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A Study on Low Power and Shutdown Level 2 PSA for APR1400 DC Project

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1. Introduction

In the APR1400 nuclear power plant NRC DC (Design Certification) project, Level 1/ Level 2 PSA analysis were carried out with regard to internal and external accidents for full power and low power shutdown (LPSD) operations.

The methodology employed was in accordance with the safety evaluation methods of the U.S. ASME/ANS RA-Sa-2009 and NUREG-1150, as well as the method presented in RG 1.200. In particular, the low power shutdown PSA model for the plant reflects the operation conditions of the plant planned outage, so shutdown operation optimization for the plant becomes possible during the maintenance and testing of each system. In this study, the methods of POS Analysis, PDS Analysis, STC Analysis, Initiating Events Analysis, DATA Analysis, and Accident Sequence Analysis for Level 2 internal accident PSA during low power shutdown operation were explained and their evaluation results were detailed.

2. Analysis and Results

2.1 Plant Operating States (POS) Analysis

Analysis was carried out for each LPSD operation, and a conservative evaluation method was applied for POS with low CDF. The most severe accident sequences during LPSD operations are similar to those during full power operations, and the containment event tree accident scenarios are also similar. However, opening of the equipment hatch and pressurizer manway occur only during LPSD operation, so this is not similar to the Containment Event Tree (CET) accident scenario during full power operation and was considered in the accident scenario. For the POS, analysis was carried out to reflect the developed power plant arrangement classification based on the 6 plant operation modes detailed in the technical specifications manual.

Table 1. Power plant operation modes (6 modes)

Operation mode	Reactivity condition	Rated Thermal Power
Power operation	≥ 0.99	> 5%
Startup	≥ 0.99	$\leq 5\%$
Hot standby	< 0.99	N/A
Hot S/D	< 0.99	N/A
Cold S/D	< 0.99	N/A
Refueling	N/A	N/A

2.2 Plant Damage States (PDS) Analysis

Level 2 PSA during LPSD operation was not grouped into PDS, unlike the case of the analysis for the full power operation; the core damage accident scenario was associated with CET. A single CET model considering the pressurizer manway opening was developed and applied for POS 4B, 5, 6, 10, 11, and 12A. For POS 1, 2, 3A, 13, and 14, Conditional Probability of a Large Release (CPLR), conservatively obtained through analysis for Level 2 PSA during full power operation, was applied and quantified. For POS 3B and 4A, the nuclear containment building equipment hatch can be opened, so analysis was conservatively carried out with the equipment hatch assumed to be opened during the entire duration of the corresponding POS.

Table 2. Plant Operating States (17 states)

POS	POS Description	Comment
1	Low power operation	C/V Closed, at-power CPLR
2	SG Cooldown to 350F	C/V Closed, at-power CPLR
3A	Cooldown with SCS to 212F	C/V Closed, at-power CPLR
3B	Cooldown with SCS to 140F	Hatch close HEP/at-power CPLR
4A	RCS Draindown (M/W closed)	Hatch close HEP/at-power CPLR
4B	RCS Draindown (M/W open)	C/V Closed
5	Reduced Inventory Operation	C/V Closed
6	Fill for Refueling	C/V Closed
7	Refueling (Core-alteration)	-
8	Cavity drained	-
9	Refueling (Core-alteration)	-
10	RCS Draindown after refueling	C/V Closed
11	Reduced Inventory Operation	C/V Closed
12A	Refill RCS (M/W open)	C/V Closed
12B	Refill RCS (M/W closed)	-
13	RCS Heatup/SCS Isolate at 350F	C/V Closed, at-power CPLR
14	RCS Heatup with SGs	C/V Closed, at-power CPLR
15	Reactor Startup	C/V Closed, at-power CPLR

2.3 Source Term Category (STC) Analysis

The full power operation Level 2 PSA was defined as 21 STCs. However, for a conservative approach to LPSD operation Level 2 PSA, only 4 were defined through simplification. Among these 4 STCs, RC (Release Category)-1-LPSD and RC-4-LPSD correspond to the release groups RC10 and RC11 during full power operation. RC-2-LPSD and RC-3-LPSD correspond to large release; the LRF (Large Release Frequency) was analyzed using the sum of these 2 STCs.

2.4 Initiating EventsAnalysis

Under the POS of low power operation and heat removal through the steam generator, transient scenarios that induce shutdown of the reactor or that hamper the secondary heat removal were selected as the initial scenario. When the shutdown cooling system is operating, accidents including shutdown cooling malfunction, reactor coolant loss, and thermal power increase were selected as inducing factors. The final initial scenarios were selected and analyzed considering the analysis subject nuclear power plant design and operation characteristics for the selected factors.

2.5 Data Analysis

Data analysis was conducted by analyzing human reliability, initiating event frequency, and component reliability.

2.5.1 Human reliability data

For human reliability, the modeling and error probability were evaluated according to the behavior of the operators, as considered in the event tree and fault tree.

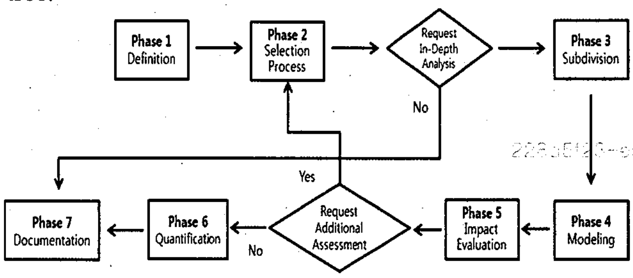

Fig. 1 HRA Flowchart for APR1400 DC Project

2.5.2 Initiating event frequency

The initiating event frequency refers to the frequency of initiating event occurrence for each operating state of each selected initial accident. The initiating event frequency was evaluated mainly using data from Shin-Kori Units 1 and 2, along with Surry Unit 1 LPSD PSA carried out by the US NRC.

Table 3. Distribution of CDF in initiating event group

Initiating event	Contribution (%)
Excess water supply	31
Plant power outage	23
Shutdown cooling	15
Shutdown cooling malfunction that cannot be partially restored	12
Pilot operated safety relief valve	9
Misc.	10

2.5.3 Component reliability data

The reliability database necessary for the system fault tree and core damage accident scenario quantification was used as the data for component reliability. For category B, overhaul of the primary component coolant system, primary component seawater coolant system, essential cooling system, safety injection pump, shutdown cooling pump and containment building spray pump were carried out from POS 7 to POS 8. When the overhaul of category B was completed, overhaul of category A was conducted for evaluation from POS 8 to POS 9.

2.6 Accident Sequence Analysis

In the accident scenario quantification process, assessment of the cutsets were performed to determine whether the PSA models of the fault tree and the event tree were appropriately constructed and whether there were any errors. Also, errors in the utilized reliability data were inspected. Then, operator error reliability evaluation and recovery operation analysis were carried out to finally obtain the total core damage minimal cutset and core damage frequency for each accident scenario and initiating event.

3. Conclusions

The LRF for the PSA Level 2 internal accident during LPSD operation was high for POS 5 and POS 11, which correspond to the mid-loop operation; however, consideration of the SIP operation in accordance with the Severe Accident Management Guideline contributed to reducing the LRF value. For the next 3 items of POS 3A, 3B-JL, and POS 3B-other, LRF was evaluated by applying CPLR according to the analysis results of Level 2 during full power operation.

The evaluation results revealed that the sum for the 5 POS items was roughly 3/4 of the total POS CDF value. Analysis results showed that the reduced inventory and high core residual heat operations require the greatest caution.

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