubes and prompt operator action before a second failure of the other tube wall results in a sodium-water reaction. This promotes high reliability of the overall system. Figure 4.1-18 shows the leak detection system for the SG. During normal operation, helium plenum pressure is kept constant at an intermediate value below that of the water-steam side and above that of the sodium side.

In case of inner tube failure, moisture leaks into the helium in the wire mesh layer. This leaking moisture migrates into the helium plenum from the feedwater header or the steam header and displaces helium. The moisture migrating into the helium plenum is detected by a moisture gauge in the sampling system because of increasing moisture concentration. In addition, the abnormality can be detected by pressure increasing in the helium plenum. The sampling system is composed of a cooler, moisture gauge, circulator, and heater in both the feedwater side and steam side.

The outer tube leak detection system is designed to detect a penetrating failure of the outer tube. If an outer tube failure occurs, the helium in the wire mesh layer will leak into the intermediate sodium loop. The pressure decrease of the helium plenum caused by the outflow of helium is detected as an outer tube failure. A small portion of helium flowing into the sodium is dissolved in the sodium and the rest migrates into the cover gas area as a gas. The outer tube leak detection system consists mainly of SG cover gas monitoring for helium content in the cover gas to detect an initial small-scale and subsequent medium-scale helium leakage promptly.

This system is designed to prevent double tube failure, which would result in a sodium-water reaction, by permitting shutdown of the plant in case of prompt detection of a small-scale and medium-scale failure of either side of the double-wall tube. If a sodium-water reaction caused by double tube failure occurs, the cover gas pressure system and rupture disk failure detection system are used as an alarm signal and interlock signal, respectively.

In case either side of tube leakage is detected, the feed water pump trips and the water/steam system stars blow down triggered by interlock signal. Then, scram signal is transmitted due to excessive increase in the IHX primary outlet temperature, and RVACS removes decay heat.

Penetration leakage would occur in case detection of either side of tube leakage fails or delays. In such a case, the pressure of the intermediate system increases due to accumulation of hydrogen gas, which causes the rupture disk to burst. The reactor scrams and feedwater pump trips upon rupture disk burst signal, and the water/steam system will start to blow down. Subsequently, reactor scrams followed by scram signal triggered by excessive increase in the IHX primary outlet temperature.

Further, even if scram fails, the reactor power decreases to decay heat power level due to negative reactivity feedback, and decay heat is removed by RVACS.

Although it is extremely unlikely, when rupture disk fails or delays to function upon penetration leakage, boundary of the IHX could be damaged. Subsequently, pressurized intermediate coolant would enter the reactor vessel which may cause damage of the reactor vessel and guard vessel due to the effect of reaction products.

The event sequence No.3, No.16, No.19, No.41, and No.63 of Figure 4.1-11 are categorized in BDBA. The event sequence No.16 is ruled out, however, because it succeeds in scram and subsequent decay heat removal. However, sodium water reaction which is a characteristic phenomenon of SG tube rupture accident does not occur in the event No.3, No. 19, and No41, which would be expected to have no potential to cause core damage due to success in scram and subsequent decay heat removal, and No.16 which result in ULOF and are encompassed in the scenario initiated by “loss of primary flow (IE5)”. Hence, from the view point of occurrence of sodium water reaction, the event sequences No. 63 is selected.

4.1.13 Loss of Offsite Power: IE13

The event sequence initiated by loss of offsite power is equivalent to that of “loss of primary flow (IE5)” event except that the primary EMPs, intermediate EMP, and feedwater pump trip simultaneously upon loss of power supply and that scram signal is triggered by voltage reduction of normal bus bar. After simultaneous trip of the pumps followed by loss of offsite power, the reactor is automatically shutdown by scram signals triggered by excessive reduction of normal bus bar voltage and increase of IHX primary outlet temperature. Then, decay heat is removed by RVACS and IRACS.

Even if scram fails, reactor power decreases to the decay heat power level passively due to negative reactivity feedback via increase in the reactor core temperature. In case such reactor shutdown fails, this event results in the ULOF type core damage.

The event sequences No.5, No.12, and No.18 of Figure 4.1-13 are categorized in BDBA. They are ruled out, however, because they succeed in scram and subsequent decay heat removal except the event sequence No.18. Further, the event sequence No.18 is ruled out because it is encompassed in “loss of primary flow (IE5)” event.

4.1.14 Spurious Shutdown: IE14

The reactor could be shutdown spuriously due to transmission of an erroneous scram signal or malfunction of the reflector or shutdown rod. After reactor shutdown, decay heat is removed by RVACS and IRACS.

This event is ruled out because there are no event sequences categorized in BDBA (Figure 4.1-14). Moreover, this event is encompassed in “loss of primary flow (IE5)” event.

Table 4.1-2 summarizes the result of the scenario selection initiated by internal events.

Furthermore, nine initiators were identified to have the more significant potential to cause core damage from the safety evaluation of the past fast reactors such as CRBR and PRISM [1][2].

Those initiators, however, are confirmed to be encompassed into the initiating event groups that are identified by 4S preliminary PRA as shown in Table 4.1-2 except “blockage of flow path of RVACS” that had been discussed during the safety evaluation for the PRISM reactor [2]. In the 4S preliminary PRA, the blockage of flow path of RVACS is not considered as an initiator, but it is taken into consideration in the process of establishing event trees as a system response of the RVACS. Considering the historical significance, the consequence of the blockage of flow path of RVACS is discussed in subsection 4.2.5.

Table 4.1-1. Initiating Event Group

No.	Item
IE1	Positive reactivity insertion
IE2	Rapid positive reactivity insertion
IE3	Excessive reactivity insertion at startup
IE4	Local fault
IE5	Loss of primary flow
IE6	Sudden loss of primary flow
IE7	Loss of one primary pump flow
IE8	Reactor vessel leak
IE9	Loss of intermediate flow
IE10	Intermediate heat transfer system leak
IE11	Loss of heat removal by water/steam system
IE12	Steam generator tube leak
IE13	Loss of offsite power
IE14	Spurious shutdown

Table 4.1-2. Result of the Scenario Selection Initiated by Internal Event

No.	Initiating Event	Selected Scenario	Remarks
IE1	Positive reactivity insertion	NA	All control rod withdrawal without scram, Failure of core support structure
IE2	Rapid positive reactivity insertion	NA	
IE3	Excessive reactivity insertion at startup	NA	
IE4	Local fault	NA	Fuel loading error, Inlet blockage of subassemblies, Gas passage in the core
IE5	Loss of primary flow	IE5(a)-No.4	-
IE6	Sudden loss of primary flow	NA	Sudden loss of flow without scram
IE7	Loss of one primary pump flow	NA	-
IE8	Reactor vessel leak	NA	-
IE9	Loss of intermediate flow	NA	-
IE10	Intermediate heat transfer system leak	IE10-No.16	-
IE11	Loss of heat removal by water/steam system	IE11-No.22	-
IE12	Steam generator tube leak	IE12-No.63	Sodium-water reaction
IE13	Loss of offsite power	NA	-
IE14	Spurious shutdown	NA	-

Note)
The initiators to have the more significant potential to cause core damage from CRBR and PRISM are shown in the remarks column [1][2].

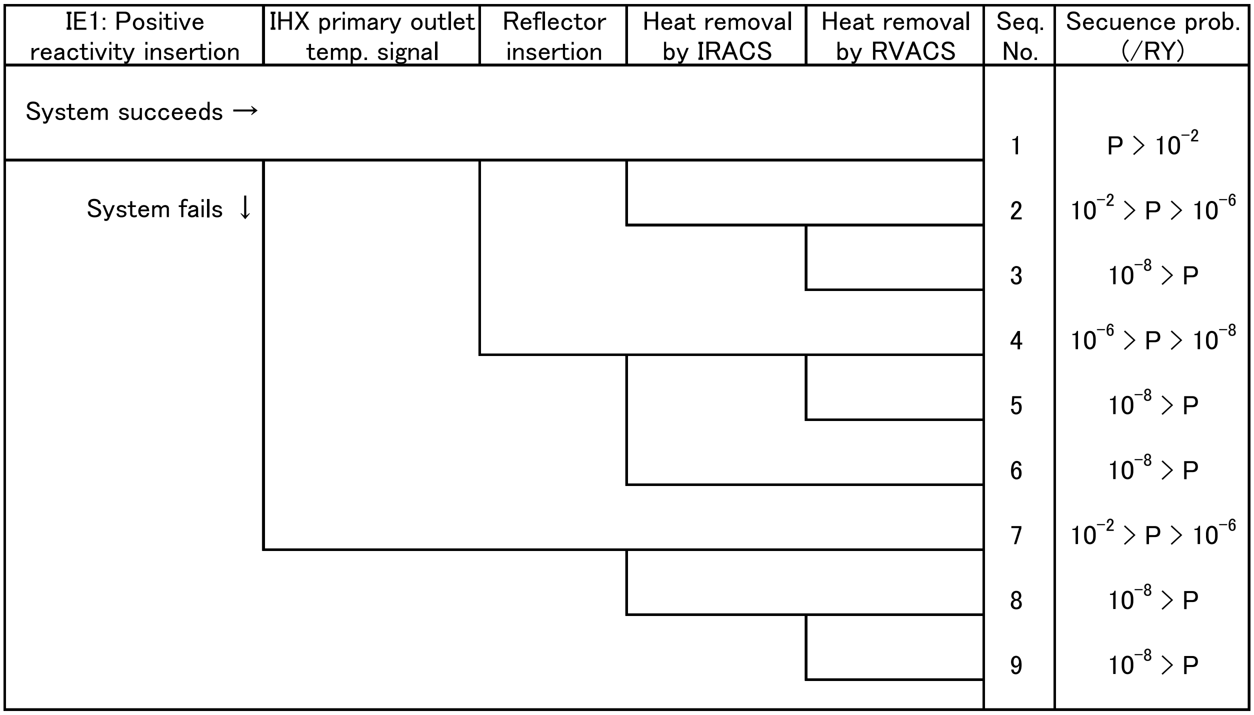


Figure 4.1-1. Event Tree IE1: Positive Reactivity Insertion

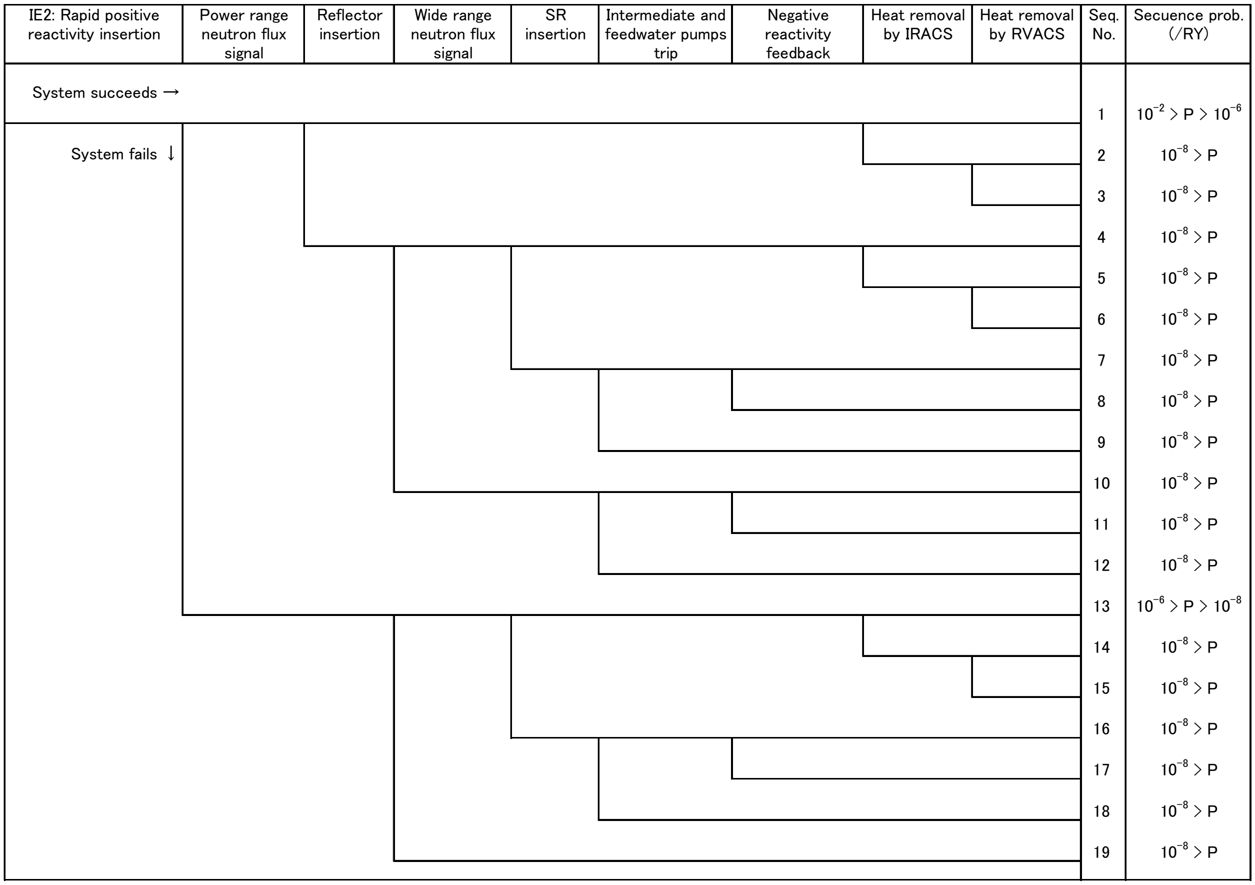


Figure 4.1-2. Event Tree IE2: Rapid Positive Reactivity Insertion

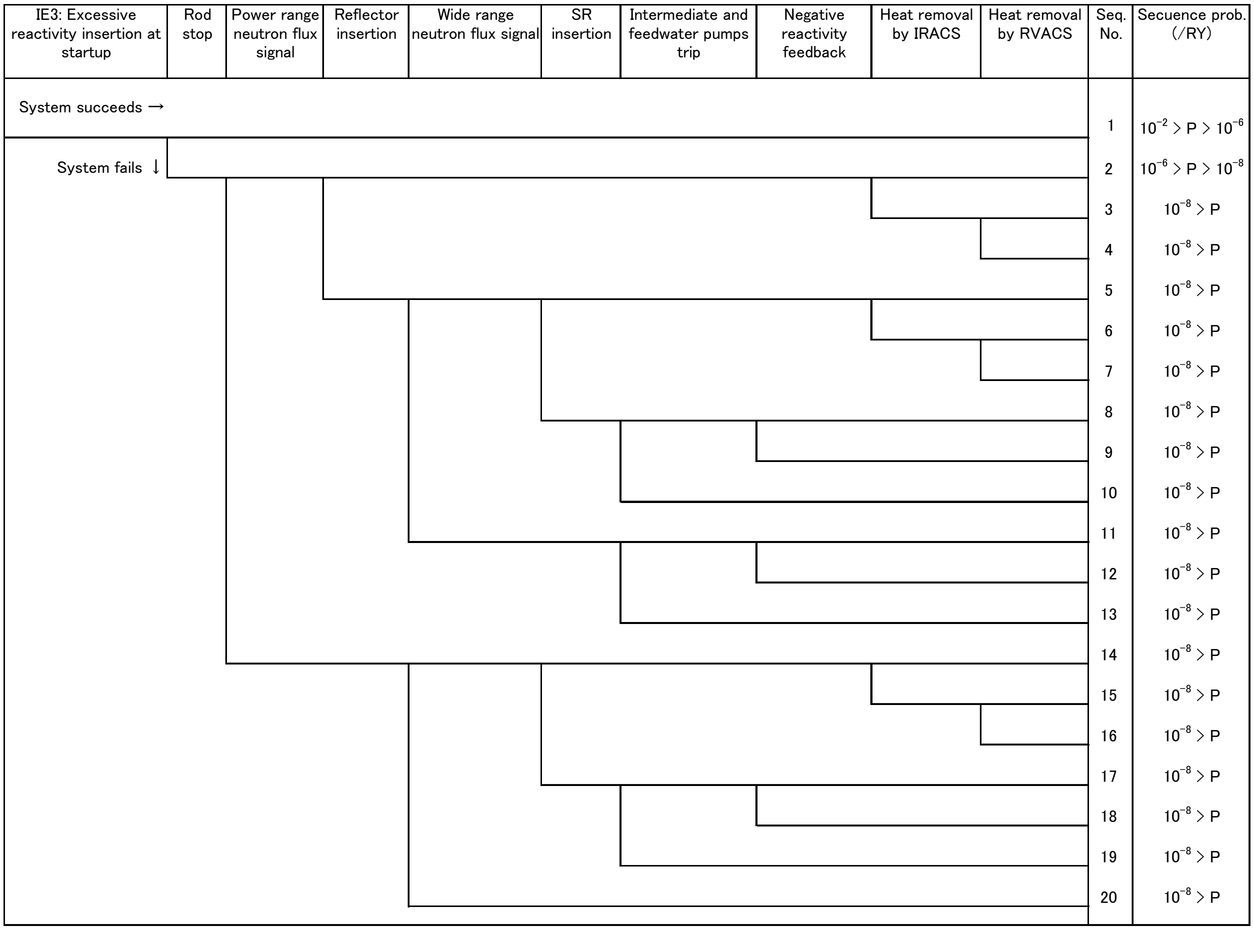


Figure 4.1-3. Event Tree IE3: Excessive Reactivity Insertion at Startup

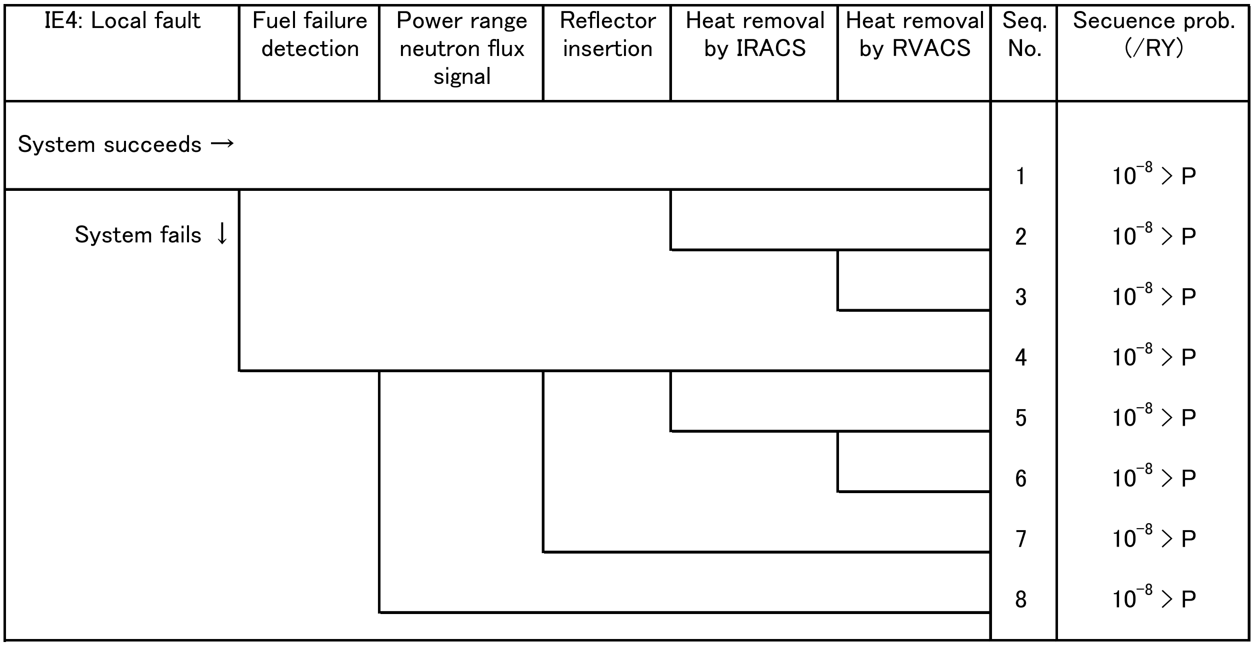


Figure 4.1-4. Event Tree IE4: Local Fault

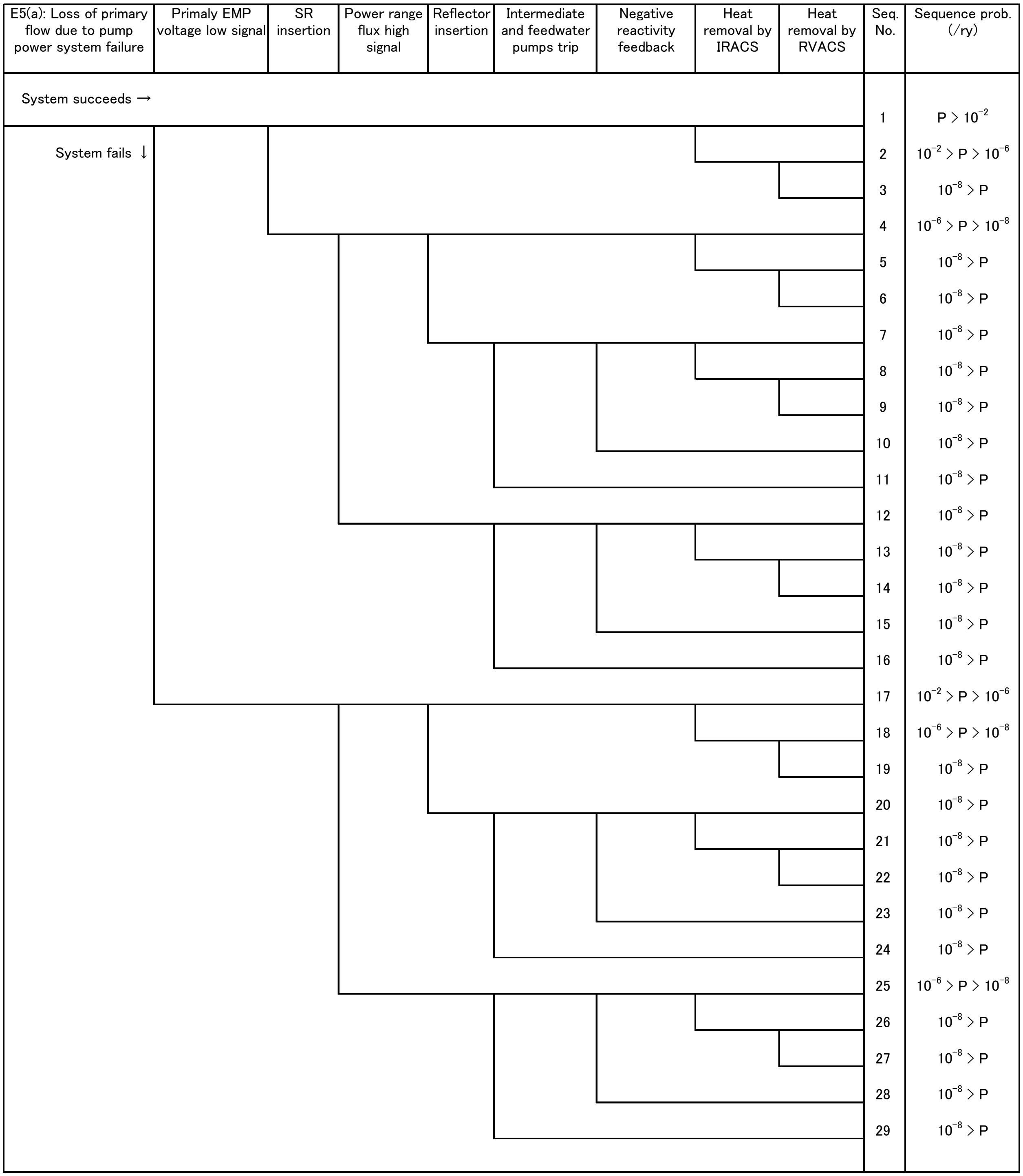


Figure 4.1-5(a). Event Tree IE5: Loss of Primary Flow Caused by Failure of the Power Supply System of EMP

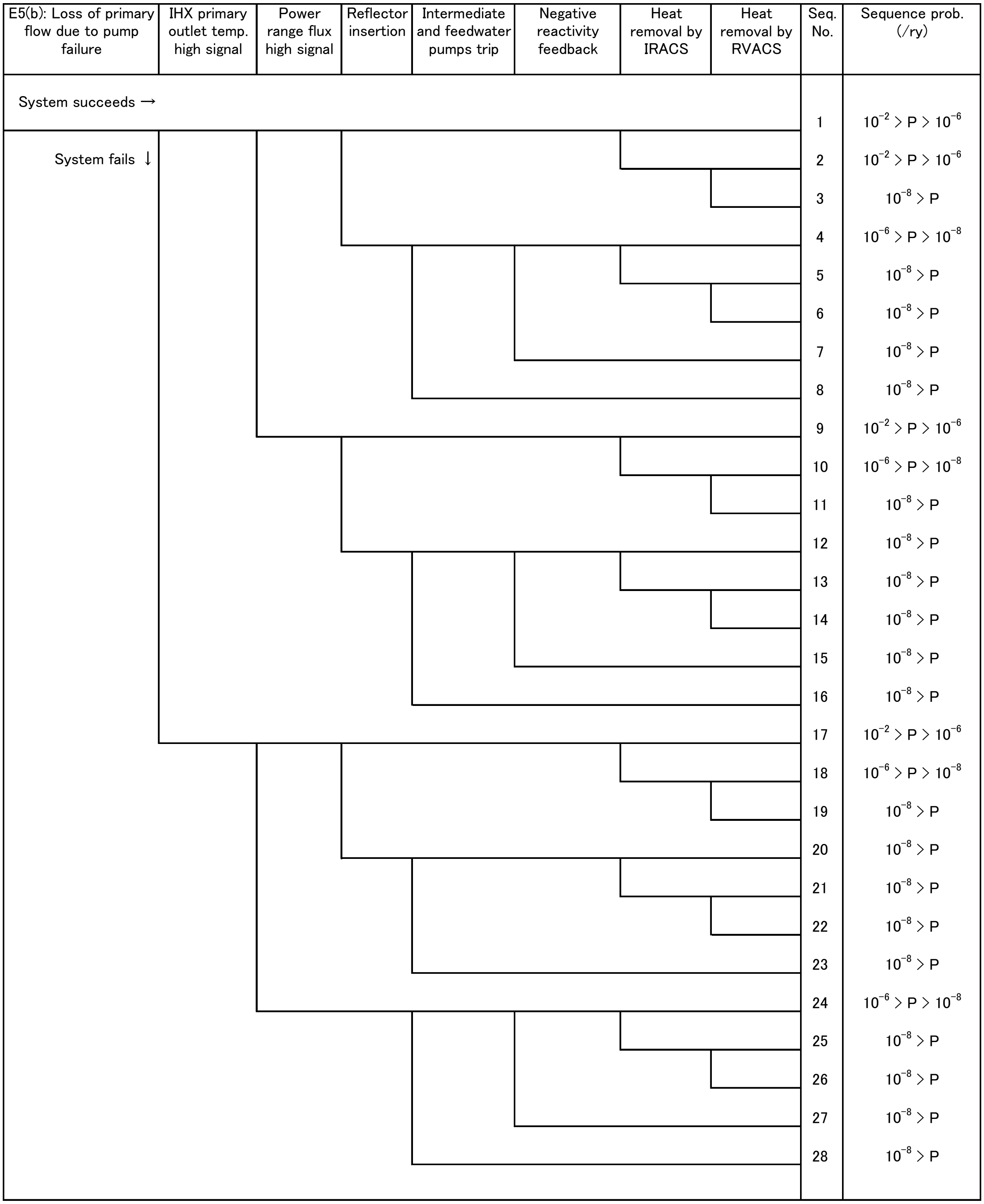


Figure 4.1-5(b). Event Tree IE5: Loss of Primary Flow Caused by Failure of EMP

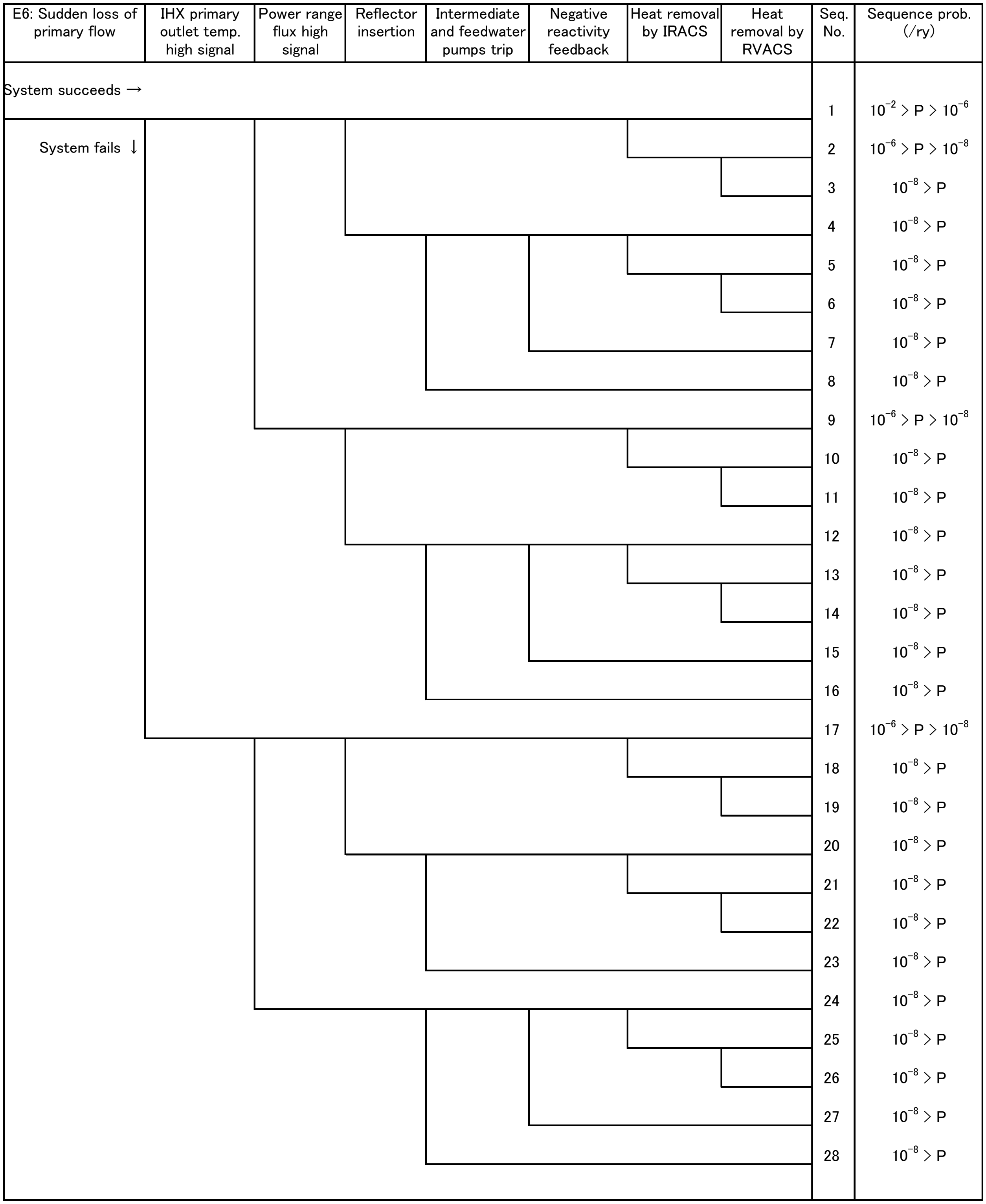


Figure 4.1-6. Event Tree IE6: Sudden Loss of Primary Flow

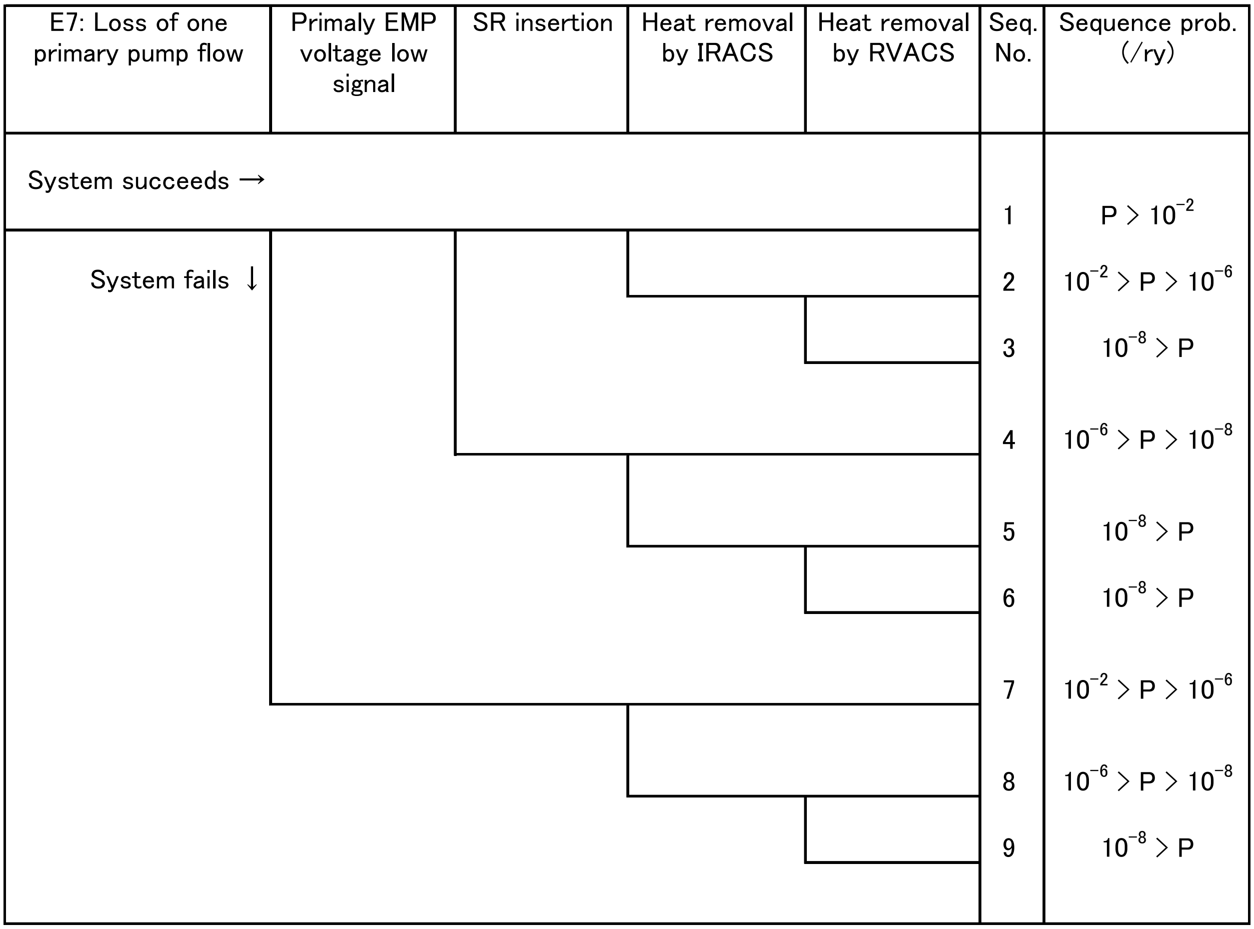


Figure 4.1-7. Event Tree IE7: Loss of One Primary Pump Flow

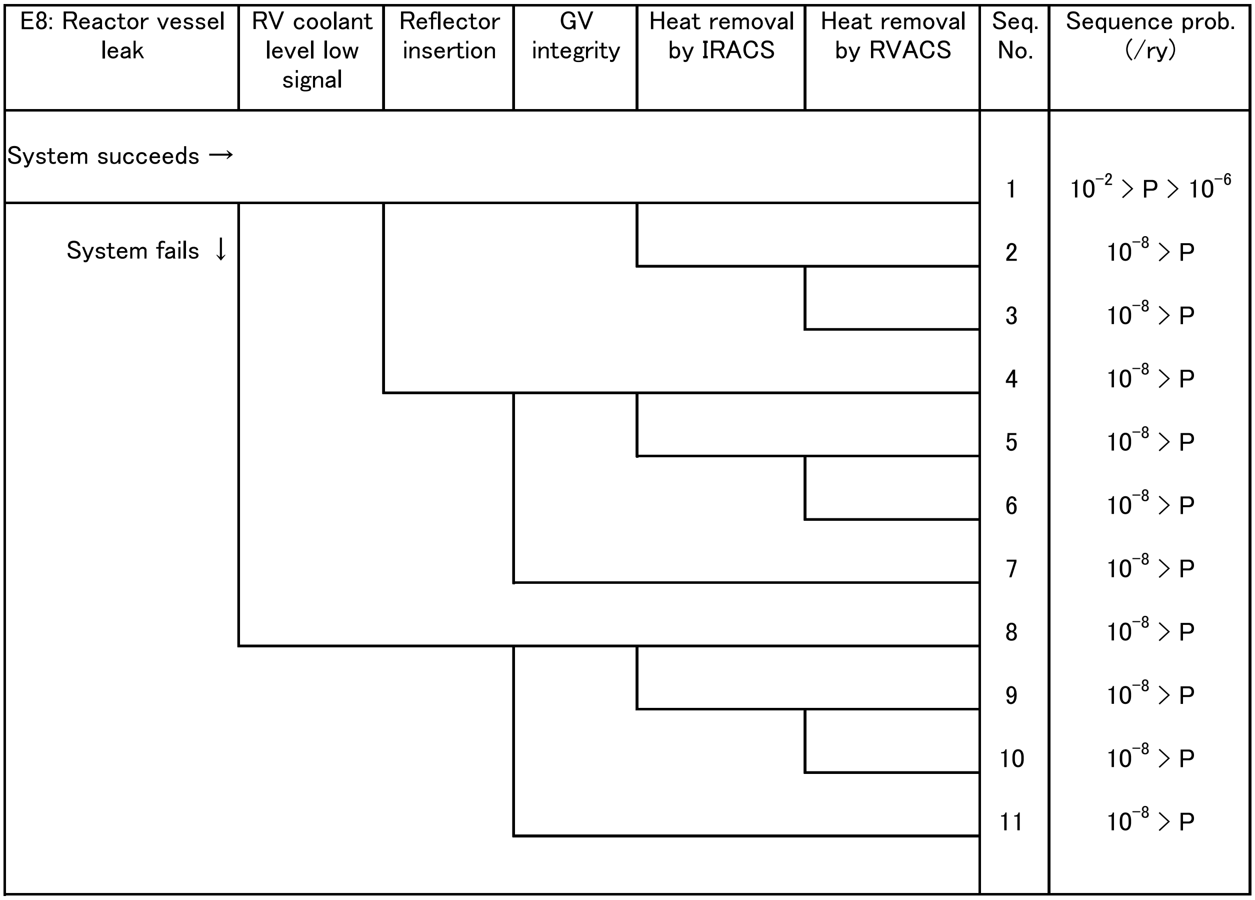


Figure 4.1-8. Event Tree IE8: Reactor Vessel Leak

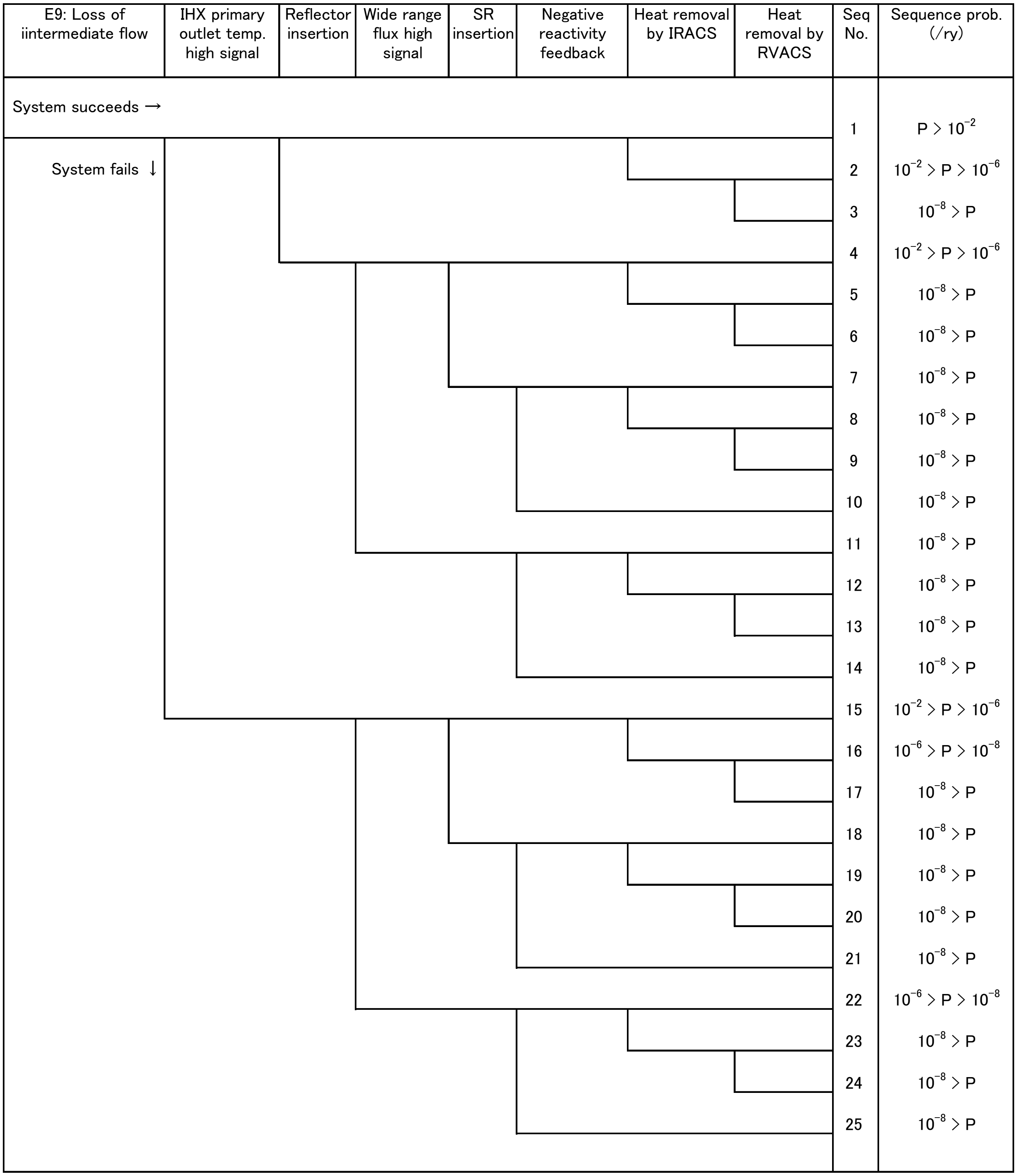


Figure 4.1-9. Event Tree IE9: Loss of Intermediate Flow

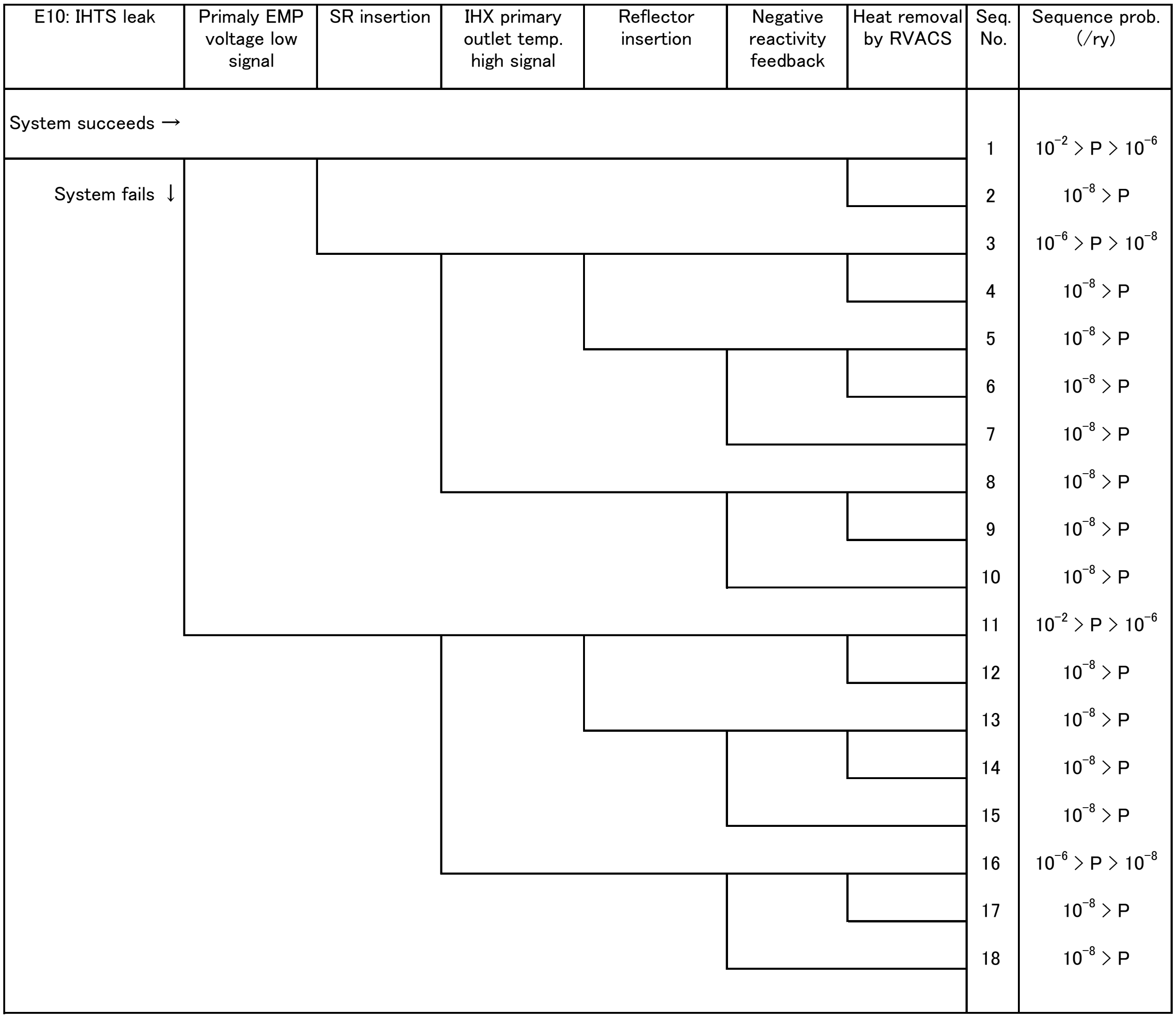


Figure 4.1-10. Event Tree IE10: Intermediate Heat Transfer System Leak

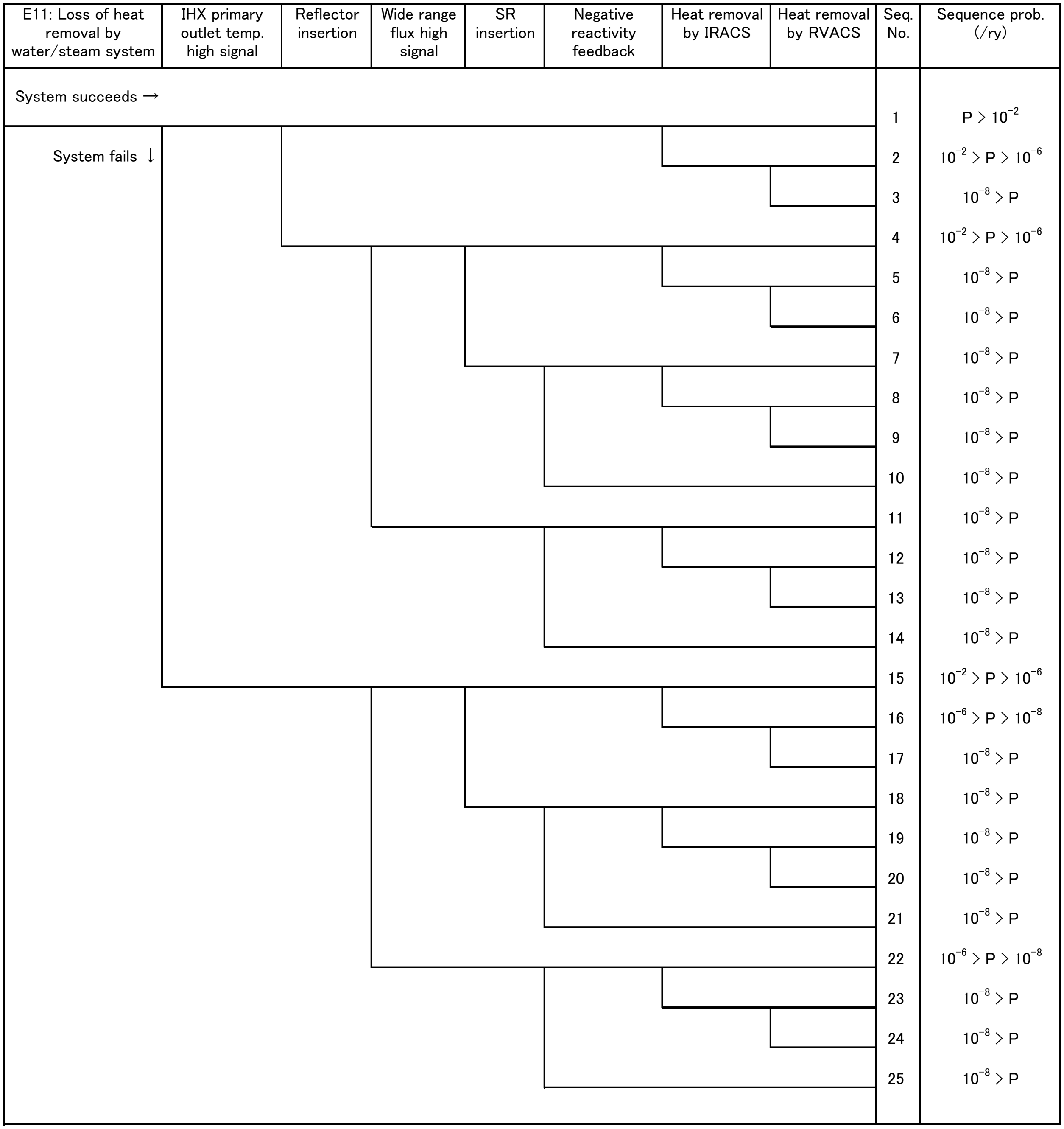


Figure 4.1-11. Event Tree IE11: Loss of Heat Removal by Water/Steam System

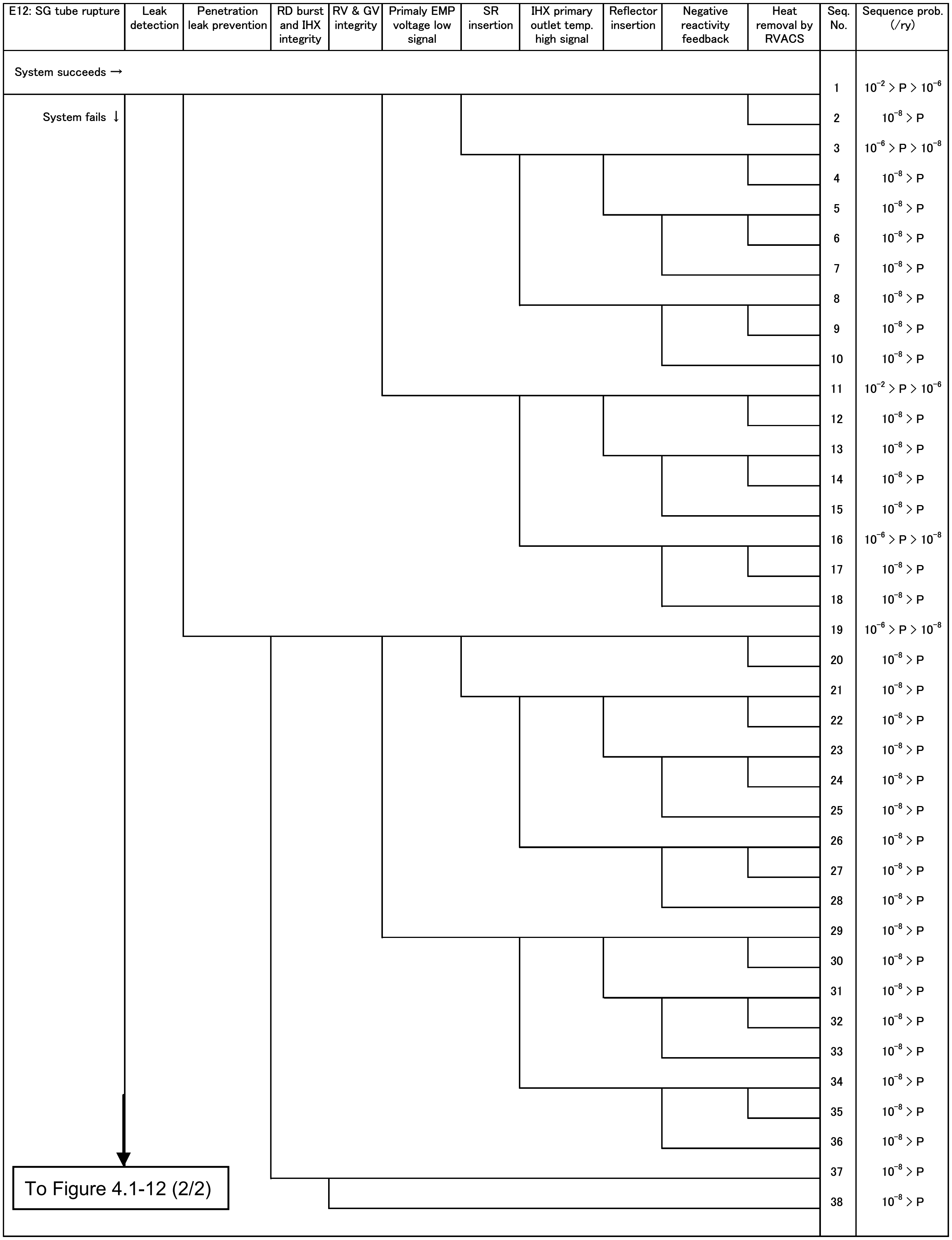


Figure 4.1-12. Event Tree IE12: Steam Generator Tube Leak (1/2)

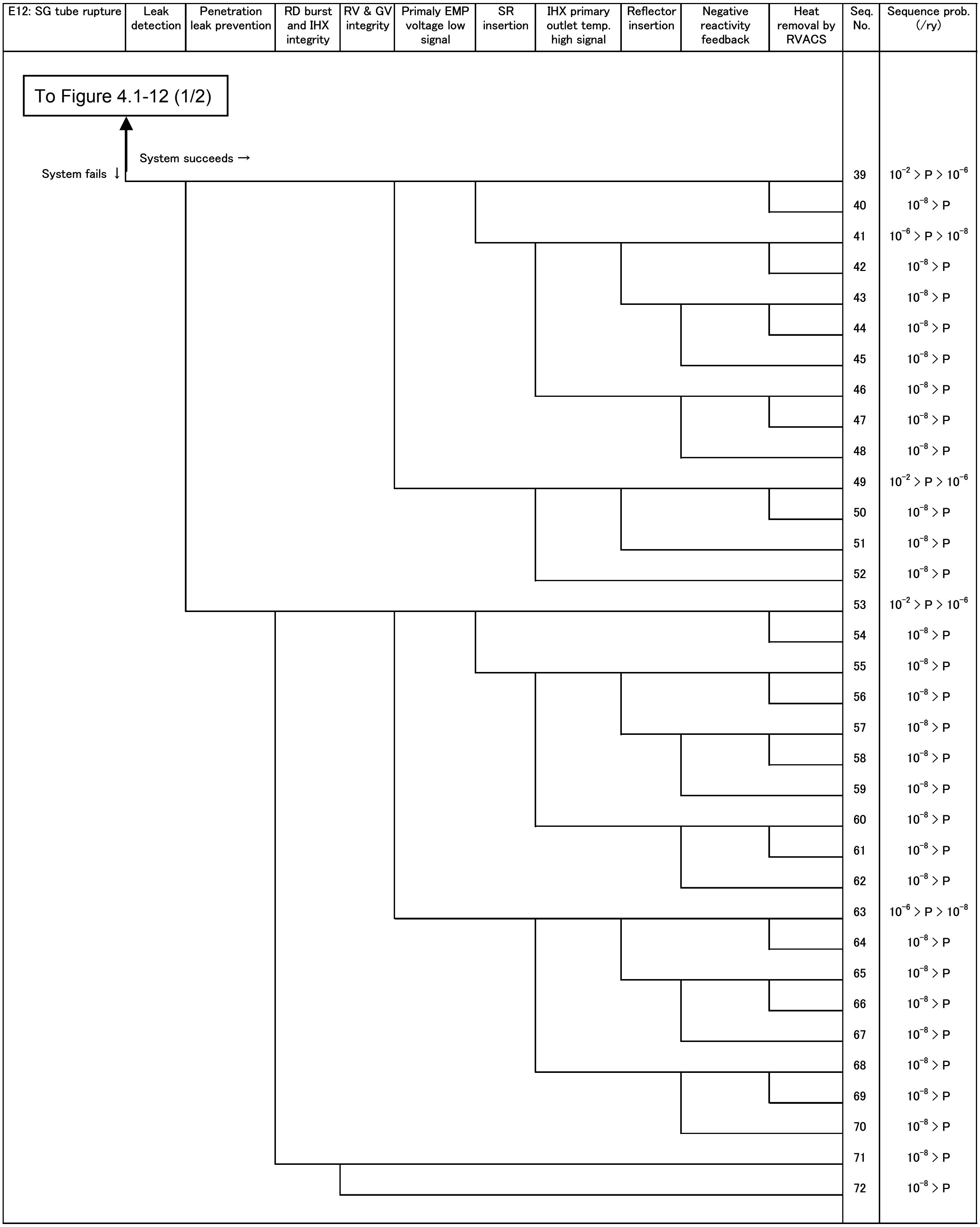


Figure 4.1-12. Event Tree IE12: Steam Generator Tube Leak (2/2)

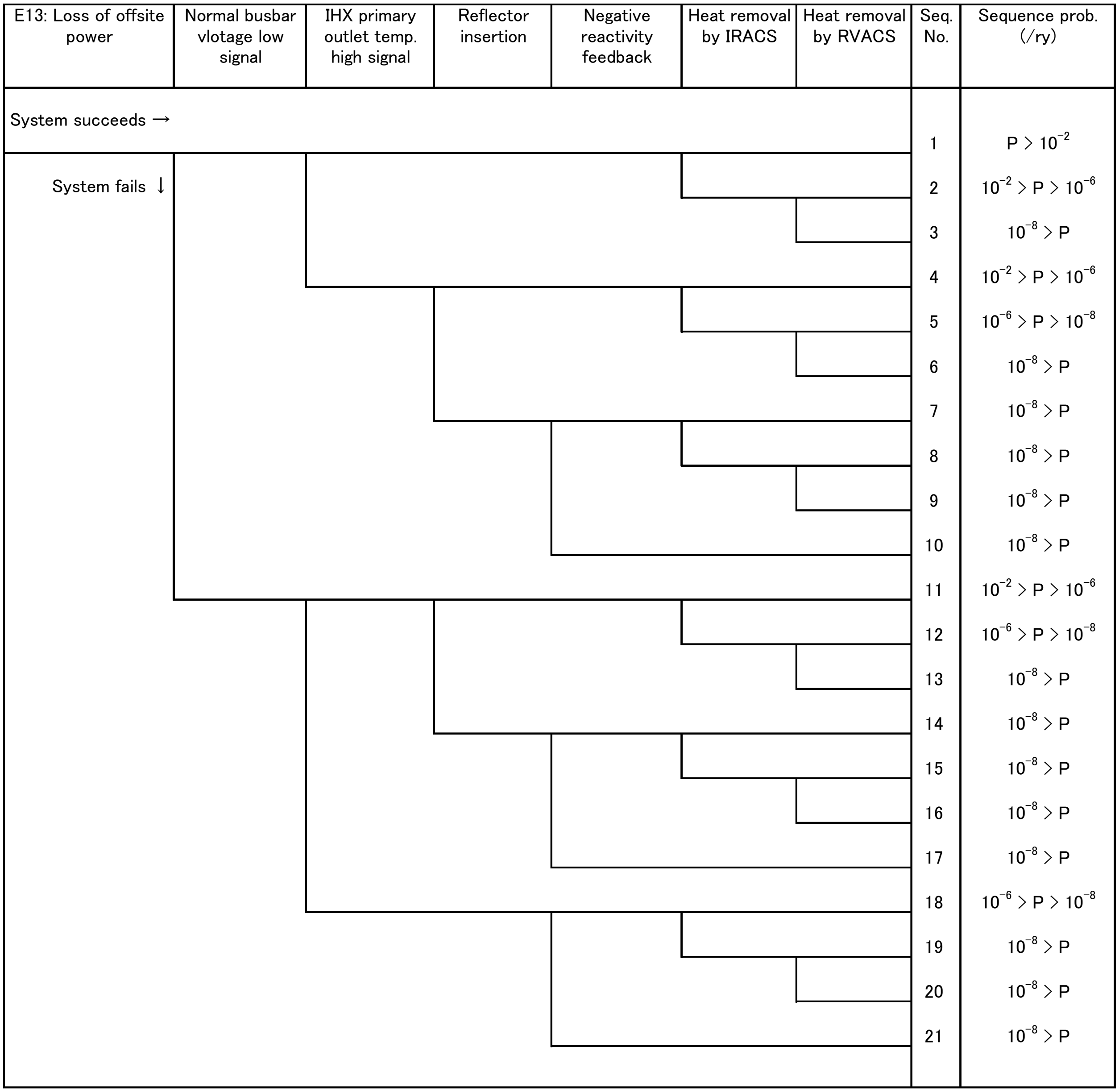


Figure 4.1-13. Event Tree IE13: Loss of Offsite Power

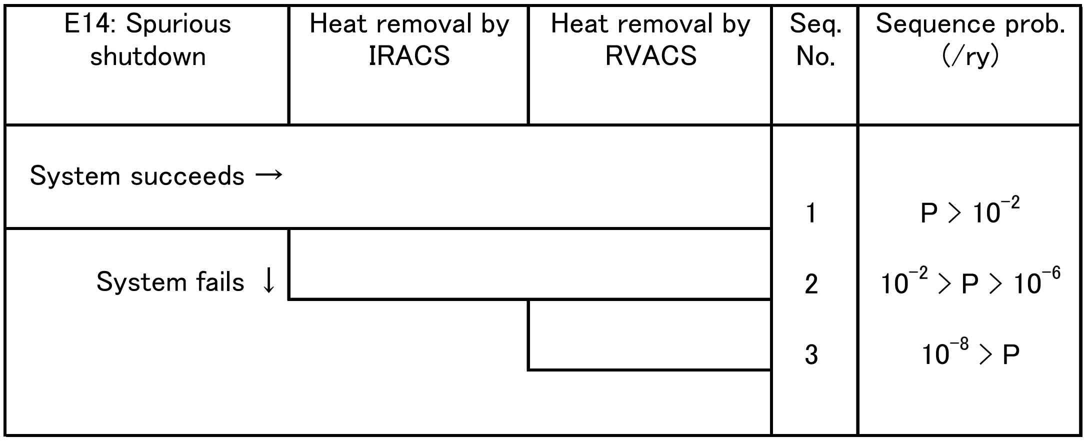


Figure 4.1-14. Event Tree IE14: Spurious Shutdown

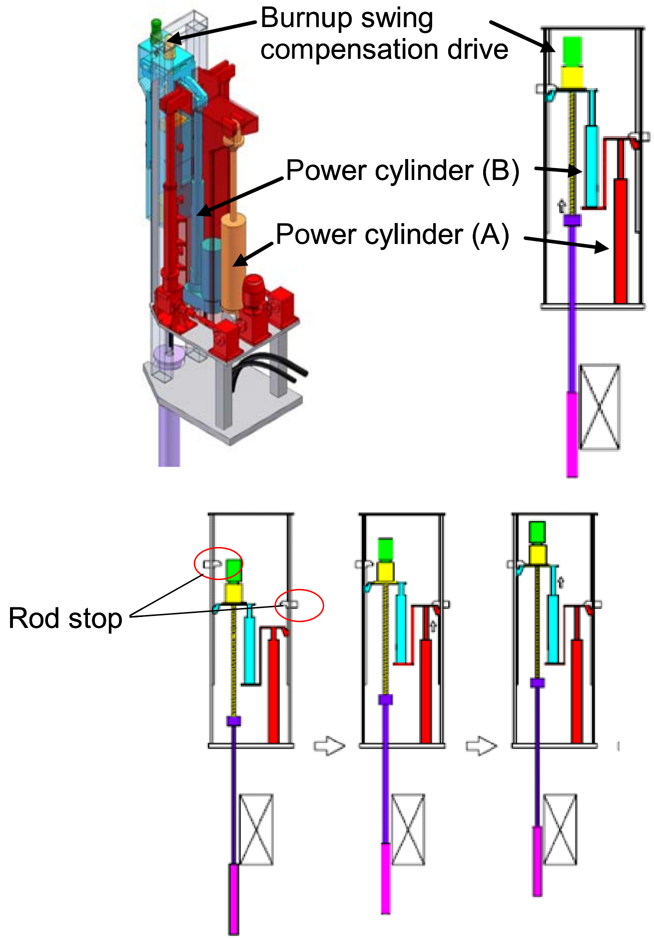


Figure 4.1-15. Reflector Drive Mechanism

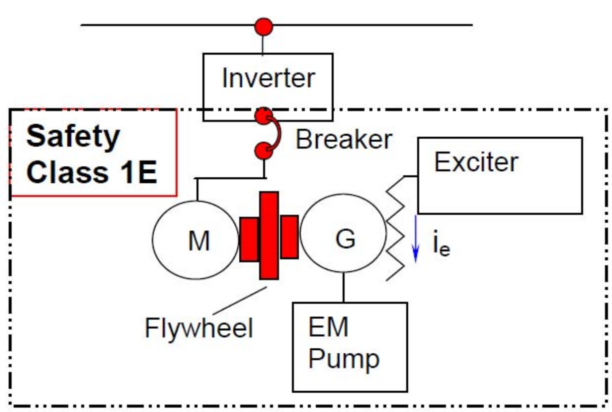


Figure 4.1-16. Flow Coastdown System of Electromagnetic Pump

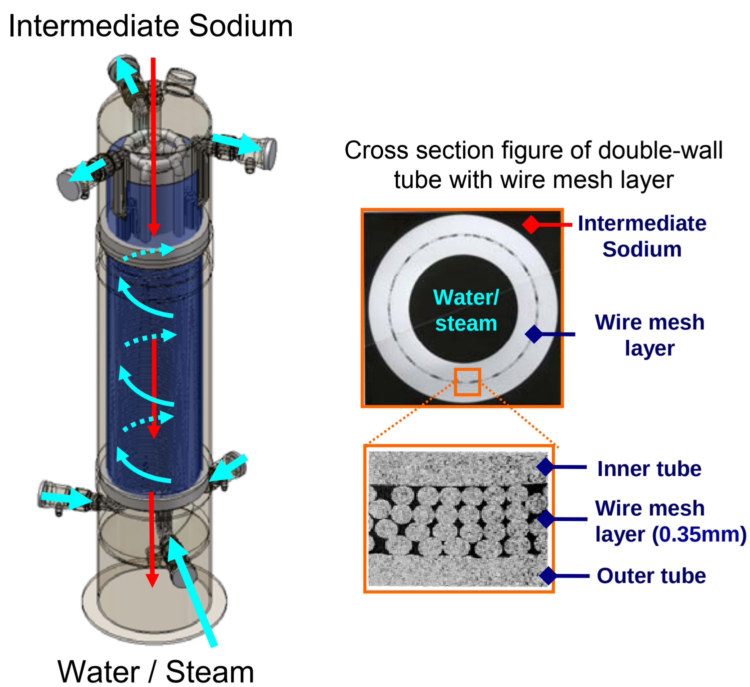


Figure 4.1-17. Section View of Double-Wall Tube and Steam Generator

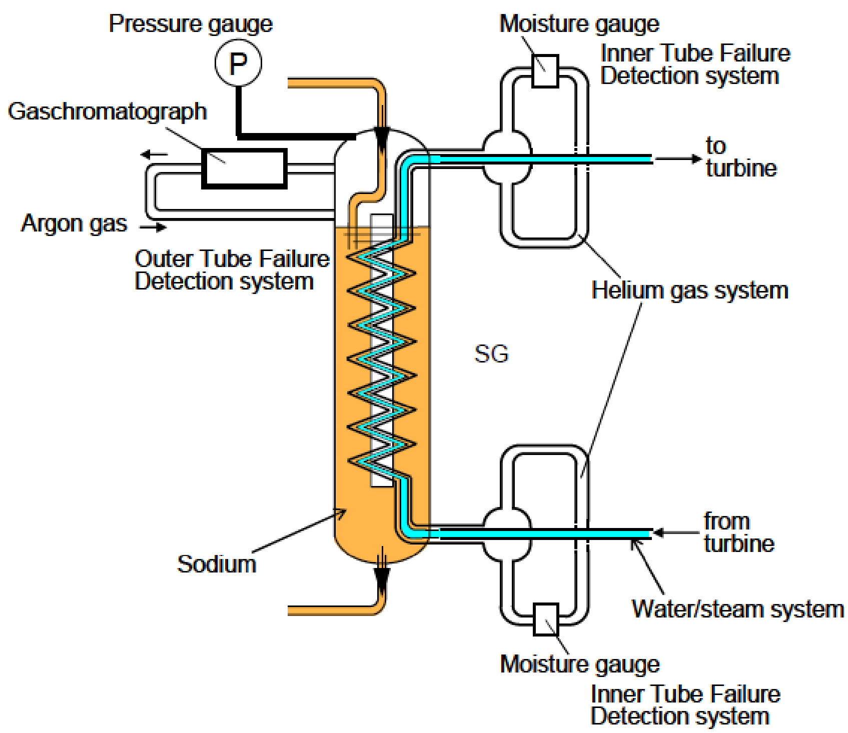


Figure 4.1-18. Detection of Tube Failure in Double-Wall Tube Steam Generator

4.2 Preventive Measures (Internal Events)

This section presents measures to prevent severe accident under the scenario selected in subsection 4.1 and plant behaviors during the sequence of events.

4.2.1 Loss of Primary Flow (without scram)

In a loss of primary flow, 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 the event sequence selected in subsection 4.1.5, however, there is a failure to scram the reactor and it results in ULOF event (Sequence No.4 of Figure 4.1-5(b)).

The plant behavior in such a case is analyzed by one-dimensional flow network plant dynamics code [13]. The plant thermal-hydraulic parameters of the 4S used for the analysis are shown in Figure 4.2.1-1. These parameters are presented in 4S Safety Analysis Report [11].

The sequence and start time of each event are shown in Table 4.2.1-2. For this event, following conditions were taken into consideration:

- Primary flow rate decreases according to the flow coast down characteristic of the primary EMP, which is 30 second of flow halving time,
- Parameters are set to be nominal value,
- The initial condition of the maximum temperature of the cladding is 570°C, and
- The air cooler of the IRACS is assumed to fail.

Figure 4.2.1-1 shows the analysis results of transient state [14]. After primary and intermediate EMPs tripped, the primary flow rate decreases to 20 percent of the rated flow which is a natural circulation state. And the core temperature increases due to the loss of flow. However, the reactor power decreases due to the negative reactivity feed back. The peak cladding temperature and the cladding cumulative damage fraction (CDF) are 743°C and 4.3×10^{-4}, respectively. Therefore, it has a margin to the safety acceptance criteria for fuel cladding integrity, i.e. CDF < 0.1 [11]. Thus, core damage would be expected to be prevented.

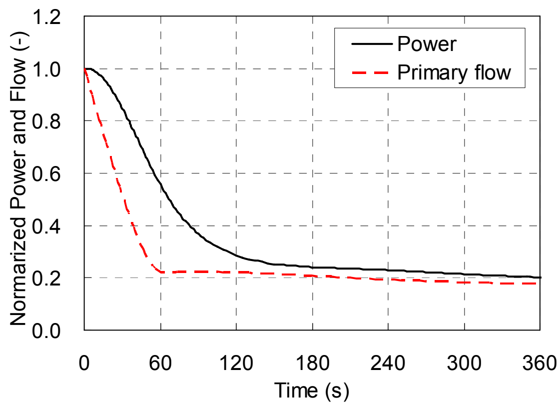
Table 4.2.1-1. Plant Thermal-Hydraulic Parameters

Item	Design Value
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
Feed water/steam temperature	210 / 453°C
Steam generator water/steam flow	4.6×10^4 kg/h
Steam pressure	10.45 MPa
Maximum cladding temperature	570°C
Flow halving time of EMP	30 sec

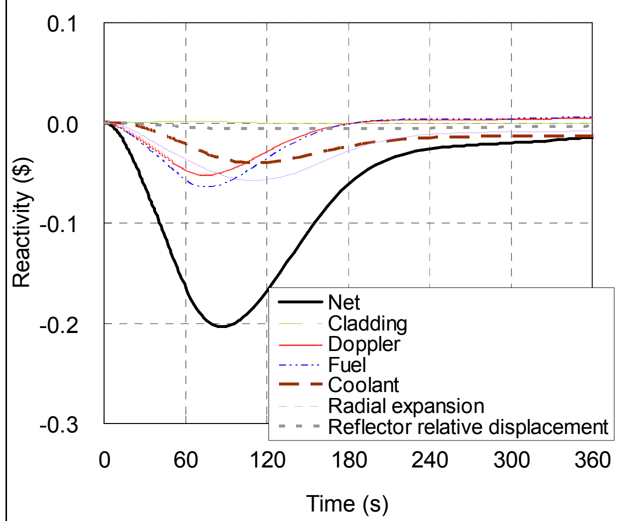
Table 4.2.1-2. Sequence of Events for Loss of Flow without Scram

Time (s)	Events
0	Trip of primary pumps due to failure
0	Trip of the intermediate and feedwater pumps
0	Switch of status of the primary pumps from normal operation to flow coastdown
0	AC damper open failure
0	Loss of heat removal by SG (Immediate loss of heat removal from water/steam system is conservatively assumed)
60	Finish of the flow coastdown state of the primary pumps
60	Residual heat removal by RVACS Natural circulation state of the primary coolant flow

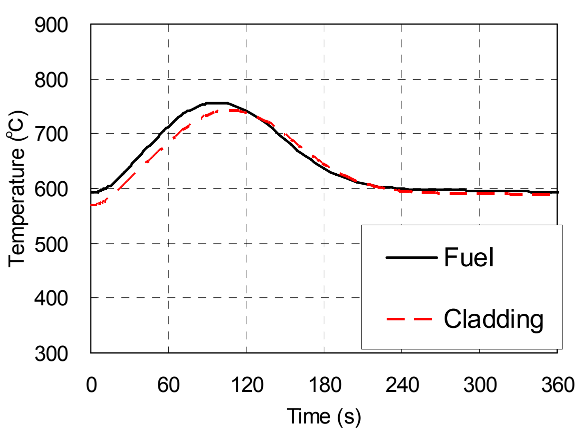
Reactor power and flow ratios



Reactivity transient



Fuel and cladding peak temperature (hottest pin)



CDF (hottest pin)

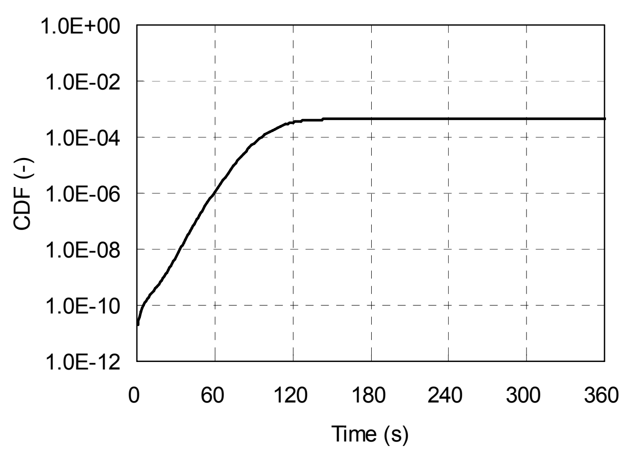


Figure 4.2.1-1. Analysis Result of Transient State of Loss of Primary Flow [14]

4.2.2 IHTS Leak (without scram)

If the intermediate-loop piping is damaged, the sodium coolant in the system piping leaks and burns. Then, sodium leakage is detected by a leakage detector, prompting the operator to immediately shut down the reactor. Moreover, the reactor is normally scrammed automatically by the scram signal actuated by the temperature increase in the IHX primary outlet temperature followed by the leakage. For the event sequence selected in subsection 4.1.10, however, there is a failure to scram the reactor and it results in ULOF event (Sequence No.16 of Figure 4.1-10).

The plant behavior in such a case is analyzed by one-dimensional flow network plant dynamics code [13]. The analysis conditions are the same as those described under loss of flow in subsection 4.2.1 with the following exceptions:

- Heat removal by IHX is assumed to be lost immediately considering draining of sodium from intermediate loop.

- The scram signal transmitted upon this event is that actuated by the IHX primary outlet temperature increase. The signal is set to be transmitted when the IHX primary outlet temperature reaches 390°C with the delay time of 30 seconds. Although the signal is considered to be transmitted, scram itself is assumed to fail e.g. due to failure of the reflector.

The sequence and start time of each event are shown in Table 4.2.2-1.

Figure 4.2.2-1 shows the analysis results of transient state. After the event occurs, the temperature at the IHX primary outlet exceeds the scram setpoint in few seconds, and 30 seconds later, the primary EMPs trip. The primary flow rate decreases to 15 percent of the rated flow which is a natural circulation state. The primary flow rate at natural circulation mode is lower by 5 percent compared to that of the ULOF event shown in Figure 4.2.1-1 because the temperature difference between cold and hot coolant plenum becomes smaller due to loss of heat removal by IHX. The decrease of the core flow leads to temperature increase in the core, which results in decrease of the reactor power to the decay heat power level due to negative reactivity feedback. The effect of negative reactivity feedback for this case is larger than that for the ULOF case (Figure 4.2.1-1) because of the higher temperature of the core.

The peak cladding temperature and the cladding CDF are 800°C and 1.5×10^{-2}, respectively. Therefore, it has a margin to the safety acceptance criteria for fuel cladding integrity, i.e. CDF < 0.1 [11].

In case the fuel temperature increases excessively, fuel melting would occur prior to the cladding melting for metallic fuel. Molten fuel reacts with inside wall of the cladding, forming the liquid phase rapidly, and cause cladding failure due to erosion and creep. The cladding material is HT-9, with a eutectic start temperature at around 650°C. In such a temperature range, the eutectic reaction progresses at a very low speed. When the cladding temperature reaches around 800°C, however, rapid eutectic reaction would be expected to occur [12]. If the higher temperature range is kept for a certain period, the cladding would fail.

The analysis result shows, however, that the peak cladding temperature is kept in the higher range around 800°C in as short period as around 130 seconds 100 seconds after event occurrence. Therefore, loss of thickness due to corrosion is estimated to be 10 μm. Taken into account the cladding thickness 1.1 mm, the integrity of the cladding would be maintained. Thus, core damage would be expected to be prevented.

Table 4.2.2-1. Sequence of Events for IHTS Leak

Time (s)	Events
0	Intermediate heat transfer system leak
0	Loss of heat removal by IHX
t_1*+ 30	Trip of the primary pumps
t_1 + 30	Switch of status of the primary pumps from normal operation to flow coastdown
t_1 + 90	Residual heat removal by RVACS Natural circulation state of the primary coolant flow

Note:

* t_1 : Time when IHX primary outlet temperature reaches scram setpoint.

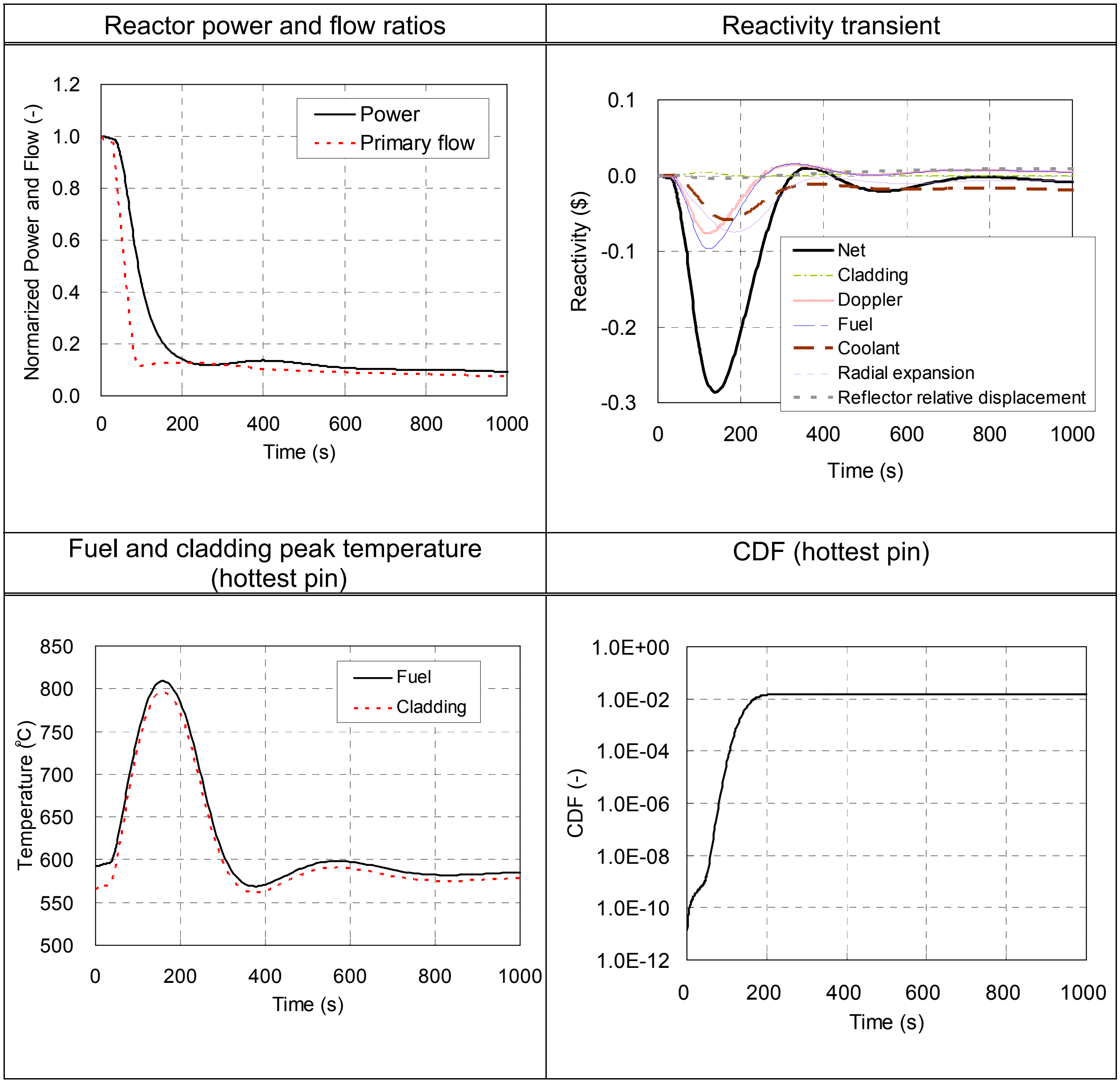


Figure 4.2.2-1. Analysis Result of Transient State of IHTS Leak

4.2.3 Loss of Heat Removal by Water/Steam System (without scram)

When the amount of the heat removal by SG is reduced due to abnormal flow condition in the water/steam system, the reactor is normally shut down automatically. The scenario selected in subsection 4.1.11 for this event, however, results in failure of the reactor scram (Sequence No.22 of Figure 4.1-11). This event results in ULOHS event.

The plant behavior in such a case is analyzed by one-dimensional flow network plant dynamics code [13]. The analysis conditions are the same as those described under loss of flow in subsection 4.2.1 with the following exception:

- Heat removal by SG is assumed to be lost immediately due to abnormality in the water/steam system.

The sequence and start time of each event are shown in Table 4.2.3-1.

Figure 4.2.3-1 shows the analysis results of transient state [15].

The coolant temperature of the IHX primary outlet increases via intermediate coolant temperature increase followed by loss of heat removal by SG. Therefore, the temperature of the core support plate is increased due to the coolant temperature increase at the core inlet. As a result, negative reactivity is added, and the power output decreases. The system average temperature becomes stable at the higher range than that of the rated condition. The peak cladding temperature and CDF are 575°C and 7.2×10^{-9}, respectively. Therefore, it has a margin to the safety acceptance criteria: CDF < 0.1 [11].

Table 4.2.3-1. Sequence of Events for Loss of Heat Removal by Water/Steam System without Scram

Time (s)	Events
0	Loss of heat removal by water/steam system
0	Loss of heat removal by SG
0	Residual heat removal by IHX and RVACS Rated flow mode of the primary and intermediate coolant flow

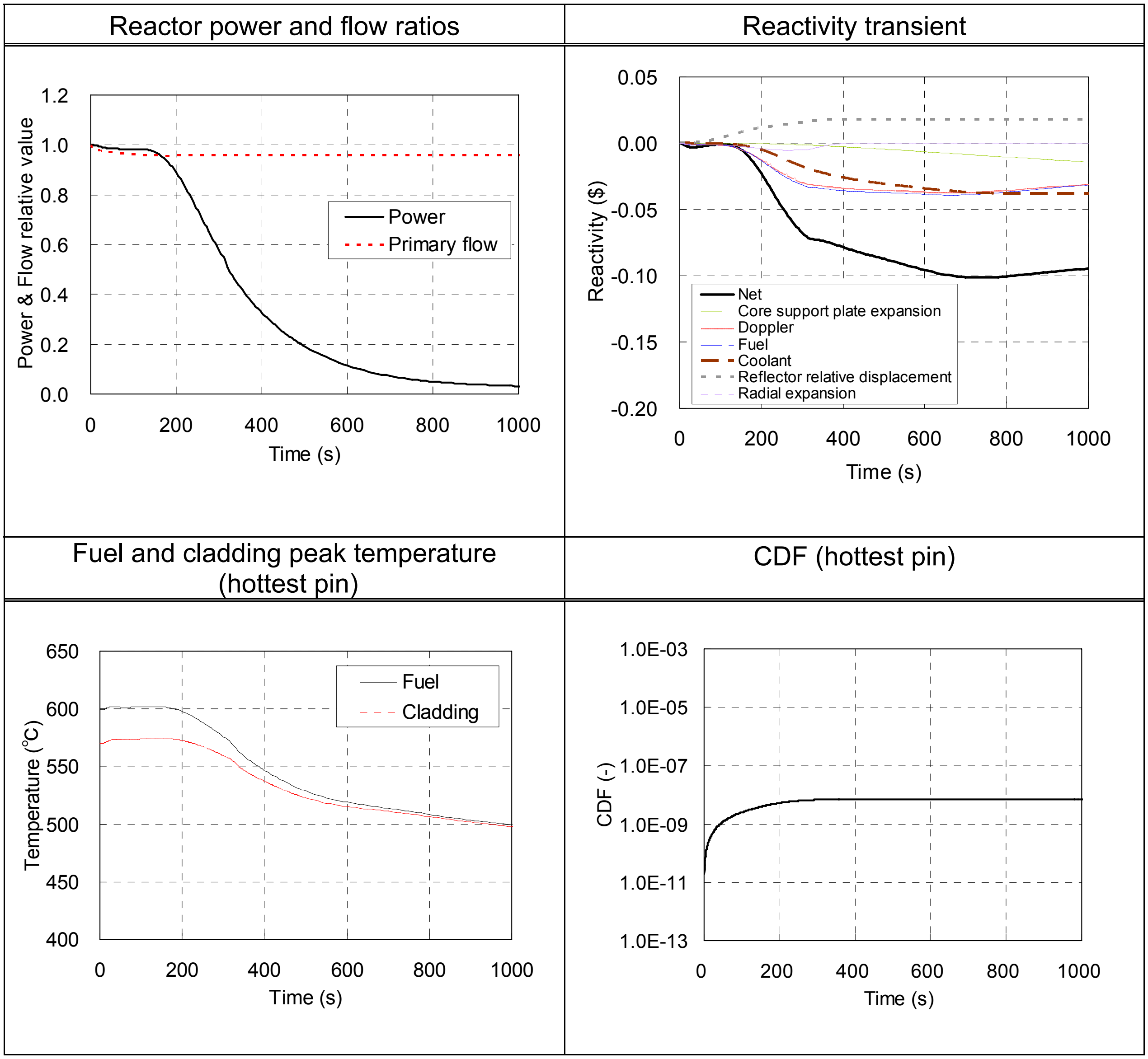


Figure 4.2.3-1. Analysis Result of Transient State of Loss of Heat Removal by Water/Steam System without Scram [15]

4.2.4 SG Tube Leak

The SG design incorporates inner and outer tube failure detection systems. Unless either the inner and outer tubes fail simultaneously, or one tube failure detection system fails in addition to the other side tube failing, a complete penetration tube failure would not occur and lead to a sodium-water reaction. Therefore, the frequency of a sodium-water reaction event is extremely low. Although it is unlikely, this section presents the plant behavior and measures in case of a failure in leak detection as selected in subsection 4.1.12.

If both the inner and outer tubes fail and operation continues, water and steam transfer into the intermediate sodium and a sodium-water reaction occurs.

The intermediate EMP trips on the interlock signal of the intermediate pressure gauge placed in the intermediate system cover gas region, because intermediate system pressure increases if a sodium-water reaction occurs, and subsequently the reactor shuts down.

If detection of sodium-water reaction fails, although it is unlikely, the pressure of the intermediate system is decreased by the sodium-water reaction pressure release system (SWRPS), and the integrity of the primary coolant boundary is maintained.

A configuration diagram of the SWRPS is shown in Figure 4.2.4-1. Assuming the extremely unlikely event of both inner and outer tube failure and detection failure of a sodium-water reaction, the sodium-water reaction would continue followed by a pressure increase of the intermediate system. When the pressure of the intermediate system exceeds the setting value, the SWRPS rupture disk bursts, and the intermediate system pressure decreases. The reactor scrams and feedwater pump trips upon rupture disk burst signal, and the water/steam system will start to blow down.

Solid and liquid materials produced by the sodium-water reaction are transported to the sodium water reaction product storage tank. Hydrogen is burned using an igniter and is released to atmosphere. Intermediate system pressure is decreased, the integrity of the primary coolant boundary is maintained, and residual heat is mainly removed by RVACS.

The consequences of a sodium-water reaction could be different depending on the range of leak rates of water/steam. Therefore, the consequences were analyzed for different leak rate ranges [12]. The relation between leak rate and the associated main event is typically classified as follows:

- Large leak rate (>1 to 2 kg/s): e.g., generation of large amount of hydrogen gas
- Intermediate leak rate (10 g/s ~ 1 to 2 kg/s): e.g., overheating rupture, target wastage
- Small leak rate (0.1 g/s ~ 10 g/s): e.g., wastage
- Very small leak rate (<0.1 g/s): e.g., self-plugging

The scenarios resulting in the occurrence of a large leak rate can be ruled out from the view point of possibilities, and that of a very small leak rate can be enveloped in a small leak rate [12].

For the intermediate leak rate case, in general, the leak rate leading to overheating rupture and target wastage is assumed to be in this range. The influence of this leak rate on the double-wall SG tube in 4S was evaluated in the 4S technical report on phenomena identification and ranking tables [12], and the time from leakage initiation to failure of the inner tube by overheating rupture was obtained as 5.5×10^3 s. From this result, it can be seen that the integrity of the SG system and the intermediate sodium loop system would be kept intact, if the time to opening of the rupture disk is shorter than 5.5×10^3 s. Additionally, the time needed for blowdown must also be considered and if the relationship of [“time to rupture” > (“time to burst of rupture disk” + “time needed for blowdown”)] is maintained, the system would be kept in intact, with no tube failure. Figure 4.2.4-2 shows the relationship between the time to overheating rupture and the time needed for blowdown. The time to overheating rupture is relatively long due to the use of the double-wall tube. Besides, during blowdown, the environment of the water/steam-side changes and this affects the time to failure of the inner tube. The detail consideration of change of time to failure during blowdown was shown in the section 5.5 of the Reference 12.

Then, the time from water leakage initiation to burst of the rupture disk due to sodium-water reaction was considered using the analysis code. As a result, the relation between the time to burst of the rupture disk and the water leak rate is obtained as shown in Figure 4.2.4-3. It can be seen from Figure 4.2.4-3 that the relationship between the time to opening of the rupture disk and the water leak rate is almost linear in a logarithmic scale, and the time to opening of the rupture disk is less than 5.0×10^2 s in the range of the intermediate leak rate. Hence, even if the time of blowdown is designed to be 5.0×10^2 s, the system would be kept intact unless the time to tube failure were less than 1.0×10^3 s.

Therefore, an overheating rupture almost never occurs with the intermediate leak rate. Figure 4.2.4-4 shows the scenario for the intermediate leak rate.

Pertinent to the small leak rate case, the pressure generated by the sodium-water reaction within this range of leak rate is not enough to burst the rupture disk. The neighboring tube failures due to wastage propagate sequentially. If detection of the sodium-water reaction by the intermediate cover gas pressure detector fails, the rupture disk bursts due to the pressure increase. In this case, many more tubes would be failed than in the case of intermediate leakage because the pressure increase does not progress so rapidly.

Therefore, the scenario leading to opening of rupture disk is somewhat different. If detection of the sodium-water reaction by increasing SG cover gas pressure fails, SG pressure increases with the succession of penetration leakage (PL) and leads to opening of the rupture disk. The scenario after opening of the rupture disk is expected to be the same as that for the intermediate leakage rate. Figure 4.2.4-5 shows the scenario for the small and very small leak rate, and Figure 4.2.4-6 shows the relation between time after leak initiation and total leak rate.

Opening of rupture disk causes the transmission of pump trip signal, resulting in reactor scram. Then, water/steam blowdown system is actuated, and pressure in the overall system is decreased.

In conclusion, if detection of sodium-water reaction fails, the pressure of the intermediate system is decreased by the SWRPS, and the integrity of the primary coolant boundary is maintained. Thus, core damage is prevented.

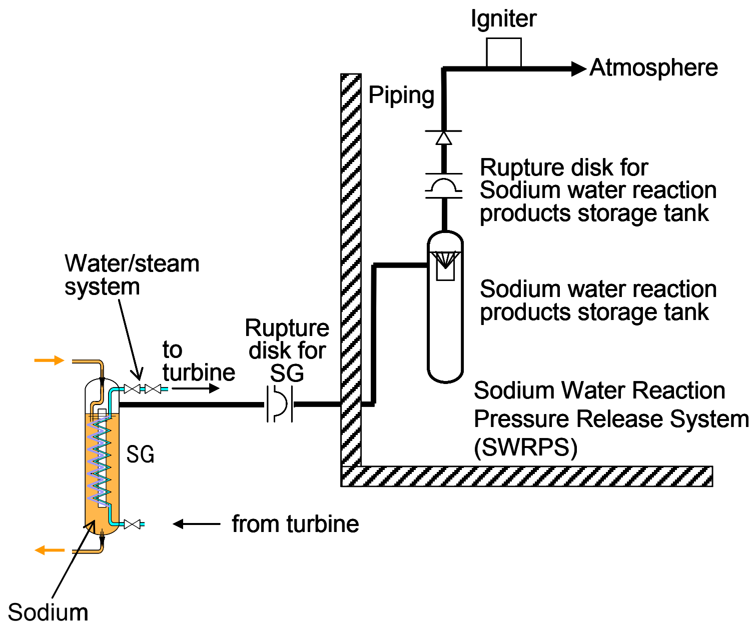


Figure 4.2.4-1. Sodium-Water Reaction Pressure Release System

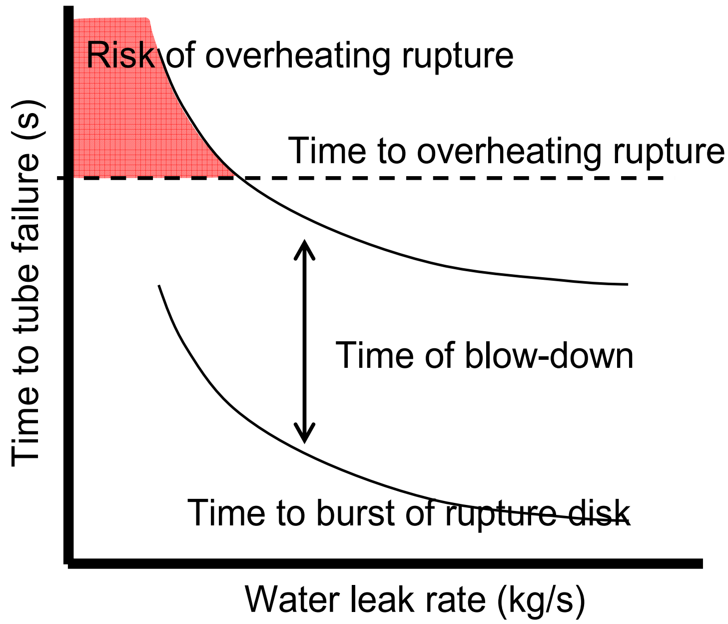


Figure 4.2.4-2. Relation between Time to Tube Failure and Time of Blowdown

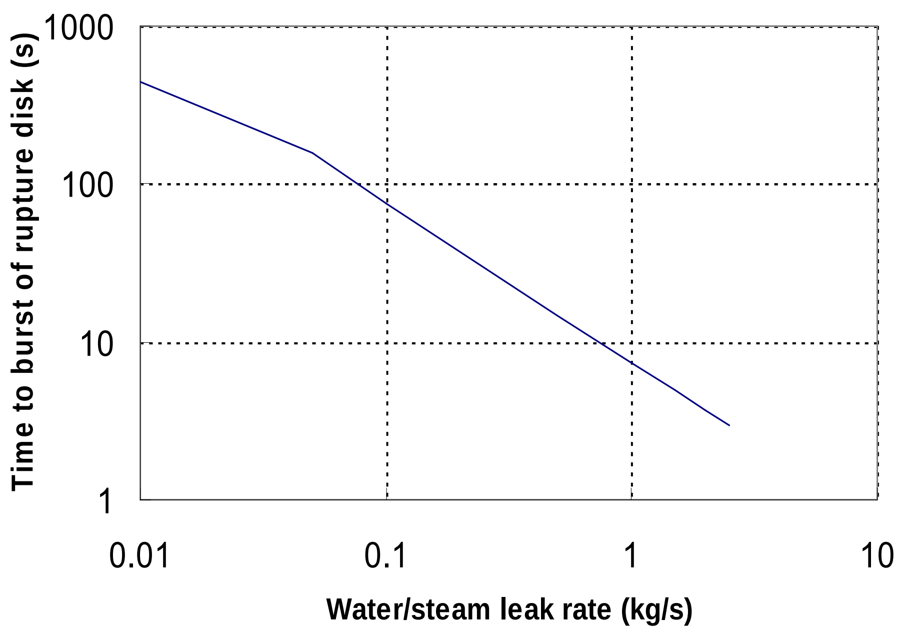


Figure 4.2.4-3. Relation between Time to Burst of Rupture Disk and Leak Rate [12]

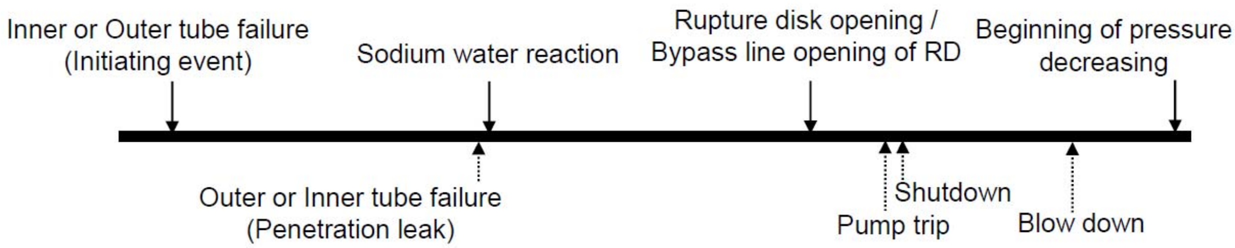


Figure 4.2.4-4. Scenario for the Intermediate Leak Rate

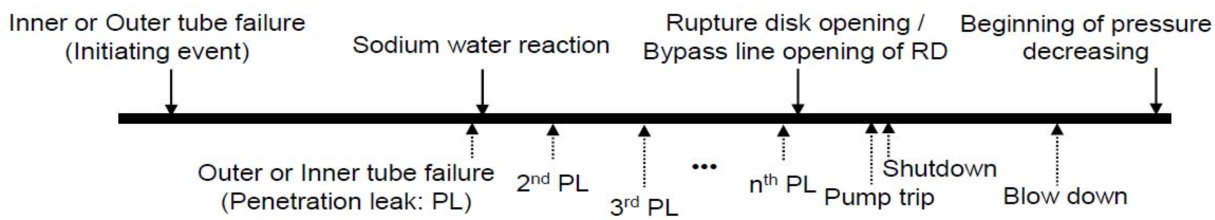


Figure 4.2.4-5. Scenario for the Small and Very Small Leak Rate

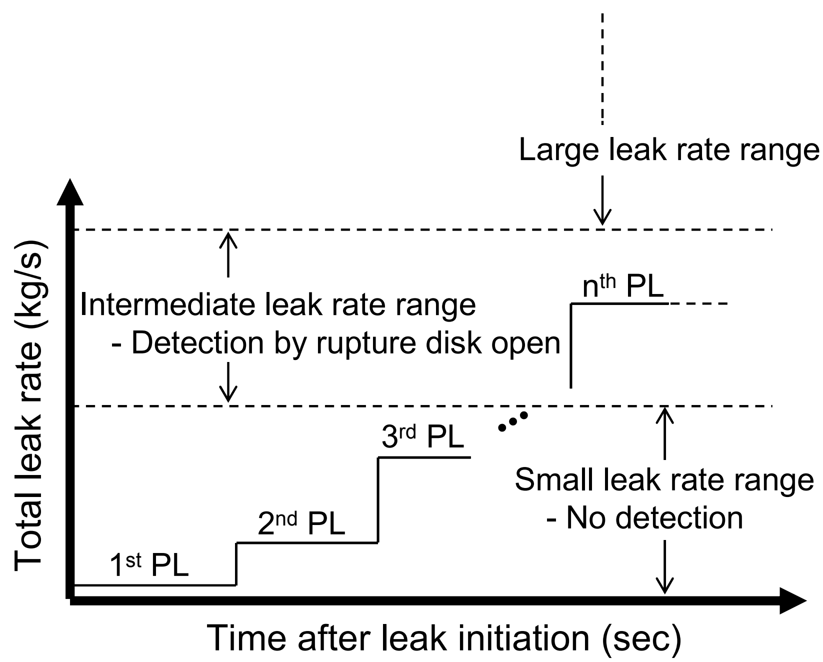


Figure 4.2.4-6. Relation between Time after Leak Initiation and Total Leak Rate

4.2.5 Blockage of Airflow Path of RVACS

With respect to the safety analysis for PRISM reactor, which has been previously evaluated by the NRC, the NRC imposed a bounding event defined as "*Loss of forced cooling plus loss of ACS/RVACS with 25% unblocked after 36 hours*" [16]. By considering this requirement, the analysis was performed assuming a uniform 75% blockage of the RVACS airflow pathways, plus the loss of forced cooling [7]. An indefinite period of time is conservatively assumed for this analysis.

Table 4.2.5-1 shows the analysis scenario for RVACS blockage in addition to IRACS failure. In this case, an unprotected loss of heat sink is assumed due to failure of the water steam system. An immediate reactor shutdown is assumed by detecting failure of the water steam system. A flow coastdown system provided for the EMPs serves to prolong flow coastdown when the normal power supply is stopped. The AC damper is assumed to fail, therefore, the IRACS is not available as a heat sink. One-dimensional flow network plant dynamics code [13] is used under the same analysis conditions used for loss of flow in subsection 4.2.1 with the following exceptions as aforesaid:

- 75 Percent of the RVACS airflow pathways are assumed to be blocked immediately.
- Heat removal by SG is assumed to be lost immediately due to failure of the water/steam system.
- IHX and intermediate system are assumed to be adiabatic when intermediate coolant flow becomes zero.

The analysis result is shown in Figure 4.2.5-1[7]. This result demonstrates that RVACS is tolerant to a wide range of postulated events.

Further, to mitigate the influence of RVACS blockage, another ventilation path is added through emergency exhaust vent, which is protected by reinforced structure and would be activated by heavy machinery only in case of emergency situation. As shown in Figure 4.2.5-1, RVACS can keep required residual heat removal capability with up to 75 percent blockage of its airflow pathway. Even if the entire air exhaust stack is blocked, i.e. loss of ultimate heat sink, blockage rate of air flow pathway can be limited to 70% or less by utilizing the third emergency ventilation path.

Table 4.2.5-1. Sequence of Events for Blockage of Airflow Path of RVACS

Time (s)	Events
0	Manual trip
0	Trip of the primary and intermediate loop and feedwater pumps
0	Switch of status of the primary pumps from normal operation to flow coastdown
0	AC damper open failure
0	RVACS blockage
0	Loss of SG as a heat sink
60	Finish of the flow coastdown state of the primary pumps, start of natural circulation state of the primary coolant flow
2180	Loss of IHX and intermediate system as a heat sink*

Note:

* IHX = intermediate heat exchanger. These systems are assumed to be adiabatic when secondary flow becomes zero.

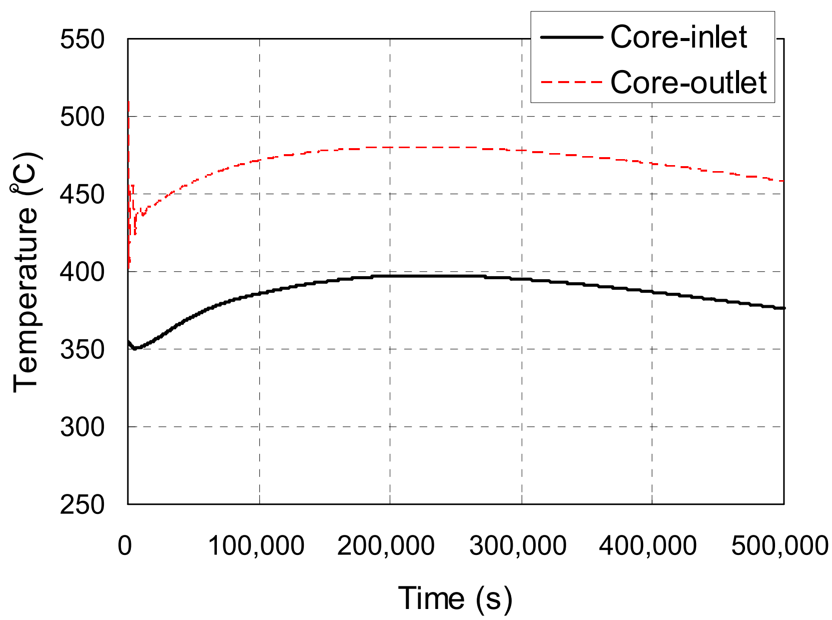


Figure 4.2.5-1. Analysis Result for RVACS 75% Blockage [7]

4.3 Summary of Prevention of Severe Accidents Initiated by Internal Events

Table 4.3-1 summarizes the preventive measures against scenarios initiated by internal events.

Table 4.3-1. Preventive Measure against Internal Events

Event No.	Events	Preventive Measures
IE5	Loss of primary flow	Core damage is avoidable even if without scram due to negative reactivity feedback and passive residual heat removal system.
IE10	IHTS leak	Cladding integrity is maintained even if without scram due to negative reactivity feedback, fuel/cladding compatibility, and passive residual heat removal system.
IE11	Loss of heat removal by water/steam system	Core damage is avoidable even if without scram due to negative reactivity feedback and passive residual heat removal system.
IE12	SG tube leak	Penetration leak is prevented by double wall tube configuration with inner and outer tube failure detection system. Even if detection of tube failure fails, the integrity of the primary coolant boundary is maintained by SWRPS. Therefore, this event does not contribute to core damage.
-	Blockage of flow path of RVACS	The RVACS air exhaust stacks are installed well away from each other to avoid simultaneous breakage. Another ventilation path is added through emergency exhaust vent, which is protected by reinforced structure and would be activated by heavy machinery only in case of emergency situation.

5 PREVENTION OF SEVERE ACCIDENTS INITIATED BY EXTERNAL EVENTS

5.1 Event Selection (External Events)

The external events to be taken into consideration are identified based on the Commission Paper “SECY-11-0025” [5] as follows:

- Seismic event
- Flooding
- Other natural external events other than seismic and flooding
- Station Blackout

5.2 Preventive Measures (External Events)

5.2.1 Seismic Event

4S incorporates seismic base isolation (SBI) in its reactor building design [17]. SBI is a building protection technique that reduces the horizontal seismic force input to a structure by the installation of isolation devices, generally between the building and its supporting base. Usually, such isolation devices are composed of multiple alternating layers of steel plates and rubber as shown in Figure 5.2.1-1. Figure 5.2.1-2 shows the vertical section of a 4S reactor building. The main specification of the building is given in Table 5.2.1-1.

Base isolation is provided at the bottom of the reactor building, and the isolators are set on a base mat on the ground. The outside of the reactor and reactor building is surrounded and secured by soil retaining walls with a seismic gap of sufficient size.

The seismic hazards evaluation is dependent on the site because it requires site specific assessments such as seismic margin, fragility, and so on. Hence, when the site is specified, the seismic hazards evaluation, including the effect of vertical vibration, will be performed using the latest seismic conditions to be applied to the intended site.

Table 5.2.1-1. Size and Weight of the Reactor Building

Plant size	Approx. 30m x 24m
Depth from ground surface	Approx. 20m
Weight of isolated building including all mechanical components	14,126 tons

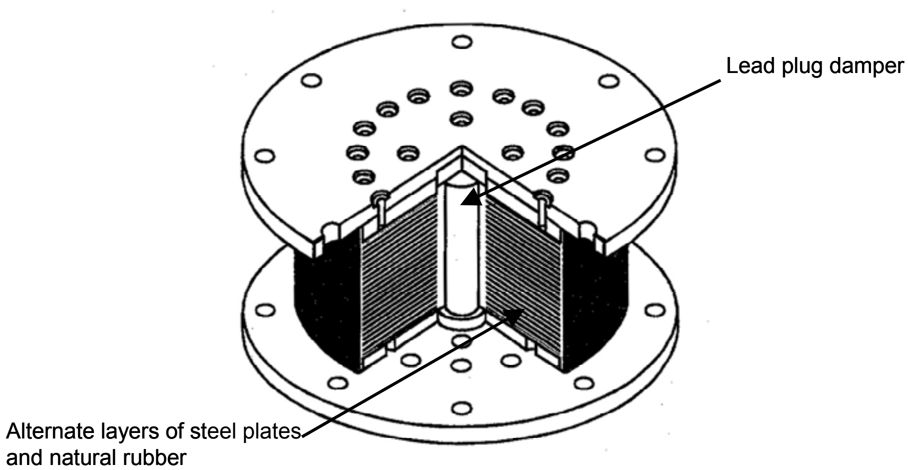


Figure 5.2.1-2. A Seismic Isolator

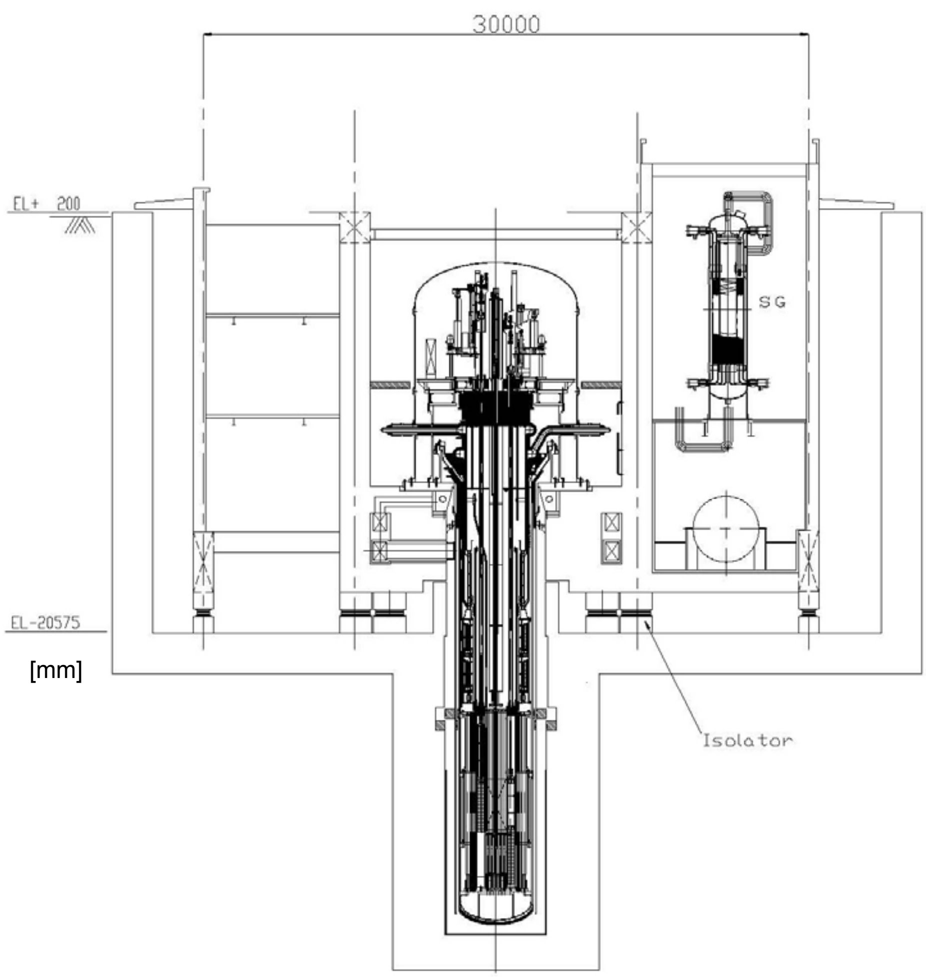


Figure 5.2.1-1. Vertical Section of Reactor Building

5.2.2 Flooding

4S incorporates passive decay heat removal system in its design which requires no external power supply or emergency power system, so that the risk against loss of ultimate heat sink is reduced. Moreover, precautions are taken against flooding such as watertight bulkhead to prevent water from entering the reactor building, flow path of the RVACS, and area where seismic isolators are installed.

As like the seismic hazard evaluation, the evaluation of flooding hazards will be performed using the latest conditions to be applied to the intended site when the site is identified.

5.2.3 Other External Events

Other than seismic event and flooding, following natural phenomena are identified to be considered according to NRC letters [18] and NEI document [6].

- Tornado
- Hurricane
- Severe wind
- Precipitation
- Ambient temperatures
- Evaporation and drift loss of cooling water
- Water freezing in the water storage facility
- Other specific phenomena for a particular site, such as avalanche, biological events, coastal erosion, forest and grass fires, hail, high tide, ice cover, lightning, river diversion, sand and dust storms, seiche, snow, storm surge, thunderstorms, tsunami, volcanic activity, water spouts, and waves.

The plant is designed to withstand the maximum probable natural phenomena at the intended site under compliance with applicable NRC regulations and associated guidance. The evaluation of those events will be performed using the latest conditions to be applied to the intended site when the site is identified.

5.2.4 Station Blackout

As described in section 3, the residual heat removal systems use the natural air draft outside the guard vessel, i.e. RVACS, and both the natural circulation of sodium in the intermediate loop and the air draft at the air cooler, i.e. IRACS. Due to the passive residual heat removal capability of these systems, heat is removed from the reactor even in the case of station blackout (SBO). The event sequence is equivalent to that of loss of offsite power as described in subsection 4.1.13. Primary pumps, intermediate pump, and feedwater pump trip simultaneously upon SBO, which cause the reactor to automatically shut down by scram signals such as primary EMP supply voltage low signal or IHX primary outlet temperature high signal. Subsequently, decay heat is removed by the RVACS and IRACS.

Figure 5.2.4-1 shows the core temperature behavior during an SBO [11]. This case shows that residual heat is successfully removed through the RVACS and IRACS using only natural circulation. The plant thermal-hydraulic parameters of the 4S used for the analysis are the same as those used for the analysis of loss of primary flow event as described in subsection 4.2.1.

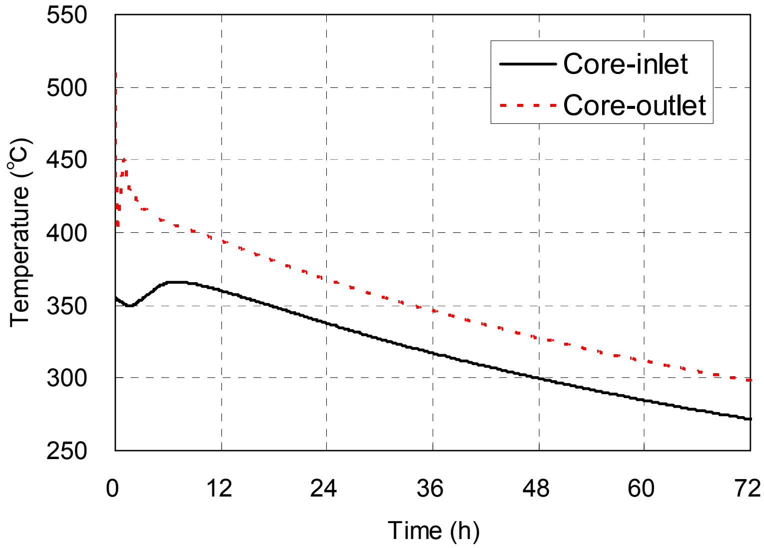


Figure 5.2.4-1. Analysis Result of Heat Removal by IRACS and RVACS with Natural Circulation [11]

5.3 Summary of Prevention of Severe Accidents Initiated by External Events

Table 5.3-1 summarizes the preventive measures against scenarios initiated by external events.

Table 5.3-1. Preventive Measure against External Events

Events	Preventive Measures
Earthquakes	Supporting the reactor building by seismic isolator.
Flooding	Redundant shutdown system and passive decay heat removal system without external power supply and emergency power system, and watertight bulkhead to prevent water from entering the reactor building, airflow pathway of the RVACS, and seismic isolators.
Other external event than earthquake and flooding	Constructed under ground.
SBO	Core damage is avoidable without any emergency power supply system by passive decay heat removal system with natural circulation, not necessary the pump. There is no limitation for duration time.

6 CONCLUSION

The 4S preventive measures against severe accident which is defined as a type of accident that may challenge safety systems at a level much higher than expected were reviewed. The report focused on the scenarios initiated by internal and external events. For the internal events, the initiating events were selected based on the 4S preliminary PRA and the historical experience of the fast reactors obtained by the safety evaluation for CRBR and PRISM. For the external events, the initiating events were selected according to the Commission Paper "SECY-11-0025".

Four scenarios initiated by internal event were selected to have the more important potential to cause severe accident, namely, loss of primary flow, IHTS leak, loss of heat removal by water/steam system, and SG tube rupture. Besides, the scenario associated with blockage of air flow path of RVACS was included to analyze its consequence. In each scenario, however, core damage was estimated to be avoidable due to negative reactivity feedback, passive decay heat removal system, EMP, double wall tube SG, and another ventilation pathway for RVACS.

For the external events, four events were selected, namely, seismic event, flooding event, other natural external events other than seismic and flooding, and SBO. The plant is designed to withstand the maximum probable natural phenomena at the intended site by adopting seismic isolator, watertight bulkhead, and so on.

In conclusion, it is demonstrated that severe accident is avoidable and core integrity is maintained by the preventive measures provided with the 4S reactor. The hazards evaluation on external events remains to be performed. After identifying the site, those remaining evaluation will be performed and documented under the latest hazard conditions to be applied to the intended site.

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