A Methodology for Reliability Assurance Program in Low Power Shutdown Probabilistic Safety Assessment

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1. Introduction

The primary objective of this study is to provide the methodology for the APR1400 Reliability Assurance Program (RAP), in compliance with the requirements of the U.S. Nuclear Regulatory Commission's Standard Review Plan [1], Section 17.4.

The RAP during Lower Power Shutdown (LPSD) operation applies to ~~those~~ both safety-related and non-safety-related systems, structures, and components (SSCs) that are identified as being risk-significant (or significant contributors to plant safety). The SSCs within the scope of the RAP are identified by using a combination of probabilistic, deterministic, or other methods of analysis, including information obtained from sources such as the probabilistic risk assessment (PRA), severe accident evaluations, industry operating experience, and expert panels.

The purposes of the RAP during LPSD are to provide reasonable assurance of the following:

- A reactor is designed, constructed, and operated consistent with the key assumptions and risk insights for the within-scope SSCs
- The SSCs function is reliable when challenged

The process of identifying risk-significant SSCs begins with a review of the PRA model. The PRA staff generates a list of SSCs that meet numerical criteria for potential classification as risk-significant (RS). This list is provided to the Expert Panel for review. The panel reviews the PRA recommendations, supplements them with qualitative (deterministic) information and makes a final determination as to whether the SSCs should be classified as risk significant. If they are RS, then the SSCs are added to the RAP list.

2. Quantitative Input

2.1 PRA Criteria

The RAP list is a primary source of input for the plant's Maintenance Rule program, in compliance with Reference 2, once plant operation commences. Therefore, it specifies criteria that will align the RAP with the anticipated Maintenance Rule criteria, as much as possible. The NRC has endorsed NUMARC 93-01 [3] as an acceptable. The endorsement is documented in Reference 4. Reference 3 recommends the following PRA importance measures (criteria) for classification as risk-significant:

- A basic event has a Risk Achievement Worth (RAW) of 2.0 or more. This threshold means that if a component failure increases PRA risk by a factor of at least 2X, then the event should be considered for classification as risk significant.
- A basic event has a Fussell-Vesely (FV) importance of 0.005. This threshold means that the failure contributes at least 0.5% to the plant risk.

In addition, Reference 5 also recommends criteria that have been used by the industry for other Reliability Assurance Programs. These include the following:

- If the component appears in a Common Cause Failure (CCF) in the PRA model with a RAW of at least 20, then the component is recommended for classification as risk significant.

2.2 SSC Identification

RAP Systems, Structures and Components (SSCs) were identified as follows. There are six separate PRA model quantifications of core damage risk for APR1400 internal events, internal fires and internal flooding at both full power and shutdown. Each analysis generates its own, unique set of importance measures, each of which has been reviewed to identify basic events for individual component failures and component CCF events.

2.3 DFM Identification

The RAP Dominant Failure Modes (DFMs) were identified as follows. Each risk-significant SSC has at least one, and sometimes several, DFMs. Each PRA component's basic events model has one or several failure modes. For example, a circuit breaker can fail to trip or spuriously re-position. If any component meets the PRA criteria for risk significance due to a specific failure mode, then that mode should be evaluated as potentially dominant.

3.4 Redundant Trains

It is possible that the basic events for one train can meet the criteria for risk-significance, while those for the opposite train do not. As a conservative measure, if the criteria are met for one train, then both trains are recommended for classification as risk-significant.

3. Qualitative Input

The numerical review described in the previous section 2 is only the starting point for the process of identifying risk-significant SSCs for the RAP list. Those quantitative criteria identify components that are recommended for classification as RS.

The Expert Panel supplements the PRA recommendations with the available, qualitative information, and accepts or rejects the recommendations. The panel makes the final determination on both SSCs and their DFMs.

4. Conclusions

The Expert Panel have classified the SSCs lists as risk-significant, and their associated Dominant Failure Modes. The PRA staff reviewed the quantification results in PRA and made their recommendations for SSC risk significance and DFMs. These recommendations were also reviewed by the Expert Panel (EP); the reviews and results are documented. In particular, the PRA staff and EP reviews the risk insights from DCD, and address the potential RAP impact of each. Table 1 through 3 show the RAP analysis results for the shutdown cooling system (SCS) as a part of RAP analysis in LPSD PRA.

Table 1 lists the importance measures for every individual basis event (RAW>2 or FV>0.005), and every common cause failure basic event (RAW>20) that met the PRA criteria for potential classification as risk significant in LPSD PRA. Table 2 includes the list of risk significant equipment, as identified by the RAP Expert Panel their basis for classification and their dominant failure modes in LPSD PRA. Table 3 shows Risk insight for RAP Impact as the results of RAP analysis in LPSD PRA.

Table Ⅰ: Importance Data for PRA Risk-Significant Basic Events

Basic Event	Description	F&V(%)	RAW
SICVO2B-V168	Check Valve V168 in SCS Train B HX Discharge Path Fails to open	0.0	2.7
SICVWQ4-V157/158/159/160	4/4 CCF of CS Check Valve V157/158 SC Check Valve V159/160 Fail to open	0.0	1353.6
SICVWQ4-V568/569/1001/1002	4/4 CCF to Open CSP Discharge CVs 1001 and 1002 and SCP Discharge CVs 568 and 569	0.0	1353.6
SIHEY1B-HE01B-SC	SC HX 2 HE01B Fails to Operate	0.0	2.7
SIMPKQ3-CSP1A/SCP1A/B	3/4 CCF of CSP PP01A and SCP PP01A/PP01B Fail to run	0.0	24.9
SIMPKQ3-CSP1B/SCP1A/B	3/4 CCF OF CSP PP01B and SCP PP01A/PP01B Fail to run	0.0	24.7

Table Ⅱ: RAP Systems, Structures and Components

System	SSC ID	SSC Description	Risk Significance Basis	Dominant Failure Modes
SI	CV159/160	IRWST Suction Check Valves	Level 1 LPSD Level 2 LPSD	CCF to open
SI	PP01A/B	Shutdown Cooling Pumps	Level 1 LPSD Level 2 LPSD	CCF to start Fail to run
SI	CV568/569	Shutdown Cooling Pump Discharge Check Valves	Level 1 LPSD Level 2 LPSD	Fail to open
SI	HE01A/B	Shutdown Cooling Heat Exchangers	Level 1 LPSD Level 2 LPSD	Loss of heat transfer
SI	CV168/178	Shutdown Cooling Heat Exchanger Discharge Check Valves	Level 1 LPSD Level 2 LPSD	Fail to open

Table Ⅲ: Review of Risk Insight for RAP Impact

Insight	RAP Impact
The following are some important aspects of the shutdown cooling system (SCS) as represented in the PRA: The SCS has two separate and redundant trains, each with the heat removal capacity to cool the RCS to cold shutdown conditions. The SC and CS pumps are designed to be independent, but identical and functionally interchangeable. Either pump in a division can provide flow to either the CSS header or the SCS heat exchanger. During plant shutdown operations, the SCS can be aligned to the IRWST to provide RCS inventory makeup. The SC pump trains are powered from the A and B trains; hence, during an SBO, power from the AAC can be supplied to either SC pump.	The shutdown cooling pumps, heat exchangers and many of their valves are on the RAP list.

REFERENCES

[1] NUREG-0800, Standard Review Plan, Section 17.4, “Reliability Assurance Program,” Rev. 1, U.S. Nuclear Regulatory Commission, May, 2014.
[2] 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, Washington, DC, August, 2007.
[3] NUMARC 93-01, “Industry Guideline For Monitoring the Effectiveness of Maintenance At Nuclear Power Plants,” Rev. 4A, April, 2011. (NEI has also issued revisions 4B and 4C, but Reference 7 has only endorsed Revision 4A at this time.)
[4] Regulatory Guide 1.160, Rev. 3, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” May, 2012.

[5] EPRI 1023008, "Advanced Nuclear Technology: Design Reliability Assurance Program Implementation Guidance," December, 2011.