XA9846617

APPLICATION OF PSA LEVEL 1 FOR THE FUGEN PROTOTYPE ATR PLANT

H. IKEDA, Y. IGUCHI, M. SOTSU
Fugen Nuclear Power Station,
Power Reactor and Nuclear Fuel Development Corporation,
Fukui-ken

O. SEKI, S. SATOU
Advanced Reactor and Nuclear Fuel Cycle Department,
Hitachi Ltd,
Ibaraki-ken

Japan

Abstract

This paper presents application of PSA Level-1 for Prototype Advanced Thermal Reactor Plant "Fugen" in consideration of the design characteristics of such reactor and verify the safety aspect of Fugen using thus established procedure. ATR resembles the boiling water reactor (BWR) in a number of points, but there are also some differences between the ATR and the BWR. Therefore, PSA procedure have been established by taking such difference into consideration and by referring to experience of PSA in USA and Japan. Moreover, the core damage frequency was calculated on Fugen by using thus established procedure. As the result, it was verified that results including the maximum value of the uncertainty estimation were found to be quite satisfactory against the target value of reactor damage frequency defined by International Atomic Energy Agency (IAEA).

1. INTRODUCTION

In Japan, a nuclear plant is licensed, in principle, through the process of Deterministic Safety Assessment. However, it has been known that Probabilistic Safety Assessment (PSA) which enables the quantitative evaluation of the safety aspect of a nuclear plant including the accidents exceeding design basis events is an effective method, complementing the deterministic safety assessment, and it is not unusual that the results of PSA are also taken into consideration in the course of issuing the final approval.

The objectives of this study are to establish the PSA procedure to be applied to an ATR in consideration of the design characteristics of such reactor and with the results of PSA implemented in the U.S.A. as well as PSA actually performed on a number of domestic light-water reactors for reference and, at the same time, to verify the safety aspect of the Fugen using thus established procedure. Safety researches for ATR are shown in the table 1-1. This study has been performed on the basis of the result of these safety researches. The part of these safety researches have been continued.

2. CHARACTERISTICS OF FUGEN

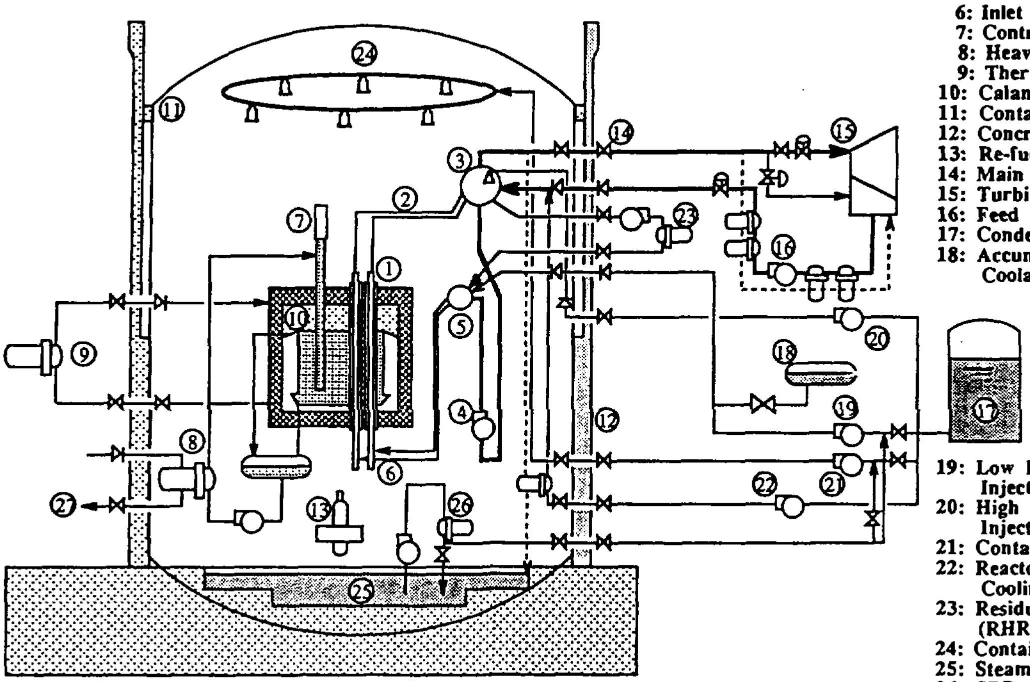
The Fugen is of a boiling water cooled and heavy water moderated Pressure Tube type reactor equipped with a safety system shown in Fig. 2-1 and Table 2-1. The Fugen is a direct cycle power generator plant and resembles the boiling water reactor (BWR) in a number of points such as that it uses about 7 MPa boiling light water for coolant and the steam generated in the reactor goes into steam drums and sent on to turbine generators after separated into liquid and vapor forms in the drum. But there are also some differences between the ATR and the BWR, therefore, PSA procedure has been established by taking such difference into consideration as follows:

TABLE 1-1 SAFETY RESEARCHES FOR ADVANCED THERMAL REACTOR

Category	Item
· Integrity of MOX fuel assembly structure and fuel rods	· Irradiation test of MOX fuel · Post irradiation examination · Irradiation test of segmented fuel for ramp test · Ballooning test of fuel cladding · Development of fuel failure detection etc
· Thermohydraulic aspects	· Test of Critical heat flux · Heat transfer test of natural circulation · Heat transfer test of sudden decrease of channel flow rate · Heat transfer test at LOCA · Safety analysis with statistical method etc
· System.Components	· Insertion test of control rods at earthquake · Hydrogen water chemistry · Post irradiation examination of test specimens of pressure-tube material · Development of remote-controlled pressure-tube monitoring equipment · Development of detector(microphone) for small break of pipes · Test of LBB(Leak Before Break) for inlet feeder pipes and outlet pipes · Test of pipe whip for inlet feeder pipes and outlet pipes · Development of remote-controlled monitoring equipment for inlet feeder pipes and outlet pipes etc
· Beyond design basis accidents	· Test of core coolability by coolant in primary loop under condition of beyond design basis accidents · Test of core coolability by heavy water moderator · Test of void formation in heavy water for evaluation of reactivity accidents under condition of ATWS · Test of integrity of the calandria tube and vessel after pressure tube rupture · Development of analysis method for thermohydraulic aspects in containment · Probabilistic Safety Assessment(PSA) · Accident manegement · Development of symptom-oriented Emergency Operating Procedures(EOP) etc

(a) The reactor core is to have two halves that shall be cooled separately with each independent cooling system.

The cooling system consists of two loops, each charging one half of the reactor core. Figure 2-2 shows the reactor core cooling system, which operates each of the two loops of Fugen coolant system at LOCA. If a LOCA occurs where the piping system may be ruptured in a point from which the coolant may be discharged, the system shall be scrammed and emergency measures shall be taken to shut off the Main Steam Isolation Valve and to inject coolant into the reactor core through the Reactor Isolated Cooling System (RCIC). Since coolant would be lost from the ruptured part of the damaged loop, the Emergency Core Cooling System (ECCS) shall be actuated to secure necessary coolant. The fuel in the other unruptured loop shall be injected coolant by RCIC and the decay heat removed by the Residual Heat Removal System (RHR). As seen from above, Fugen



1: Pressure tube
2: Outlet pipe
3: Steam drum
4: Re-circulating pump
5: Water drum
6: Inlet pipe
7: Control rod drive
8: Heavy water cooling system
9: Thermal shield cooling system
10: Calandria vessel
11: Containment vessel
12: Concreat
13: Re-fueling machine
14: Main Steam Isolation valve
15: Turbine system
16: Feed water system
17: Condensate storage tank
18: Accumulated Pressurized Coolant Injection (APCI) system
19: Low Pressure Coolant Injection (LPCI) system
20: High Pressure Coolant Injection (HPCI) system
21: Containment spray system
22: Reactor Core Isolation Cooling (RCIC) system
23: Residual Heat Removal (RHR) system
24: Containment spray header
25: Steam Release Pool (SRP)
26: SRP Cooling (SRPC) system
27: reactor auxiliary cooling system

FIGURE 2-1 SCHEMATIC DIAGRAM OF COOLING SYSTEMS PROVIDED IN FUGEN

TABLE 2-1 THE SAFETY SYSTEM OF FUGEN

<table>
<tr><td>Reactor Shutdown System</td><td colspan="2">Control Rod System (CRS)
Helium Gas Circulation System (HEC)</td></tr>
<tr><td>Number of Systems</td><td colspan="2">2</td></tr>
<tr><td>Number of DGs</td><td colspan="2">2</td></tr>
<tr><td rowspan="8">Composition of ECCS and Heat Removal System</td><td colspan="2">Reactor Core Isolation Cooling System(RCIC)*</td></tr>
<tr><td>A-Residual Heat Removal System(RHR)</td><td>B-Residual Heat Removal System(RHR)</td></tr>
<tr><td>A-High Pressure Coolant Injection System(HPCI)</td><td>B-High Pressure Coolant Injection System(HPCI)</td></tr>
<tr><td>A-Low Pressure Coolant Injection System(LPCI)</td><td>B-Low Pressure Coolant Injection System(LPCI)</td></tr>
<tr><td>A-Steam Release Pool Cooling System(SRPC)</td><td>B-Steam Release Pool Cooling System(SRPC)</td></tr>
<tr><td colspan="2">Accumulated Pressure Coolant Injection System(APCI)*</td></tr>
<tr><td colspan="2">Moderator Cooling System(MCS)**</td></tr>
<tr><td colspan="2">Containment Air Cooling and Cleanup System (CAC)**</td></tr>
</table>

* There are two systems of dynamic devices or injection valves.
** MCS and CAC are planned to be effective on some specific initiating events only.

differs from BWR in that different cooling systems function and different series of processes take place in each of the two separate loops at the occurrence of an abnormal condition. At the occurrence of a transient event such as Loss of Offsite Power Accident, the same sequence of events take place in both loops with RHR and RCIC working concurrently.

(b) Heavy water used as moderator.

For moderator in Fugen, heavy water is used which is low in neutron absorption and able to burn fuel more efficiently as compared with light water. The Fugen is equipped with a vertical cylindrical tank called a calandria, which is filled with heavy water moderator and is separated from the coolant in the Pressure Tubes.

(c) The Fugen has 224 fuel channels called Pressure Tubes.

Fuel assembly used in the ATR is housed in 224 Pressure Tubes each one of which is independent from each other. Furthermore, each one of Pressure Tubes comprising this assembly is connected to the water- and steam-drums through an Outlet, respectively, Inlet tubes to carry coolant to each fuel component of the assembly. The existence of many more small size tubes than BWR is a particular feature. A safety systems have been designed for Fugen against the rupture of these tubes.

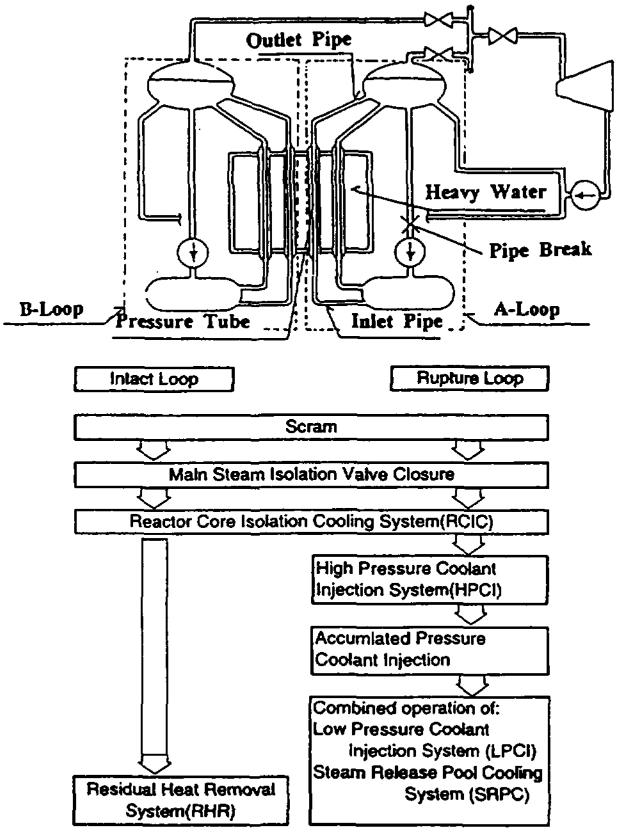


FIGURE 2-2 COOLING SYSTEM AT LOCA

3. DEVELOPMENT OF PSA LEVEL 1 PROCEDURE

3.1 Summary of Assessment Procedure

It has been first selected that a variety of events which initiate internal events that are the objectives of the intended assessment. It has been, then, drew out from the group of the Initiating Events the elements of safety functions required to prevent any damage to the reactor core and define the Success Criteria. Subsequently, Event Tree has been developed on the basis of the Success Criteria thus defined and performed a Fault Tree analysis (FTA) on the database of frequency of machine failure for the purpose of determining the branching ratio of the Event Tree. And, finally, it had been tried to determine the accident sequence with a probability to develop into a core damage along with the estimated level of core damage frequency through the Event Tree analysis. It shall be discussed in the following paragraphs how the characteristics of Fugen mentioned above can be reflected upon the evaluation method of Fugen.

3.2 Determination and quantification of Initiating Events

3.2.1 Selection of Initiating Events

At the selection of Initiating Events related to Fugen in this study, we have first developed the Master Logic Diagram of internal events. Then, these Initiating Events were divided into different groups such as the one to which the plant responses in the same way, the group of safety assessment events, the group of Initiating Events for prior PSA, etc.

The characteristics of ATR have been taken into consideration at the selection of the Initiating Events as follows:

(a) With respect to the reactor core divided into two halves with each loop for cooling, the characteristic of ATR is in the fact that different systems operate on each one of the two loops as mentioned in 2.(a) above, therefore, it was not taken reflected upon the selection of the Initiating Events but on the "Event Tree analysis."

(b) The use of heavy water for moderator was reflected on the Initiating Events. Such events as "Abnormal rise in the moderator temperature" or "Rupture of a heavy water tube" may be considered as events peculiar to ATR, therefore, were selected and grouped as Initiating Events: "Trouble in heavy water moderator system."

(c) ATR contains many small size tubes such as Outlet/Inlet tubes or Pressure Tubes which are some of the characteristic feature of Fugen and the possibility of the rupture of such tubes must be taken into consideration. Since the rupture of these tubes would be success criteria similar to the small LOCA, they would be grouped under the Initiating Events of the "Small LOCA" and shall be taken into consideration at the quantification of Initiating Events.

3.2.2 Quantification of Initiating Events

(1) Quantification of LOCA

The quantification of LOCA was done by reflecting the characteristics of ATR: the existence of a large number of small size tubes such as Outlet/Inlet tubes and Pressure Tubes. The method for quantification of the frequencies of rupture among general tubing and among the tubing peculiar to ATR is discussed below. It should be noted that the rupture of Outlet/Inlet and Pressure Tubes was handled as the small LOCA and included in the initiating frequency of the small LOCA.

1) Rupture of similar tubing to BWR

The frequency of rupture occurring to similar tubing to BWR other than those characteristic of ATR was quantified from actual results obtained from light water reactors (especially BWR) in Japan and U.S.A. (about 510 reactor years in total) with reference to prior PSA since they are basically the same in the composition.

2) Rupture of tubing characteristic of ATR

The frequency of rupture among tubing characteristic of ATR such as Outlet/Inlet pipes and Pressure Tubes was quantified from the results of PSA level 1 Study on prior LWRs[1] and also the results of another assessment based on probabilistic rupture mechanics.

(2) Quantification of transient events

1) Similar events to BWR

The frequency of transient events excluding Unusually heavy water moderator system that are characteristic of ATR was quantified from the results obtained from domestic light water reactors (especially BWR) and also from the actual operating results of the Fugen, (about 245 reactor years in total).

2) Events characteristic of ATR

"Unusually heavy water system" are events characteristic of ATR, therefore, were quantified on the basis of operating results of Fugen (13 reactor years in total) only.

The quantified frequency of Initiating Events thus obtained are shown in Table 3.2-1. The sum of the Initiating Event frequencies in this study for ATR determined from the above described source data came to be 4.3×10^{-1} reactor years which reflects the successful operating results of domestic LWRs and Fugen. The estimation of the intervals of Initiating Events was made using χ^2 distribution.

TABLE 3.2-1 INITIATING EVENTS FREQUENCY

Initiating Event	Frequency (1/reactor. years)
Large LOCA	4.4×10^{-5}
Intermediate LOCA	1.4×10^{-4}
Small LOCA	5.3×10^{-4}
Turbine Trip	2.7×10^{-1}
Failure of Heavy Water Moderator System	7.7×10^{-2}
Main Steam Isolation Closure	7.3×10^{-2}
Loss of Offsite Power	8.2×10^{-3}
Total	4.3×10^{-1}

3.3 Determination of Success Criteria

After extracting safety functions required to prevent core damage in reactors from Initiating Events selected as described in 3.2 above, the success criteria have been defined for each safety function. These success criteria were defined by investigation of design information and also by analyzing data on the basis of the authorization code of nuclear plants and other available information, as necessary. The definition of core damage in this study shall be deemed to have occurred when the clad surface temperature reaches over 1200°C which is in accordance with the current judgment criterion of the Deterministic Safety Assessment. The characteristics of Fugen were reflected upon the determination of the success criteria as follows:

(a) The fact that the core is divided into two halves each one of which is cooled by separate loop of coolant was dealt with by setting up a pair of success criteria, one for each loop. The state of the ruptured loop and unruptured loop will be different at the time LOCA, therefore, different criteria shall apply to the respective loops. The unruptured loop will display the same behavior at the time of LOCA as in the transient event, therefore, shall be subject to the same success criteria as in the transient event.

(b) The success criteria in consideration of another of the Fugen characteristics: the existence of many small size pipes and tubes such as Outlet/Inlet and Pressure Tubes, are handled in the way similar to that of the small LOCA since they are similar in the cross sectional area as well as the cooling system required.

(c) The use of heavy water for moderator was considered upon by incorporating the cooling effect on the core of the heavy water moderator system in the ATR-specific success criteria. The cooling function of the heavy water moderator system corresponds to about 28 MWt and there is a possibility that the reactor core may continue to be cooled through the following mechanism even after such other cooling systems as RCIC, RHR and ECCS were rendered unserviceable. Fig. 3.3-1 shows the principle of the heavy water core cooling system. After a reactor comes to scram as the result of LOCA or a transient event and the core cooling functions breaks down, the coolant within Pressure Tubes shall be lost through leakage or decay heat. Despite such loss of coolant from Pressure Tubes, however, the decay heat will be transferred from fuel rods to Pressure Tubes by conduction and from Pressure Tubes to calandria tubes by radiation and, then, on to the moderator, or heavy water, to complete the core cooling process. Furthermore, the core cooling effect shall further increase as the heat transfer to heavy water is further intensified by heat transfer through contacts between the Pressure Tube and Calandria tube coming into contact by ballooning of the Pressure Tube. A number of tests have been conducted by PNC (Power Reactor and Nuclear Fuel Development Corporation) to verify the above described mechanism[2] and we have analyzed the cooling effect of heavy water in ATR on the basis of the results obtained through the tests. Fig. 3.3-2 shows an example of such analyses made on the cooling mechanism of the reactor core with the heavy water moderator system. It was shown by the analysis of a situation: "Small break LOCA resulting from 2% ($0.002152m^2$) break of the downcomer flow area($0.1076m^2$) + Failure of ECCS", it is also expected that the required criteria shall be met since the water pressure within the recirculation system is so strong that Pressure Tubes will be subject to the ballooning effect which eventually increases heat transfer to the heavy water moderator system. As a result of investigation of application for PSA. On the basis of the results of the analysis, we have decided to incorporate the core cooling function of the heavy water cooling system into the success criteria in the case of the small LOCA and transient events (excluding the failure of the heavy water moderator system).

Table 3.3-1 shows the success criteria reflecting above mentioned considerations.

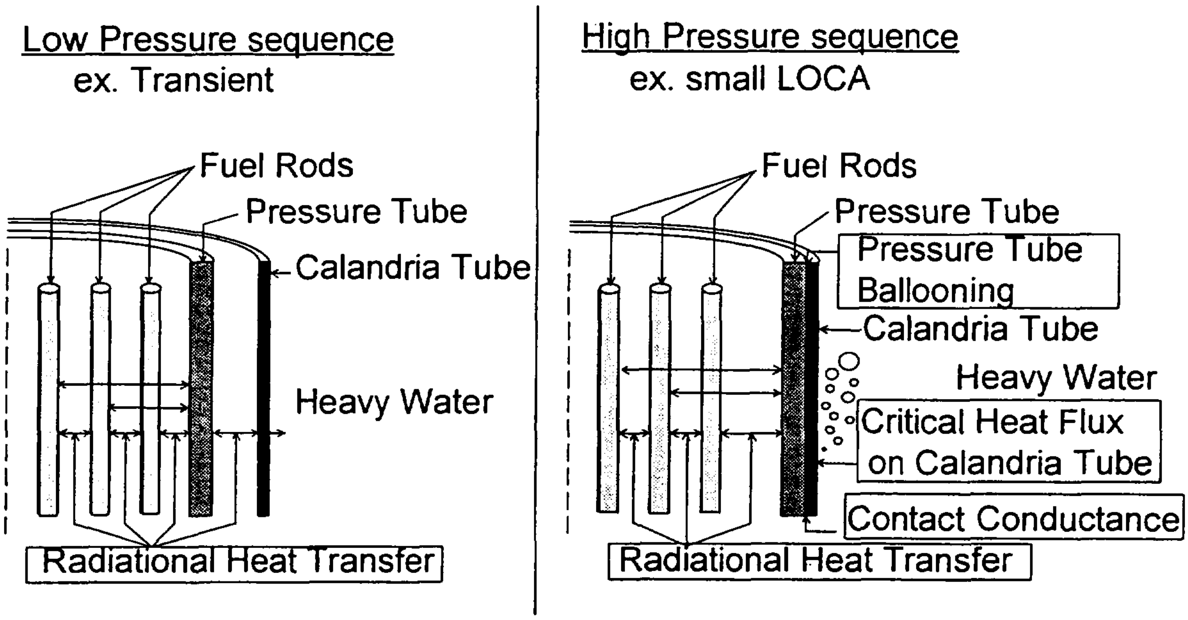


FIGURE 3.3-1 THE CONCEPT OF CORE COOLING BY MCS

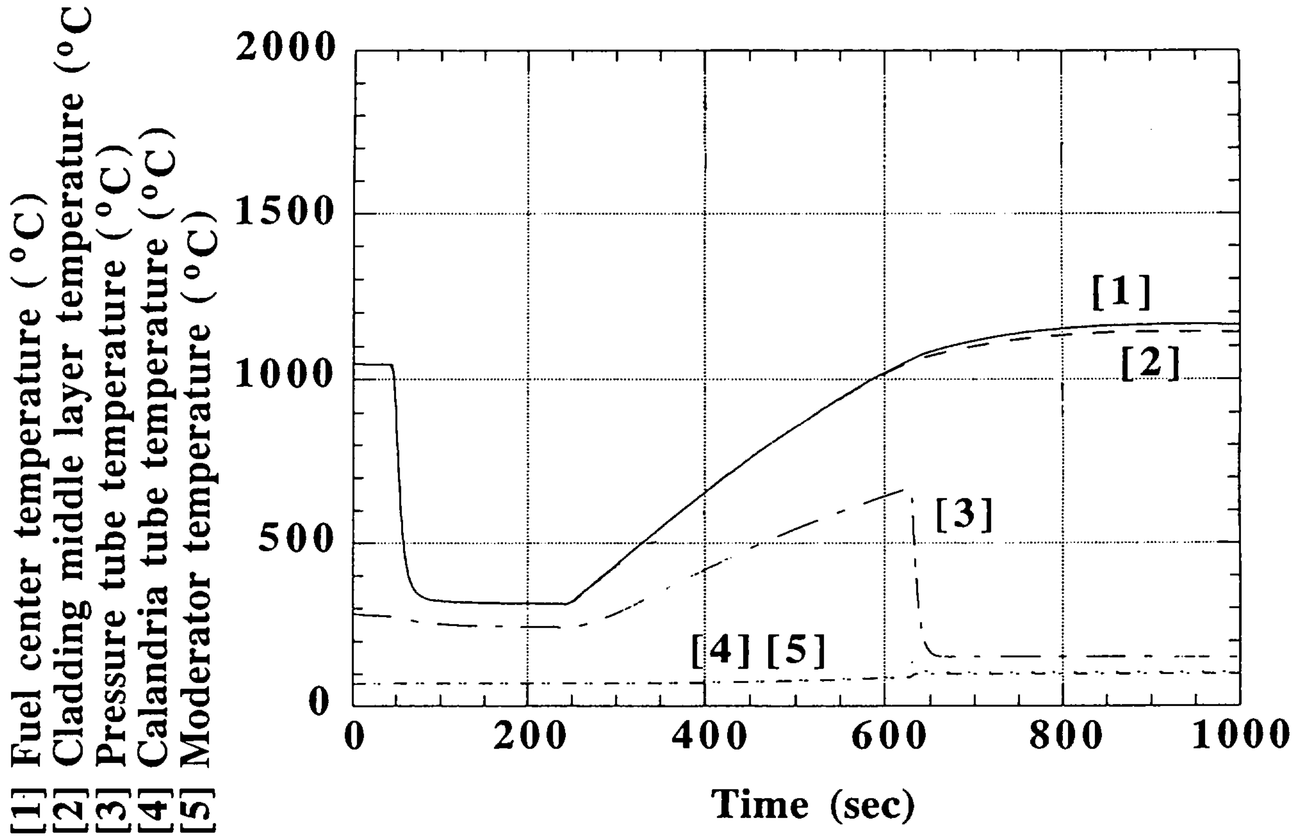


FIGURE 3.3-2 RESULT OF CORE COOLING BY MCS ANALYSIS

TABLE 3.3-1 SUCCESS CRITERIA FOR FUGEN *1

Protective function	Large LOCA*2		Intermediate LOCA*2		Small LOCA*2		Transient*2*3	
	Rupture Loop	Normal Loop	Rupture Loop	Normal Loop	Rupture Loop	Normal Loop		
Reactor Shutdown	[CR] or [HEC]		[CR] or [HEC]		[CR] or [HEC]		[CRD] or [HEC]	
Pressure Reduction + Coolant Injection	[LPCI]	[RCIC] or [HPCI] or [PR]*4	[LPCI]	[RCIC] or [HPCI] or [PR]	[HPCI + LPCI] or [PR+LPCI]	[RCIC] or [HPCI] or [PR]	[PCS]*5or	[RCIC] or [HPCI] or [PR]
Heat Removal function	[SRPC]	[SRPC] or [RHR]	[SRPC]	[SRPC] or [RHR]	[SRPC]	[SRPC] or [RHR]		[RHR] or [SRPC]
Note	* 1 The core damage was defined at over 1200℃ of the peak clad temperature. * 2 MCS is expected as a mitigating system. * 3 CAC is expected as a mitigation system. * 4 PR means Pressure Reduction with the line by passing heat exchanger of RCIC. * 5 PCS is not expected at the Pressure boundary breaking such as failure of Safety Valve reclosure.							

3.4 Development of Event Tree

An Event Tree of seven Initiating Events was developed to delineate accident sequences to be considered for analysis purposes. The characteristics of Fugen were taken into consideration at the development of the Event Trees as follows:

(a) The notable feature of the Event Tree for Fugen is that it has been developed in such a way that two loops displaying two different sets of behaviors can be evaluated concurrently. Furthermore, as shown in Fig. 3.4-1, when the cooling function fails on the unruptured loop due to RCIC or RHR it will become necessary to cool the reactor core by ECCS not only by the ruptured loop but also by the unruptured loop so that a sequence shall develop where cooling by ECCS is required on both loops at the same time. Under such situations, ECCS, where RCIC and RHR have proved successful on the unruptured side, shall be capable of cooling if only one of the two systems is operative, however, both systems will have to be operative if the ECCS should be required also on the unruptured side. As can be seen from the above, the sequence of events will be more complicated with Fugen where the core is cooled by two loops of cooling system and the specific criteria required vary according to the progressed sequence of events . In order to correctly evaluate varying criteria in this study, we have developed the Event Tree by providing each tree with the heading indicating the related frontal system and, as a result, have come to end up with rather a large Event Tree to make us adopt "a large Event Tree with small Fault Trees" approach for Fugen assessment.

(b) A rupture to the large number of Outlet/Inlet pipes and Pressure Tubes that are characteristic of ATR has been effectively reflected by including it in the initiating frequency and small LOCA.

(c) The characteristics of Fugen of using heavy water for the moderator has been reflected as the core cooling effect of the heavy water moderator system in the success criteria. Consequently, the effect of the heavy water moderator system is referred to as the core cooling system in the heading of the Event Tree.

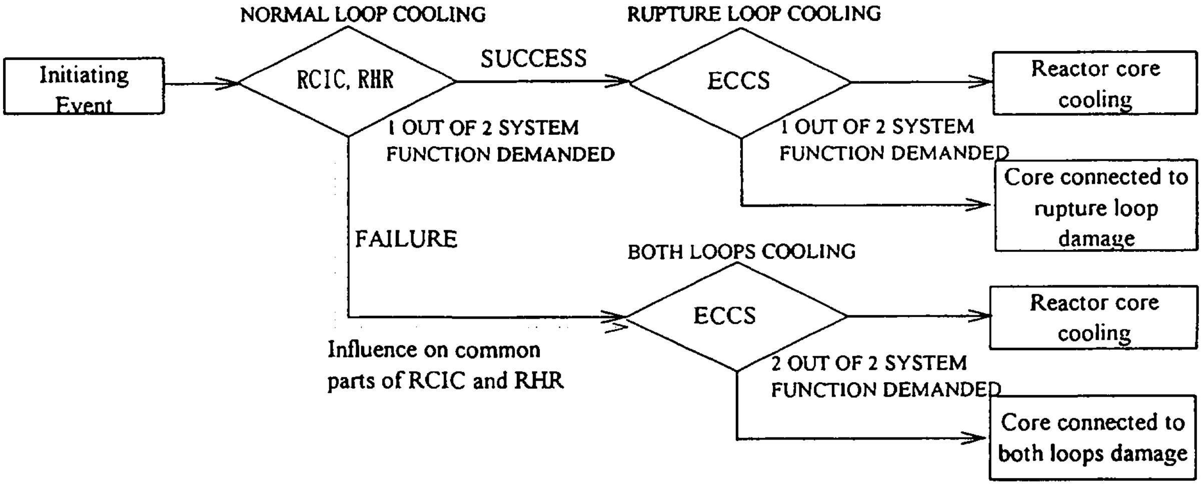


FIGURE 3.4-1 OUTLINE OF EVENT TREE ANALYSIS FOR TWO LOOP SYSTEMS

3.5 Development of Fault Tree

A Fault Tree has been developed with reference to functions required by each system in order to determine the branch probability on Event Tree. As seen in Paragraph 3.4, there are two loops of core cooling system in Fugen and the functions required of each system vary according to the progress of events on each loop. Consequently, the respective branch probabilities on the Event Tree vary depending on anticipated functions even on one and the same system. Under such circumstance, the Fault Tree has been developed using each function emerged on each system in the course of analyzing the Event Tree as the top event.

3.6 Preparation of Database

3.6.1 Component Database

A database for reliability assessment has been developed for the purpose of Fault Tree analysis. It was prepared under the following policy:

1) Whenever there are any data available on the failure of facilities and equipment which reflect the actual results of operation in Japan, such data shall be used before anything else.
2) If there is no domestic database available, ASEP Common Data[3] used for the assessment of NUREG-1150[4] in connection with machinery parts shall be used and, for electrical parts, IEEE Std-500[5] rich with data related to equipment type and failure mode shall be utilized as the basic data.
3) When any necessary data cannot be found in the above mentioned published databases or data found in above were not adequate for the intended purpose, then, LERs (License Event Reports) [6] or WASH- 1400[1] shall be applied.

3.6.2 Common Cause Failure

There may be considered two types of common cause failure: (1) concurrent failures of a number of machines or systems due to a fact that such machines or systems are supported by a common device or support system, and (2) concurrent failures of a number of machines that have been produced under the same specification and manufacturing process and used in the same environment. For the Fault Tree or Event Tree assessment of the machines or systems fallen under (1) above, analysis shall be made by the minimal cut set method or Zion method[7] in consideration of their dependency. Machines fallen under (2) above shall be analyzed by using the β-factor method applied to NUREG-1150[4].

3.6.3 Maintenance and examination

The unavailability of a system for assessment due to a maintenance service shall be quantified by using the following formula:

$$Pm = \Sigma\ (fm.Tm)$$

where:
fm = The frequency of maintenance service (10 times the failure ratio of the subject equipment) [8]
Tm = Average restoration time of the subject machine

Unavailability due to any examination shall be handled in the same way as above.

3.6.4 Human factors

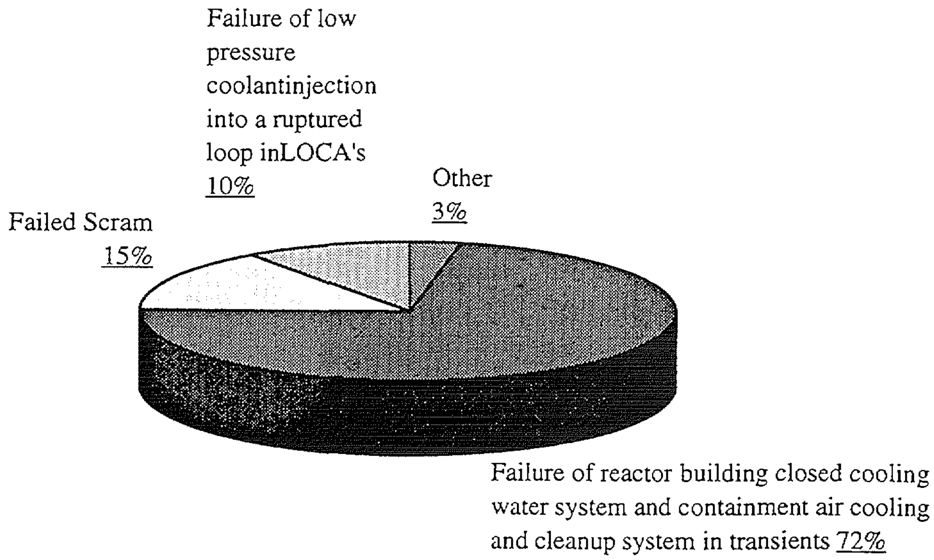
We have also taken into consideration human errors in operation by operators before and after an accident in this study. Human errors have been quantified by using the database of NUREG/CR- 1278[9], THREP procedure and the reliability curve used in NUREG/CR- 4772[10].

3.6.5 Functional Recovery

In case of a machine failure, it is possible to repair the machine to recover its functions if there is ample time available. We have carried out assessment with consideration of such possibility for recovery for some machines such as diesel generator. The success ratio of the functional recovery of a machine shall be assessed by using the following formula:

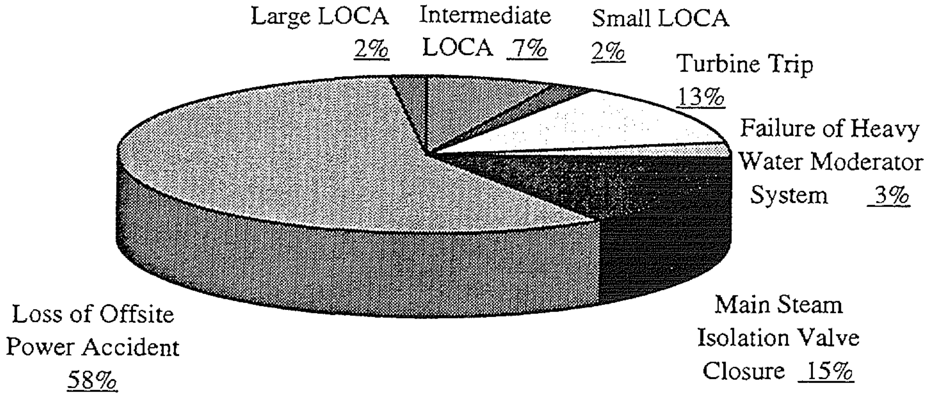
Pr = exp (-T/t)

where:
T = time available for functional recovery
t = the average repair time of the subject machine



(2) DETAILED CLASSIFICATION OF CORE DAMAGE FREQUENCY BY ACCIDENT SEQUENCE

FIGURE 4-1 RESULT OF PSA LEVEL 1 FOR FUGEN



(1) DETAILED CLASSIFICATION OF CORE DAMAGE FREQUENCY BY INITIATING EVENT

3.7 Assessment of the frequency of reactor damage

The reliability of each system was calculated by Fault Tree analysis on the basis of the developed database. An Event Tree analysis was then carried out with the values obtained through Fault Tree analysis or other process for analyzing the branch probability of the Event Tree. Both point estimation and section estimation based on Monte Carlo method were carried in this analysis.

4. ASSESSMENT RESULTS

As the result of the trial calculation of the reactor damage frequency in accordance with the results obtained through Event Tree analyses, it was verified that results including the maximum value of the uncertainty estimation were found to be quite satisfactory against the target value of reactor damage frequency defined by International Atomic Energy Agency (IAEA).

Fig. 4-1 shows the reactor damage frequency grouped by Initiating Event and also by accident sequence. Events initiated by Transients account for 89% of the reactor damage frequency by Initiating Event. Significant reasons for the reactor damage frequency by accident sequence were found to be the failure to remove decay heat in transiens (72%). In a few years, the assessment results will be changed by the system improvement.

5. CONCLUSION

The PSA procedure for ATR Plant was established in this study on the basis of PSA performed on plants in U.S.A. and domestic LWR. Furthermore, the reactor core damage frequency was calculated on ATR by using thus established procedure. As the result, it has been proved that the safety aspects of ATR are appropriately secured when compared with other LWRs.

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