R4-6B2

State the performance deficiency exactly as stated here:

Operators failed to restore the normal valve alignment for the Division II service water pump gland water supply following maintenance and prior to returning the system to service. This configuration resulted in the Division II service water gland sealing system being provided by the Division I service water pumps. In this configuration, a failure of the Division I pumps would result in loss of gland water to the Division II pumps.

Analysis

Minor Determination: In accordance with NRC Inspection Manual Chapter 0612, Appendix B, "Issue Screening," the inspectors determined that the failure to properly realign the system was a licensee performance deficiency because the system was returned to service in a condition that failed to meet the operability requirements of Technical Specification 3.7.2. This specification requires that both divisions of service water be operable. Additionally the failure to properly align the gland water system was fully within the licensee's abilities to control. The issue was more than minor because it was similar to Example 4.e in Manual Chapter 0612, Appendix E, "Examples of Minor Issues," and it met the "not minor if" criteria, in that the error resulted in improper valve manipulation (alignment).

Phase 1 Screening: The inspectors evaluated the issue using the SDP Phase 1 Screening Worksheet for the Initiating Events, Mitigating Systems, and Barriers Cornerstones provided in Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." This issue caused an increase in the likelihood of an initiating event, namely loss of service water, as well as increasing the probability that the service water system would not be available to perform its mitigating systems function. Therefore, the issue was passed to Phase 2.

Phase 2 Screening: In accordance with Manual Chapter 0609, Appendix A, Attachment 1, "User Guidance for Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors evaluated the subject finding using the Risk-Informed Inspection Notebook for Cooper Nuclear Station, Revision 1. An exposure window of 3 - 30 days was used because the condition existed for 21 days.

The analyst assumed that the failure of gland water cooling to a service water pump will result in the failure of the pump to meet its risk-significant function resulting in an increased likelihood that all service water would be lost. Therefore, the initiating event likelihood credit for loss of service water system was increased from five to four by the senior reactor analyst in accordance with Usage Rule 1.2 in Manual Chapter 0609, Appendix A, Attachment 2, "Site Specific Risk-Informed Inspection Notebook Usage Rules." This change reflects the fact that the finding increased the likelihood of a loss of service water, a normally cross-tied support system.

For the purposes of the Phase 2 estimation, the analyst assumed that the configuration of the service water system did not increase the probability that the system function would be lost by an order of magnitude because both pumps in Division I would have to be lost before the condition would affect Division II. Therefore, the order of magnitude assumption was that the service water system would continue to be a multi-train system.

Q-2

The analyst solved the Loss of Service Water System worksheet, and using the counting rule worksheet, this finding was estimated to be YELLOW. However, because several assumptions made during the Phase 2 process were overly conservative, a Phase 3 evaluation was required.

Phase 3 Evaluation: The analyst conducted an evaluation of the finding using a Standardized Plant Analysis Risk (SPAR) model simulation of the cross-tied service water divisions, as well as an assessment of the licensee's evaluation provided by the licensee's probabilistic risk assessment staff (Glen A. Seeman). The SPAR runs were based on the following analyst assumptions:

a. The Cooper SPAR model was revised to better reflect the failure logic for the service water system. This model, including the component test and maintenance basic events, represents an appropriate tool for evaluation of the subject finding.

b. NUREG/CR-5496, "Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980 - 1996," contains the NRC's current best estimate of both the likelihood of each of the loss of offsite power (LOOP) classes (i.e., plant-centered, grid related, and severe weather) and their recovery probabilities.

c. The service water pumps at Cooper will fail to run if gland water is lost for 30 minutes or more. If gland water is recovered within 30 minutes of loss, the pumps will continue to run for their mission time, given their nominal failure rates.

d. The condition existed for 21 days from January 25 through February 11, 2004 representing the exposure time.

e. The nominal likelihood for a loss of service water at the Cooper Nuclear Station is as stated in NUREG/CR-5750, "Rates of Initiating Events at Nuclear Power Plants: 1987 - 1995," Section 4.4.8, "Loss of Safety-Related Cooling Water System." This reference documents a total loss of service water frequency at 9.72×10^{-4} per critical year.

f. The nominal likelihood for a partial loss of service water at the Cooper Nuclear Station is as stated in NUREG/CR-5750, "Rates of Initiating Events at Nuclear Power Plants: 1987 - 1995," Section 4.4.8, "Loss of Safety-Related Cooling Water System." This reference documents a partial loss of service water frequency (loss of single division) at 8.92×10^{-3} per critical year.

g. The configuration of the service water system increased the likelihood that all service water would be lost. The increase in loss of service water initiating event likelihood best representing the change caused by this finding is one half the nominal likelihood for the loss of a single division. The analyst noted that the nominal value represents the likelihood that either division of service water is lost. However, for this finding, only losses of Division I equipment result in the loss of the other division.

h. The SPAR HRA method used by Idaho National Engineering and Environmental Laboratories during the development of the SPAR models and published in Draft NUREG/CR-xxxxx, INEEL/EXT-02-10307, "SPAR-H Method," is an appropriate tool for evaluating the probability of operators recovering from a loss of Division I service water.

i. The probability of operators failing to properly diagnose the need to restore Division II service water gland water upon a loss of Division I service water is 0.4. This assumed the nominal diagnosis failure rate of 0.01 multiplied by the following performance shaping factors:

- Available Time: 10. The available time was barely adequate to complete the diagnosis. The analyst assumed that the diagnosis portion of this condition included all activities to identify the mispositioned valves.

- Stress: 2. Stress under the conditions postulated would be high. The operators would understand that the consequences of their actions would represent a threat to plant safety.

- Complexity: 2. The complexity of the tasks necessary to properly diagnose this condition was determined to be moderately complex. The analyst determined that there was some ambiguity in the diagnosis of this condition.

The SPAR Revision 3.03 model was modified to include updated loss of offsite power curves as published in NUREG CR-5496, as stated in Assumption b. The changes to the loss of offsite power recovery actions and other modifications to the SPAR model were documented in Table 2. In addition, the failure logic for the service water system was significantly changed as documented in Assumption a. These revisions were incorporated into a base case update, making the revised model the baseline for this evaluation. The resulting baseline core damage frequency, CDF_{base}, was 4.82×10^{-9} /hr.

The analyst changed this modified model to reflect that the failure of the Division I service water system would cause the failure of the gland water to Division II. Division II was then modeled to fail either from independent divisional equipment failures, or from the failure of Division I. The analyst determined that the failure of Division II could be prevented by operator recovery action. As stated in Assumption **, the analyst assumed that this recovery action would fail 40 percent of the time. The model was requantified with the resulting current case conditional core damage frequency, CDF_{case}, of 1.74×10^{-8} /hr.

The change in core damage frequency (ΔCDF) from the model was:

$$\begin{aligned}\Delta CDF &= CDF_{case} - CDF_{base} \\ &= 1.74 \times 10^{-8} - 4.82 \times 10^{-9} = 1.26 \times 10^{-8} \text{ /hr.}\end{aligned}$$

Therefore, the total change in core damage frequency over the exposure time that was related to this finding was calculated as:

$$\Delta CDF = 1.26 \times 10^{-8} \text{ /hr} * 24 \text{ hr/day} * 21 \text{ days} = 6.35 \times 10^{-6} \text{ for 21 days}$$

Risk Associated with External Initiators: In accordance with Manual Chapter 0609, Appendix A, Attachment 1, Step 2.5, "Screening for the Potential Risk Contribution Due to External Initiating Events," the analyst assessed the impact of external initiators because the Phase 2 SDP result provided a Risk Significance Estimation of 7 or greater.

Seismic, High Winds, Floods, and Other External Events: The analyst determined, through plant walkdown, that the major divisional equipment associated with the service water system were on the same physical elevation as its redundant equipment in the alternate division. All four service water pumps are located in the same room at the same elevation. Both primary switchgear are at the same elevation and in adjacent rooms. Therefore, the likelihood that internal or external flooding and/or seismic events would affect one division without affecting the other was considered to be extremely low. Likewise, high wind events and transportation events were assumed to affect both divisions equally.

Fire: The analyst evaluated the list of fire areas documented in the IPEEE, and concluded that the Division I service water system could fail in internal fires that did not directly affect Division II equipment. These fires would constitute a change in risk associated with the finding. The analyst identified two fire areas of concern: Pump room fires and a fire in Switchgear 1F.

Service water pump room fires were determined to represent only a nominal increase in total risk because fires had an equivalent chance of affecting Division II equipment as Division I. Using the revised baseline and current case SPAR models, the analyst determined that the increase in risk associated with pump room fires was 4×10^{-11}/hr.

The analyst also assessed the affect of this finding on a postulated fire in Switchgear 1F. The analyst walked down the switchgear rooms and interviewed licensed operators. The analyst identified that, by procedure, a fire in Switchgear 1F would require deenergization of the bus and subsequent manual scram of the plant. Additionally, the analyst noted that no automatic fire suppression existed in the room. Therefore, the analyst used the fire ignition frequency stated in the IPEEE, namely 3.70×10^{-3}/yr ($L_{switchgear}$), as the frequency for loss of Switchgear 1F and a transient.

The analyst used the revised baseline and current case SPAR models to quantify the conditional core damage probabilities for a fire in Switchgear 1F. The resulting CCDPs were 1.88×10^{-4} ($CCDP_{base}$) for the baseline and 1.70×10^{-2} ($CCDP_{current}$). The change in core damage frequency was calculated as follows:

$$\Delta CDF = L_{switchgear} * (CCDP_{current} - CCDP_{base})$$

$$= 3.70 \times 10^{-3}/yr \div 8760 \text{ hrs/yr} * (1.70 \times 10^{-2} - 1.88 \times 10^{-4})$$

$$= 7.10 \times 10^{-9}/hr$$

The total change in external risk was, therefore, calculated as 7.14×10^{-9}/hr. Over the exposure time of 21 days, this represented a ΔCDF from external initiators of 3.6×10^{-6} over the exposure period.

Risk Contribution from Large Early Release Frequency (LERF): In accordance with Manual Chapter 0609, Appendix A, Attachment 1, Step 2.6, "Screening for the Potential Risk Contribution Due to LERF," the analyst assessed the impact of large early release frequency because the Phase 2 SDP result provided a risk significance estimation of 7.

In BWR Mark I containments, only a subset of core damage accidents can lead to large, unmitigated releases from containment that have the potential to cause prompt fatalities prior to population evacuation. Core damage sequences of particular concern for Mark I containments are ISLOCA, ATWS, and Small LOCA/Transient sequences involving high reactor coolant system pressure. A loss of service water is a special initiator for a transient. Step 2.6 of Manual Chapter 0609 requires a LERF evaluation for all reactor types if the risk significance estimation is 7 or less and transient sequences are involved.

In accordance with Manual Chapter 0609, Appendix H, "Containment Integrity SDP," the analyst determined that this was a Type A finding, because the finding affected the plant core damage frequency. The analyst evaluated both the baseline model and the current case model to determine the LERF potential sequences and segregate them into the categories provided in Appendix H, Table 5.2, "Phase 2 Assessment Factors - Type A Findings at Full Power.

Following each model run, the analyst segregated the core damage sequences as follows:

- Loss of coolant accidents were assumed to result in a wet drywell floor. The analyst assumed that during all station blackout initiating events the drywell floor remained dry. The Cooper Nuclear emergency operating procedures require drywell flooding if reactor vessel level can not be restored. Therefore, the analysts assumed that containment flooding was successful for all high pressure transients and those low pressure transients that had the residual heat removal system available.

- All Event V initiators were grouped as intersystem loss of coolant accidents (ISLOCA)

- Transient Sequence 65, Loss of dc Sequence 62, Loss of service water system Sequence 71, small loss of coolant accident Sequence 41, medium loss of coolant accident Sequence 32, large loss of coolant accident Sequence 12, and LOOP Sequence 40 cutsets were considered anticipated transients without scram (ATWS)

- All LOOP Sequence 39 cutsets were considered Station Blackouts. Those with success of safety-relief valves to close or a single stuck-open relief valve were considered high pressure sequences. Those with more than one stuck-open relief valve were considered low pressure sequences.

- Transients that did not result in an ATWS were assumed to be low pressure sequences if the cutsets included low pressure injection, core spray, or more than one stuck-open relief valve. Otherwise, the analyst assumed that the sequences were high pressure.

- Small break loss of coolant accident, Sequence 1 cutsets, that represent stuck-open relief valves and other recoverable incidents, were assumed to result in a dry floor. All other cutsets were assumed to provide a wetted drywell floor.

The resulting ΔLERF for internal events was 6.42 x 10^{-6}, as documented in Table 5. Additionally, the analyst used the internal events LERF ratios to estimate the external events contribution to LERF. As documented in Table 3.a, the external events ΔLERF was calculated as 3.2 x 10^{-6}. This resulted in a total ΔLERF for the finding of 9.5 x 10^{-6}, [ ] Ex 5

Sensitive Assumptions: The analyst reviewed the licensee's assumptions and determined that the following differences dominated the dissimilarities between the licensee's and the analyst's assessments:

1. The licensee used a Human Error Probability of 9.2 x 10^{-2} for the probability that operators would fail to realign gland water prior to failure of the Division II pumps. The analyst determined that this assumption was responsible for about 30% of the difference in the final results.

2. The licensee's model uses a Loss of Offsite power frequency of 1.74 x 10^{-8}/hr as opposed the the NUREG/CR-5496 value of 5.32 x 10^{-6}/hr. The analyst determined that this assumption was responsible for the vast majority of the difference in the final results. The analyst noted that the majority of risk was from core damage sequences that were initiated by a loss of offsite power.

3. [ ] Ex 5