PROBABILISTIC SAFETY ASSESSMENT OF JAPANESE SODIUM-COOLED FAST REACTOR IN CONCEPTUAL DESIGN STAGE

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Abstract

Probabilistic safety assessment was applied to the safety design concept of a loop-type large scale and medium scale of the Japanese sodium-cooled fast reactor (JSFR). Representative core damage sequences initiated from internal abnormal events in power operation were identified and connected to the safety system characteristics. The point estimation and its sensitivity analysis showed that the total core damage frequency of JSFR was sufficiently lower than the reference value of $10^{-6}$ per reactor-year.

1. Introduction

The present paper describes an application of probabilistic safety assessment (PSA) to the safety design concept of a loop-type large scale and medium scale of the Japanese sodium-cooled fast reactor (JSFR), which was performed in the Phase-II (i.e., Japanese fiscal years 2001 - 2005) as part of the feasibility study on commercialized fast reactor cycle systems that has been conducted by Japan Atomic Energy Agency with participation of all the parties concerned in Japan. Since we are aiming at ensuring that the risk from advanced reactors is much lower than the risks encountered in daily activities, without taking into account the need for offsite emergency responses, we set the reference value of the core damage frequency (CDF) that is less than $10^{-6}$ per reactor-year (ry).

2. Analytical model and data sources

Our PSA is based on the standard probabilistic risk assessment method [1], which is broadly used. In order systematically to identify core damage sequences, typical initiating events were selected, considering the similarity of event progression and functional dependency between the initiating events and safety systems. Event trees for each initiating event were developed in a simplified manner based on the plant design features and on some available transient analysis results. Fault trees were also developed to quantify branching probabilities in the event tree, taking into account common-cause failure of active components that have the same design specifications so as to evaluate the effectiveness of the diversity incorporated into the system design.

Quantitative estimation of both the initiating event frequency and component failure rate in the sodium cooling system was based on data consisting of operating time and the number of failure instances stored in the component reliability database system for sodium-cooled fast reactor systems, named CORDS [2]. Those data came from the Japanese fast reactors JOYO and MONJU, and the U.S. fast reactors EBR-II and FFTF. Since there is too little operational experience for steam generators in Japanese fast reactor systems to determine their statistical reliability, the failure rate of steam generator tubes was estimated on the basis of operational experience data for various foreign fast reactor systems. The failure rates of other components in the electrical system, water/steam system, etc. were estimated on the basis of the operational experience of Japanese light water reactors [3].

3. Types of core damage sequences in JSFR

In terms of difference of safety systems required for preventing core damage, typical event sequences leading to core damage are categorized into three types as follows: (1) Anticipated transient without scram (ATWS) that is defined as a failure in rapid reactor shutdown under abnormal transient condition...
(e.g., loss of off-site power), (2) Loss of reactor sodium level (LORL) that is a failure in making up the reactor sodium level under primary cooling system leakage condition, and (3) Protected loss of heat sink (PLOHS): i.e., a failure in maintaining decay heat removal under the adequate reactor sodium level condition after reactor shutdown.

Among these types, only ATWS events have the slightest possibility of bringing about dependent failure of the containment function because of in-vessel retention combined with post-accident heat removal. On the other hand, LORL and PLOHS events could lead to dependent failure of the containment function because the degraded core cannot be cooled, thus resulting in molten fuel leakage through the reactor vessel and containment. Therefore, it is necessary to reduce the frequency of LORL and PLOHS events much lower than the reference value of core damage frequency.

4. ATWS events

Since the reactor shutdown system has not been designed in detail yet, we assumed that JSFR has a main reactor shutdown system and a backup one as shown in Fig. 1. The two reactor shutdown systems having no shared components are double-redundant (i.e., success criterion is at least one out of the two systems). In addition, they consist of different types of active components with inter-system diversity, and each shutdown system is actuated with at least one reactor trip signal against representative initiators from the standpoint of occurrence frequency and consequence. Each system also satisfies a single-rod-stuck condition and a single-failure criterion. In order to enhance the safe shutdown, the self-actuated shutdown system (SASS) consisting of the Curie point electromagnet is installed as a part of the control rod releasing mechanisms in the backup reactor shutdown system.

Figure 1 Schematic diagram of reactor shutdown systems assumed in the present study

Owing to high redundancy and to adequate diversity, the point estimation value of ATWS frequency stayed within the range from $1 \times 10^{-8}$/ry to $3 \times 10^{-8}$/ry, corresponding to an unreliability value for the SASS of 0.1 to 1, respectively. The unreliability of the SASS is defined as probability, with which the reactor shuts down by means of SASS actuation not before core damage. The timing of SASS actuation depends not only on actuation temperature but also on core performances (e.g., reactivity coefficient, average of core outlet temperature around the control rod equipped with the SASS) and cooling system characteristics (e.g., the halving time of the primary flow rate). On this account, the unreliability of the SASS depends on uncertainty of these parameters, and it was shown by calculation that around 0.1 is the unreliability of the SASS in the representative fast reactor with a large-scale core [4]. The present result of ATWS frequency indicates small sensitivity to the SASS reliability. There are two reasons. The one
reason is that the SASS can recover only loss of actuation signals for the backup reactor shutdown system. In other words, the SASS is not effective under the control-rod-stuck situation. The second reason is that we considered a conservative possibility of rod-stuck failure due to an unknown common cause with probability of 2x10^{-5}/scram-demand per backup reactor shutdown system.

There is, however, a quite large uncertainty in estimation of the control-rod-stuck probability because of the lack of instances of failure in sodium-cooled fast reactor operating experience. It cannot be also expected for the control rod to be stuck even though reactor shutdown obviously depends on the active motion of the control rod. This is because once the control rod is released from the control rod drive mechanism after the SASS actuates, the control rod drops into the core through the liquid sodium inside the guide tube with a large clearance. On this account, it is meaningful to understand the sensitivity of ATWS frequency to the rod-stuck failure probability of the backup reactor shutdown system. Assuming that the rod-stuck probability of the backup reactor shutdown system could be reduced by one-tenth, the point estimation value of ATWS frequency fell to the range from 3x10^{-9}/ry to 3x10^{-8}/ry, corresponding to an unreliability of the SASS from 0.1 to 1, respectively. This result indicates that the SASS can be effective under a severe accident condition, although the effect of its introduction depends on the reliability of the rod insertion.

5. LORL events

In order to prevent the LORL event with high reliability, the systematic design measures described below are adopted in JSFR.

1. The primary coolant boundaries in the primary heat transport system (PHTS) are located in the position above the liquid surface level in the reactor vessel in order to eliminate leaking by the siphon effect.

2. The primary coolant boundaries are enclosed with a leak-tight backup structure (i.e., guard vessel and guard pipe) so as to restrict coolant leakage against the boundary failure.

3. Decompressing operations (i.e., PHTS pump trip, isolation of the reactor cover gas from its supply system) are automatically actuated so as to prevent the LORL combined with the above item (1) against double failures in the PHTS boundary and its backup structure.

4. The reactor vessel and its guard vessel have no penetration at either the sides or bottom.

5. The pressure of the secondary heat transport system (SHTS) is kept slightly higher (about 0.1MPa) than that of the PHTS so as to prevent the leaking of primary coolant at the interface breach.

6. The open space between the primary pipe and its guard pipe is partitioned to limit the volume of the leak and prevent LORL by some operator’s action (e.g., control of coolant temperature), even if two of the segments partitioned inside the guard vessels/pipes are fully filled with leaked sodium.

*1: PHTS leakage only at the downstream of PHTS pump requires success in PHTS pump trip.

Figure 2 Event tree diagram to identify loss-of-reactor sodium level sequences under primary cooling system leakage condition

In addition, the inert gas is circulated between the leak-tight backup structure and the inert gas circulation system for the sodium leakage monitoring; the inert gas inside the backup structure is automatically
isolated from its circulation system under PHTS leakage condition. Since the inert gas circulation system is usually designed so as not to maintain its integrity under high temperature condition, we assumed that unless the inert gas circulation system is isolated, its gas boundary becomes breached due to heat shock caused by ingress of the leaked sodium.

In order to identify LORL event sequences, the event tree shown in Fig. 2 was developed on the basis of all the above information. The point estimation value of the LORL frequency became almost $4 \times 10^{-9}$/ry, a very low value. The dominant event sequences leading to LORL are sequences 4 and 5. In these sequences, hot sodium leaked into the piping line of the inert gas circulation system would give a heat shock to the pipe wall, and could result in failure of the pipe wall in the inert gas circulation system and in leakage of sodium into the cell. If the leakage magnitude were restricted by smaller-diameter piping of the inert gas circulation system, there would be a longer time margin to the LORL condition. By then, it could be expected that the operators would have taken some recovery action to prevent the LORL event. It may be better to enhance the integrity of the inert gas system boundary at a high temperature so as to reduce possibility of sodium leakage into the cell as well as possibility of LORL event.

6. PLOHS events

6.1. Safety design features in decay heat removal systems

The present JSFR design adopts a combination of one loop of direct reactor auxiliary cooling system (DRACS) and two loops of primary reactor auxiliary cooling system (PRACS) as shown in Fig. 3. These decay heat removal systems (DHRSs) can be operated under fully passive condition, which means that it is required only to actuate the direct-current-power-operated dampers of the air coolers without pumps and blowers. The damper system has redundancy so that it does not lose its function even considering the single-failure criterion, i.e., each air cooler has two dampers in parallel so that an opening failure of a single damper causes less than a 50% reduction in the air flow rate. In addition, diversity is taken into account in the mechanical design of the dampers between DRACS and PRACS. JSFR is suitable for natural circulation cooling due to its simple and short piping connection and due to the lower pressure loss of the core design, as well as the sufficient height difference between the core and the heat exchangers. Since both DRACS and PRACS have a sodium-sodium heat exchanger inside the PHTS, they are not affected by the abnormal conditions initiated in the SHTS and the steam-water systems.

![Figure 3 Schematic diagram of decay heat removal systems consisting of DRACS and PRACS in a large scale of JSFR](image-url)
In order to understand high reliability of DHRSs in JSFR, it should be noted that there is at least triple redundancy in the coolant circulation function as follows. (1) Although the PHTS has only two loops, the coolant boundary is enclosed with a double-wall structure, so that there is four-fold redundancy in terms of the natural circulation of the coolant. (2) The reactor auxiliary cooling systems with high reliability owing to natural circulation capability have a total of three circuits and at least double redundancy even in the short term after reactor shutdown. (3) In addition to the reactor auxiliary cooling system, it is possible to remove the decay heat by means of one or both of the two loops of the SHTS and the turbine bypass circulation system including steam generators under offsite power condition.

6.2. Quantification result of present design concept of large-scale reactor system

Owing to the above-mentioned safety and reliability features, we evaluated the point estimation value of the PLOHS frequency corresponding to the present design concept of the large-scale reactor system to be nearly $2 \times 10^{-8}$/ry (see Table 1). The dominant sequences of the PLOHS event are listed in Table 2. There would be a sufficiently long time margin to core damage for the operators to take recovery actions as accident managements to prevent core damage. As shown in sequence 3, it has already been considered to allow for manual operation of the air cooler dampers under loss of all electric power sources. Furthermore, the installation of redundant dampers with different actuation mechanisms in a parallel manner to the PRACS air cooler dampers is expected satisfactorily to reduce the frequency of PLOHS sequences including the common-cause failures of the PRACS dampers.

Table 1 Comparison of point estimation values of PLOHS frequency among different design concepts

<table>
<thead>
<tr>
<th>Reactor design concept</th>
<th>Type of DHRS</th>
<th>Number of SHTS loops</th>
<th>Double wall-tube steam generators</th>
<th>Single wall-tube steam generators</th>
</tr>
</thead>
<tbody>
<tr>
<td>Large-scale reactor</td>
<td>Present design</td>
<td>2 PRACS + 1 DRACS</td>
<td>2</td>
<td>$2 \times 10^{-8}$/ry</td>
</tr>
<tr>
<td>Medium-scale reactor</td>
<td>Previous design</td>
<td>2 IRACS + 1 DRACS</td>
<td>1</td>
<td>$4 \times 10^{-8}$/ry</td>
</tr>
<tr>
<td>Large-scale reactor</td>
<td>Present design</td>
<td>2 PRACS + 1 DRACS</td>
<td>2</td>
<td>$1 \times 10^{-7}$/ry</td>
</tr>
<tr>
<td>Medium-scale reactor</td>
<td>Previous design</td>
<td>2 IRACS + 1 DRACS</td>
<td>1</td>
<td>$7 \times 10^{-7}$/ry</td>
</tr>
</tbody>
</table>

Table 2 Dominant sequences of the PLOHS event

<table>
<thead>
<tr>
<th>Sequence No.</th>
<th>Description of event sequences</th>
<th>Contribution</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Loss of feedwater &amp; failure to open the PRACS dampers due to common cause</td>
<td>47.6%</td>
</tr>
<tr>
<td>2</td>
<td>Loss of off-site power &amp; failure to open the PRACS dampers due to common cause</td>
<td>25.8%</td>
</tr>
<tr>
<td>3</td>
<td>Loss of off-site power &amp; common cause failure of batteries &amp; failure manually to open the PRACS dampers due to common cause operator error</td>
<td>19.5%</td>
</tr>
<tr>
<td>4</td>
<td>The others</td>
<td>7.1%</td>
</tr>
</tbody>
</table>

It was found that, in the design concept combining DRACS and PRACS, the PLOHS frequency increased only about 50% even if the steam generators with double-wall heat-transfer tubes were replaced by those with single-wall tubes. The reason is described as follows. First, a failure of steam generator tubes does not affect the decay heat removal function of DRACS and PRACS. In addition, the contribution of steam generator tube rupture to loss of decay heat removal with the turbine bypass circulation system is small. This is because, in the case of minor leakage in a steam generator, decay heat removal is still available although the occurrence frequency of tube failure per steam generator could be relatively high. In the other case of large-scale leakage, the occurrence frequency would be sufficiently low although the decay heat removal function might be lost.

6.3. Quantification result of the other design concepts

Since the medium-scale reactor has only a single loop of SHTS as well as the feedwater system, a reactor trip induced from a single failure in SHTS results in loss of decay heat removal by means of the steam generator. In case of SHTS having the double-wall tube steam generator, an SHTS pump failure becomes a dominant contributor to the occurrence frequency of SHTS failures. On the other hand, in case of the
single-wall tube steam generator, a steam generator tube leakage is dominant because of higher frequency. The occurrence frequency of SHTS pump failure is comparable to that of loss of feedwater and loss of offsite power that appear in sequences 1 through 3 of Table 2. Therefore the PLOHS frequency of medium-scale reactor is higher than that of large-scale reactor as shown in Table 1.

In the previous design concept of large scale reactor with DRACS and Intermediate reactor auxiliary cooling system (IRACS), a failure of the SHTS coolant boundary is a dominant contributor to the PLOHS event because the IRACS and turbine bypass circulation system share the SHTS with reducing redundancy for DHRS. In addition, there was a large uncertainty in estimating the probability of the SHTS coolant boundary failure as follows: (1) point estimation value of occurrence frequency of steam generator tube failure changes within a factor of at least five, depending on whether the reliability of the steam generator tube in JSFR is regarded as comparable to that in domestic LWRs or whether the reliability is estimated based on the steam generator experiences in foreign sodium-cooled fast reactors; and (2) the piping failure probability also has a large uncertainty in the estimation due to rare events. Given these uncertainty factors, the probability of the SHTS coolant boundary failure was conservatively estimated. It was found that the point estimation value of PLOHS frequency with the IRACS system including a single-wall tube steam generator, increased nearly up to the CDF reference value of $10^{-6}$/ry.

### 7. Conclusion

The obtained value of total CDF within the limited scope was sufficiently less than the reference value $10^{-6}$/ry. The sum of LORL and PLOHS frequencies was comparable to the ATWS frequency. The former, however, can be satisfactorily reduced in terms of maintaining the containment function with additional accident managements. Since it is indispensable to evaluate the contribution to the risk of measures against the external initiators in order to restrict the total risk induced from various hazardous events, we will consider external events in the next phase. The PSA study, which considered the uncertainty resulting primarily from lack of steam generator experience, led to a change in the DHRS concept from IRACS to PRACS in order to attain the CDF requirement. We believe that this system design change is adequate in terms of safety because we made the decision with integrating all the information related to the safety and reliability including the uncertainty that can be presently obtained. It is still important, however, to understand the reliability characteristics of the advanced components with no or little operational experiences (e.g., steam generators); we need to continuously accumulate operational experience data on the advanced components, not only in research and development but also through field operation after JSFR commissioning.

### 8. References


