FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

PWROG-14001-P, REVISION 1, "PRA MODEL FOR THE GENERATION III WESTINGHOUSE SHUTDOWN SEAL"

PRESSURIZED WATER REACTOR OWNERS GROUP

PROJECT NO. 694

1. INTRODUCTION

By letter dated July 3, 2014 (Reference 1), as supplemented by letters dated September 29, 2014 (Reference 2), March 3, 2015 (Reference 3), May 23, 2016 (Reference 4), September 8, 2016 (Reference 5), January 20, 2017 (Reference 6), and April 17, 2017 (Reference 7), the Pressurized Water Reactor (PWR) Owners Group (PWROG) submitted Topical Report (TR) PWROG-14001-P/NP, Revision 1, "PRA [Probabilistic Risk Assessment] Model for the Westinghouse Shutdown Seal," to the U.S. Nuclear Regulatory Commission (NRC) for review and acceptance for referencing in regulatory actions.

The TR provides the technical basis for the PRA model for the Generation III Shutdown Seal (SDS). The proposed PRA model is based on a failure modes and effects analysis as well as subsequent testing and analyses that are included in TR-FSE-14-1-P, Revision 1, "Use of Westinghouse SHIELD® Passive Shutdown Seal for FLEX Strategies" (Reference 8). The TR used a number of qualification tests and the results from one post-operational test to estimate the Generation III SDS failure probabilities. The NRC staff previously reviewed TR-FSE-14-1-P, Revision 1 and accepted the use of the Generation III SDS for compliance with the Extended Loss of Alternating Current Power evaluations for Order EA-12-049 in the associated endorsement letter (Reference 9). While TR-FSE-14-1-P, Revision 1 contains details of qualification test data for the Generation III SDS, it does not include any failure probabilities that constitute the PRA model for the Generation III SDS. Therefore, in Reference 9, the NRC staff did not approve the PRA model for the Generation III SDS. This safety evaluation (SE) provides NRC staff conclusions relating to the Generation III SDS PRA model together with the applicable “Limitations and Conditions.”

In Section 2 of this SE, the NRC staff provides a summary of the Generation III SDS design and its role in reducing the risk associated with nuclear power plant operation. Section 3 of this SE provides the scope of regulatory applicability for the evaluation provided in this SE. Section 4 of this SE provides the technical criteria that were used to review the PRA model for the Generation III SDS. Section 5 of this SE delineates the “Limitations and Conditions” of the TR.

2. BACKGROUND

For PWRs using Westinghouse Electric Company (Westinghouse) reactor coolant pumps (RCPs), the potential for the loss of all RCP seal cooling resulting in seal failure-induced loss-of-coolant accidents (Seal LOCAs) increases the likelihood of core damage. For Westinghouse RCPs, the loss of all seal cooling is the combined loss of thermal barrier cooling and seal water injection. In many plants, component cooling water (CCW) provides RCP thermal barrier cooling as well as cooling to charging pumps which in-turn provide seal water injection. For such plants, a loss of CCW alone has the potential to cause a Seal LOCA.

For RCPs installed with the high-temperature O-ring seals, the NRC staff has found it acceptable to evaluate potential Seal LOCAs using the Westinghouse Owners Group (WOG) 2000 seal leakage model presented in WCAP-15603, Revision 1-A, "WOG 2000 Reactor Coolant Pump Seal Leakage Model for Westinghouse PWRs" (Reference 10). The WOG 2000 model assumes the onset of seal failure unless seal cooling is recovered within 13 minutes following the loss of cooling and apportions four discrete failure probabilities based on flow rates with 480-gallons per minute (gpm) being the maximum rate from each RCP seal package. This relatively large rate of loss of coolant will result in core uncovery and damage unless injection sources are available. Since loss of all alternating current power that contributed to Seal LOCAs is likely to fail other injection sources, Seal LOCAs have become a dominant risk contributor in several NRC and licensee PRA models.

Westinghouse has developed an RCP SDS that will limit RCS inventory losses to very low levels in the event of a loss of RCP seal cooling. The SDS is a thermally-actuated device that is installed between the No. 1 seal and the No. 1 seal leak-off line to provide a low leakage seal in the event of a loss of all RCP seal cooling. The SDS remains in a stand-by state unless it is required to actuate. When the SDS functions as designed during a loss of seal cooling event, the loss of coolant rate would reduce from 480 gpm to less than 1 gpm. Such leak rate can be considered as a negligible amount. Consequently, operators would have significantly more time to recover from events that led to the loss of seal cooling. For example, during a station blackout event in which AC power was not recovered within 13 minutes, the leak rate is then assumed to be 480 gpm for current RCPs without the SDS. When the SDS functions as designed, significantly more time will be available to recover AC power before a consequential LOCA occurs. Therefore, core damage frequency scenarios associated with Seal LOCAs are significantly reduced.

The first two versions of the SDS (Generation I and Generation II SDS) developed by Westinghouse performed successfully during qualification tests; the PRA and deterministic models for the Generations I and II SDS are described in WCAP-17100-P, Revision 1, "PRA Model for the Westinghouse Shut Down Seal" (Reference 11) and WCAP-17100-P Supplement 1, Revision 0, "PRA Model for the Westinghouse Shut Down Seal - Supplemental Information for All Domestic Reactor Coolant Pump Models" (Reference 12), respectively. However, both Generation I and Generation II SDSs failed during post-operational tests. The operating experience from those designs showed that the Generations I and II SDS are not capable of reliable operation in plant environments. Westinghouse improved the design for the Generation III SDS by incorporating lessons learned from the operating experience of the Generations I and II SDS. As documented in the TR and in the letter dated October 13, 2015 (Reference 13), the Generation III SDS successfully performed its function during the qualification tests and one post-operational test.

3. REGULATORY EVALUATION

The NRC's policy statement on the use of PRA methods in nuclear regulatory activities published in the Federal Register at 60 FR 42622 encourages greater use of PRA to improve safety decision-making and improve regulatory efficiency. The NRC's policy statement also states that the PRA evaluations used in support of regulatory decisions should be as realistic as practicable. Use of the Generation III SDS PRA model must address the associated limitations and conditions delineated in this SE when used to meet regulations, such as: 10 CFR 50.48 and Appendix R to 10 CFR Part 50, Fire Protection; 10 CFR 50.63, Loss of All Alternating Current (AC) Power (Station Blackout); and 10 CFR 50.65, Maintenance Rule. Other programs and processes which are impacted by the SDS are the Reactor Oversight Process (ROP), Mitigating Systems Performance Indicators (MSPI), and Risk-informed Technical Specifications initiatives 4b and 5b submittals.

The Generation III SDS is installed in an existing RCP seal package and therefore constitutes a change, test, or experiment to the facility. It is expected that licensees perform a Title 10 of the *Code of Federal Regulations* (10 CFR) 50.59 "Changes, tests, experiments" assessment. This SE does not approve the installation of the SDS (Item 1 in Section 5, Limitations and Conditions). It also does not apply to licensee's compliance with current or future regulatory requirements which rely on non-probabilistic aspects of the Generation III SDS.

The NRC staff reviewed the Generation III SDS model presented in PWROG-14001-P/NP and finds that the model represents a significant change in modeling seal failure since successful actuation of the SDS with timely trip of RCPs is expected to preclude any substantial leakage. With the issuance of PWROG-14001-P/NP, Revision 1, as modified with additional NRC staff limitations and conditions identified in this SE, an alternative RCP seal package PRA model will be recognized by the NRC for those plants that opt to install the Generation III SDS in existing Westinghouse seal packages for all of their RCPs.

4. TECHNICAL EVALUATION

The NRC staff reviewed the TR using standard methods to assess data and evaluate failure probabilities. These included reviewing and evaluating the:

- apparent and root causes that contributed to post-operational test failures of Generations I and II SDS, respectively;
- modifications made during Generation III SDS design and testing in response to lessons learned from Generations I and II post-operational test failures;
- various aging and degradation mechanism of components in the Generation III SDS, and PWROG's treatment of those mechanisms in testing;
- test methods and results;
- applicability of test data to proposed failure probabilities;
- acceptability of statistical methods used to evaluate data;
- engineering issues that could influence the failure probabilities proposed in the PRA models; and,
- operating experience and performance monitoring.

The following subsections identify and discuss the aforementioned areas of consideration for the NRC staff in evaluating the PRA modeling aspects of the Generation III SDS.

4.1 Consideration of Generations I and II Design Efforts

The operating experience from the Generations I and II SDS designs demonstrated that they could not function reliably in plant environments. Following the post-operational test failures of the Generations I and II SDS designs, Westinghouse initiated efforts to define and eliminate design deficiencies that contributed to the failures. The lessons learned from previous SDS designs led to a number of design improvements implemented by Westinghouse. The most significant modification to the Generation III SDS is the direct-acting actuator, which is a simplification of the actuators used in Generations I and II SDS designs. The simplified design is intended to eliminate many failure modes and improve the overall performance of the Generation III SDS.

The NRC staff reviewed the SDS operating experience to determine whether the Generations I and II SDS failures are applicable to the failure probabilities proposed for use in the PRA model of the Generation III SDS. Specifically, the NRC staff reviewed the apparent cause analysis from the Generation I failure, the root cause analysis from the Generation II failure, attended several meetings to discuss the new design, and visited Westinghouse to gain more insight into how the previous design challenges had been addressed. The NRC staff issued three Requests for Additional Information (RAIs) related to this topic by letter dated February 6, 2015 (Reference 14), which the applicant responded to by letter dated March 3, 2015 (Reference 3).

Section 2.5.4.2 of the TR states that the performance problems experienced by previous generations of the SDS were, in part, due to the fact that design analysis and qualification testing for previous generations of the SDS did not adequately account for the effects of in-service conditions on the SDS performance. In RAI APHB-6, the NRC staff requested that the applicant identify the specific in-service conditions that had not been accounted for in the design analysis and qualification testing of previous generations of the SDS. The NRC staff also requested that the applicant explain why those specific changes made in the design analysis and qualification testing for the Generation III SDS are sufficient to address the performance problems experienced by previous generations of the SDS and to ensure the performance of the Generation III SDS. In its response, the applicant stated that the specific in-service conditions that were not considered in the Generations I and II SDS designs are discussed in Section 4.4 of TR-FSE-14-1, Revision 1, but not in PWROG-14001-P/NP, Revision 1. The NRC staff noted that, in Section 4.4 of TR-FSE-14-1, Revision 1, the applicant discussed the deficiency in design to account for **[** **]** as well as the complex actuation mechanism **[** **]**. The applicant also explained that Section 5 of TR-FSE-14-1, Revision 1, discusses improvements that were made to the Generation III SDS design process. The NRC staff noted that the design process was further enhanced through the inclusion of an independent third party to challenge assumptions and the thoroughness of the design process. Furthermore, the applicant explained that the participation of technical experts from various licensees in the Generation III design process allowed for the incorporation of plant engineering insights and operating experience into the testing and design. The applicant also explained that Section 6 of TR-FSE-14-1, Revision 1, discusses additional analyses that were performed for the Generation III SDS. The NRC staff also noted that the applicant discussed the actuator kinematic analysis which demonstrated that the direct-acting actuator could develop enough force to overcome all expected loads. Based on its review, the NRC staff finds the response acceptable because, as discussed, the applicant has made changes in the design, analysis, and qualification testing to address specific in-service conditions that had not been accounted for in the previous generations of the SDS.

In RAI APHB-7, the NRC staff requested that the applicant identify and describe the enhanced capabilities of the Generation III SDS actuator, and the specific characteristics or changes which resulted in these enhancements. In its response, the applicant stated that Section 7.5.2 of TR-FSE-14-1, Revision 1, discusses the results of tests that measured the maximum force capability of the actuator. The NRC staff noted that one of the failure mechanisms in the previous generation **[ ]**. As discussed in Table 8.1-1 of TR-FSE-14-1, Revision 1, the Generation III SDS actuator can generate a force that is at least **[ ]** the maximum force generated by Generations I and II SDS actuators. The applicant also stated that Section 5.1.3 of TR-FSE-14-1, Revision 1, discusses the improved design of the Generation III SDS actuator compared to those in the Generations I and II SDS designs. The NRC staff noted that the **[ ]** have been eliminated in favor of a more direct approach relying on the wax expansion mechanism. Furthermore, the actuator housing is now **[ ]**. This design is intended to prevent **[ ]**. Therefore, the NRC staff finds the response acceptable because, based on information provided by the applicant, it is reasonable to conclude that the actuator components that caused failure in the Generations I and II SDS designs have been eliminated from the Generation III actuator design.

In RAI APHB-8, the NRC staff requested that the applicant discuss the aspects of design, analysis, and testing that were challenged by experts within Westinghouse, licensee technical experts, and an independent third party consultant, and to identify the changes that were made due to those challenges. In response, the applicant stated that Section 5.3 of TR-FSE-14-1, Revision 1, discusses the enhanced design process utilized in the development and qualification of the Generation III SDS, including examples of challenges to the design provided by independent experts and corresponding changes that were implemented. Specifically, the NRC staff noted that in Section 5.3 the applicant stated that a contributing licensee identified that the proposed **[ ]**. As a result, a design review action item was created and vibration testing parameters and analysis were revised to envelop the appropriate frequencies. Based on its review, the NRC staff finds the response acceptable because the applicant explained and justified that the enhanced design, analysis, and testing processes are sufficient to address performance problems experienced by previous generations of the SDS as well as to ensure the performance of the Generation III SDS.

Based on its review, the NRC staff concludes that it is acceptable not to include the Generations I and II SDS post-operational test failures in the calculation of the Generation III SDS failure probabilities in the PRA models because of the substantial changes in the design, more severe environmental testing, and more robust design process for Generation III SDS. However, the NRC staff also finds that the Generations I and II SDS post-operational test failures highlighted the need for and importance of post-operational testing (see Section 4.10 Performance Monitoring).

4.2 Adequacy of Qualification Testing

PWROG-14001-P, Revision 1, states that the Generation III SDS qualification testing program was documented in Section 7 of TR-FSE-14-1, Revision 1. Section 7.3 of TR-FSE-14-1, Revision 1, describes the **[ ]** Generation III SDS assemblies, which were exposed to a series of environmental conditioning tests to simulate the in-service conditions that may be experienced during the nine-year design life. The qualification tests were performed using simplified SDS assemblies that were subjected **[ ]**. Additional tests were performed to investigate the effects of **[ ]**. These tests were designed to account for possible adverse environmental effects on the performance of the SDS including the potential **[ ]**. **[ ]**. TR-FSE-14-1, Revision 1, indicates that each test successfully demonstrated that the SDS can actuate and limit the leakage to less than 1 gpm for **[ ]**. Endurance testing extended this time to **[ ]** of sealing.

The NRC staff reviewed the qualification test methods and reported results to determine their applicability to the expected in-service failure probabilities. The NRC staff issued four RAIs related to this topic by letter dated February 6, 2015 (Reference 14), which the applicant responded to by letters dated March 3, 2015 (Reference 3), and September 8, 2016 (Reference 5).

Section 4.4.4 of TR-FSE-14-1, Revision 1, states that the root cause analysis performed after the Generation II post-operational test failure **[ ]**. In RAI APHB-1, the NRC staff requested that the applicant provide justification for the acceptance of previously conducted tests which did not include testing-to-failure. In its response, the applicant explained that the previously conducted tests discussed in Sections 7.1.2, 7.1.3, and 7.1.4 of TR-FSE-14-1, Revision 1, were based on conservatively biased parameters that bound the conditions that the SDS will experience when installed in operating plants. As such, the applicant considered those tests appropriate for qualification of the Generation III SDS. The NRC staff noted that, in the testing discussed in the aforementioned sections, while the components were not tested to failure, the components were exposed to test conditions that are more severe than those experienced in operation. Specifically, the NRC staff noted that in Section 7.1.2 of TR-FSE-14-1, Revision 1, the polymer ring radiation tests were based on conservative estimates of the radiation exposure that the Generation III SDS polymer ring will experience over its design life in the RCS.

As discussed in Section 7.1.3.1 of TR-FSE-14-1, Revision 1, the **[ ]** was significantly greater than **[ ]** expected following a loss of all seal cooling. The NRC staff also noted that the **[ ]** imposed on the shaft during the shaft movement testing is discussed in Section 7.1.3.2 of TR-FSE-14-1, Revision 1, and significantly exceeds any anticipated shaft displacement **[ ]**. As discussed in Section 7.1.4 of TR-FSE-14-1, Revision 1, **[ ]** that are more severe than those observed from RCP operating experience. Based on its review, the NRC staff finds the response acceptable because the test conditions in the previously-conducted tests exceed those conditions to be expected in operation.

Section 7.3.2 of TR-FSE-14-1, Revision 1, states that a second group of SDS assemblies underwent a series of "conditioning" tests to simulate the service conditions to which the SDS may be subjected during its nine-year design life. In RAI APHB-2, the NRC staff requested that the applicant explain why the nine-year equivalent radiation exposure described in Section 7.1.2 of TR-FSE-14-1, Revision 1, was excluded in this second qualification testing. In its response, the applicant stated that functional testing has been performed on the Generation III SDS following radiation exposure to **[ ]**, which equates to a nine-year seal life. Therefore radiation exposure was not needed for the second test. The applicant stated that the limiting failure mode of the SDS actuator is the degradation of the **[ ]**. **[ ]** were tested to support a greater-than-95 percent reliability with a 95 percent confidence level. The NRC staff noted that the applicant's explanation indicated that other components of the SDS are not sensitive to the failure mechanisms that are induced by radiation exposure during the Generation III SDS nine-year life. Based on its review and the information provided by the applicant, the NRC staff finds the response acceptable because the limiting components of the Generation III SDS have been tested adequately for radiation exposure.

In RAI APHB-3, the NRC staff requested that the applicant explain why an additional **[ ]** assemblies, which were tested due to a change in **[ ]** some Generation III SDS components, were tested using a limited set of conditioning tests. In its response, the applicant stated that this first set of tests included conditioning of the direct-acting actuator through vacuum and high pressure conditions, vibration and seismic conditions, and corrosion. The second **[ ]** tests was completed on the Generation III SDS assembly as discussed in Section 7.4.2.7 of TR-FSE-14-1, Revision 1, to qualify the parts of the SDS that experienced a change to the **[ ]**, which includes the **[ ]**. The NRC staff noted that the **[ ]** only affects areas outside the direct-acting actuator. Since the design of the direct-acting actuator and its boundary conditions were unaffected by the change in the **[ ]** in the second **[ ]** tests, it is reasonable to conclude that no new failure mechanism is introduced to the actuator. Based on its review and the information provided by the applicant, the NRC staff finds the response acceptable because no new failure mechanism was introduced to the direct-acting actuator by changing the **[ ]**.

Section 2.5.4.2 of PWROG-14001-P/NP, Revision 1, **[ ]** the failure of the Generation I and II SDS that were subjected to them. Section 2.5.4.3 of PWROG-14001-P/NP, Revision 1, also states that the tests demonstrated that Generation III SDS functioned successfully in conditions that are more severe than those that are expected in an operating plant.

In RAI APHB-9, the NRC staff requested that the applicant explain in detail how conditions experienced by the Generation III SDS testing specimens were more severe than operating plant conditions. In response, the applicant stated that Sections 7.3.2.4, 7.3.2.5, 7.3.2.6, and 7.3.3 of TR-FSE-14-1, Revision 1, provide justifications that the testing conditions for the Generation III SDS are more severe than the conditions observed during previous RCP seal operating experience. The NRC staff noted that Section 7.3.2.4 describes the vibration testing which was conducted on a shaker table. The SDS was subjected to a vibration level that was based on [ ] in which the magnitude was increased so that the total energy exceeded the maximum energy observed in the data for all waveforms. The NRC staff noted that Section 7.3.2.5 indicates that the SDS was subjected to seismic tests with [ ] the response spectra for safe shutdown and operating basis earthquakes. The NRC staff also noted that in Section 7.3.2.6, the SDS was subjected to [ ] that was considerably more degraded than that which would be expected in service. In Section 7.3.3, the applicant shows that the cyclic chemistry testing simulated the numerous chemistry changes that occur during power changes and outage operations to which the SDS will be exposed. Based on its review, the NRC staff finds it acceptable because, as discussed, the applicant provided justifications that the test conditions were more severe than those observed in previous operating experience.

Based on its review, the NRC staff finds the qualification test methods and the reported results are applicable to estimating the failure probabilities of the Generation III SDSs.

4.3 Applicability of Reactor Coolant Pump Models and Sub-Models

Section 2.1 of PWROG-14001-P/NP, Revision 1, evaluates the four basic Westinghouse RCP models used in the US: Model 93, Model 93A, Model 93A-1, and Model 100A. All of these pumps have 8 inch (nominal diameter) seals. However, there are minor differences between the RCP models that affect the design and testing of the Generation III SDS. A discussion of relevant design differences between the pump models and how those differences affect the design and testing of the Generation III SDS are included in TR-FSE-14-1-P, Revision 1.

The NRC staff had a concern applying test results from a single RCP model (i.e. Model 93A) to all RCP models. As stated in TR-FSE-14-1-P, Revision 1, the primary difference between RCP Model 93A and the other pump models is [ ]. [ ]. Certain tests were repeated and performed for various RCP models, such as, [ ]. However, these tests were not used to develop the statistical basis of the failure probabilities noted in Section 2.5 of PWROG-14001-P/NP, Revision 1.

When the NRC endorsed TR-FSE-14-P, Revision 1, in Reference 9, there was a limitation that stated "Credit for the SHIELD® seals is only endorsed for Westinghouse RCP Models 93, 93A, and 93A-1. Additional information would be needed to justify use of SDSs in other RCP models." In RAI EPNB-1, the NRC staff requested that the applicant explain why the SDS for RCP Model 100A should be acceptable for use without any additional information besides what is in TR-FSE-14-P, Revision 1. In response, the applicant stated that the design of the RCP seals for the Model 100A RCPs is identical to the design of the RCP seals for the Model 93A-1 RCPs, and all SDS testing for the Model 93A-1 RCP is applicable to the Model 100A RCP. The Model 100A RCPs were not in the scope of TR-FSE-14-P, Revision 1, because there was no analysis to demonstrate that the RCP shaft would [ ]. Since that time, an analysis was performed. The NRC staff noted that per Section 2.3.1 of PWROG-14001-P, Revision 1, a thermal-hydraulic analysis has been performed for the Model 100A RCPs that verifies [

]. Based on its review, the NRC staff finds it acceptable that no additional testing of the Model 100A RCP is required, because, as discussed, the applicant stated that the testing performed for the Model 93A-1 RCP is applicable to the Model 100A RCP, and provided justification that the Model 100A RCP [ ].

In RAI EPNB-2, the NRC staff asked if the [ ] determined by the endurance tests [ ] of all United States nuclear power plants that could use the SDS. In response, the applicant stated that it is stated in Section 3 of PWROG-14001-P, Revision 1, that the applicability of the design temperature and pressure for the Generation III SDS must be confirmed by individual plants under which they take credit in their PRA model for its capability to limit leakage to less than one gallon per minute. By letter dated January 16, 2017, the applicant further explained that if the cold leg temperature exceeds 571ºF, an analysis must be performed to demonstrate that the SDS remains at a temperature below its [ ]. The applicant explained that the purpose of the statement in the PWROG-14001 Revision 1, is to confirm that the SDS should not exceed the design [ ]. The NRC staff finds this response acceptable. Individual plants must ensure that if the cold leg [ ], an analysis must be performed to demonstrate that the SDS remains at a temperature below its [ ] (Item 2 in Section 5, Limitations and Conditions).

4.4 Statistical Analysis of SDS Failure to Actuate

The PWROG used qualification test demand successes to characterize the probability that the Generation III SDS fails to actuate. The applicant developed the failure probability by using a Bayesian update technique, in which prior information about an event is combined with other data. For demand failures, the PWROG used the Jeffrey's non-informative prior distribution that is a beta distribution. The data used to develop the resulting beta distribution was provided from the Westinghouse qualification testing program documented in Section 7.3 of TR-FSE-14-1-P, Revision 1. Specifically, [ ] Generation III SDS assemblies were subject to environmental conditioning and static actuation tests. The Generation III SDS successfully actuated and sealed on the pump shaft [ ]. Based on these test results, and using the Jeffrey's non-informative prior distribution, the mean failure probability for the SDS to actuate is calculated as [ ].

The applicant indicated that NUREG/CR-6823 provides the information necessary to calculate the variance associated with the Jeffrey's non-informative prior distribution that was used to estimate the failure probabilities for the SDS. For the beta distribution, the variance is calculated as [ ].

In RAI APHB-5, the NRC staff requested that the applicant provide a sensitivity analysis of mean failure probability, as well as the variance of the associated beta distribution, that includes the previous SDS actuation failures of the Generation I/II designs. The applicant was also requested to explain why the more conservative mean failure probability should not be utilized given the similarities to the Generation I/II designs. In its response, the applicant calculated the failure probability based on [ ] tests with two failures based on a Jeffrey's non-informative prior distribution. The resulting mean failure probability was [ ]. The applicant also explained that the post-operational test failures associated with the Generations I and II SDS designs should not be included in the calculation of the Generation III SDS failure probability because the retracting actuator was completely redesigned to eliminate the components that caused the previous failures. The applicant also indicated that additional design improvements added margin for the successful performance of the non-actuator components of the Generation III SDS and that Generation III SDS qualification testing exposed the Generation III SDS to environmental conditions that were more severe than the prior operating experience of the RCP seals.

Based on its review, the NRC staff finds it acceptable that the applicant performed the sensitivity analysis considering the two previous operation failures and provided the justification that the more conservative mean failure probability should not be used. The NRC staff also noted that based on the evaluation in Section 4.1 of this SE, it was concluded that Generations I and II SDS post-operational test failures need not be included in the calculation of the Generation III SDS failure probabilities in the PRA models because of the substantial changes in the design, more severe environmental testing, and more robust design process for Generation III SDS.

The NRC staff noted that use of the Jeffrey's non-informative prior distribution, which has a mean failure probability of 0.5, is appropriate when there is a lack of data to support an informed prior. Furthermore, the Bayesian update calculation would yield a posterior distribution with a mean value of $(x+0.5)/(n+1)$ where x is the number of observed failure and n is the number demand. The NRC staff noted that the resulting calculation is equivalent to assuming that the next demand would have a 50 percent chance of failure and a 50 percent chance of success. Therefore, the NRC staff concludes that it is acceptable to perform a Bayesian update using the Jeffrey's non-informative prior for Generation III SDS calculation. Using currently available data of **[ ]** with zero failure, the failure probability is a beta distribution with a mean value of **[ ]**. Subsequently, if industry-wide operational experience (i.e., in-service events) or post-operational testing demonstrates further successes, the licensee may perform additional Bayesian calculations to update the failure probability (Item 3 in Section 5, Limitations and Conditions). The NRC staff's findings regarding the treatment of failure data is documented in Section 4.10 of this SE.

4.5 Statistical Analysis of SDS Failure to Remain Sealed

The PWROG used time-dependent qualification test successes to characterize the probability that the Generation III SDS would fail to remain sealed. The applicant developed the failure probability by using a Bayesian update technique, in which prior information about an event is combined with evidence from the qualification test. The PWROG used the Jeffrey's non-informative prior distribution which is a gamma distribution with a high uncertainty. As discussed in Section 2.2.4 of PWROG-14001-P/NP, Revision 1, the endurance tests of the polymer rings were performed and the total duration of the tests was **[ ]** with no failures. By using the Jeffrey's non-informative prior distribution, an hourly failure rate of **[ ]** is estimated for failure of the SDS to remain sealed on the pump shaft. This hourly failure rate is used for specific mission times in the PRA. For a typical PRA mission time of 24 hours, the mean failure probability for the SDS to remain sealed is calculated as **[ ]**. The applicant indicated that NUREG/CR-6823 provides the information necessary to calculate the variance associated with the Jeffrey's non-informative prior distribution that was used to estimate the failure probabilities for the SDS. For the gamma distribution, the variance is calculated as **[ ]**.

Furthermore, the Bayesian update calculation would yield a posterior distribution with a mean value of (*x*+0.5)/(*T*) where *x* is the number of observed failure and *T* is the amount of time. The NRC staff noted that the resulting calculation is equivalent to assuming that the next demand would have a 50 percent chance of failure and a 50 percent chance of success. Therefore, the NRC staff concludes that the use of the Jeffrey's non-informative prior distribution is appropriate for the failure probability calculation. Using currently available data of **[** **]** with zero failures, the failure rate is a gamma distribution with a mean value of **[** **]**. Subsequently, if industry-wide operational experience (i.e., in-service events) or post-operational testing demonstrates further successes, the licensee may perform a Bayesian update of the failure probability (Item 3 in Section 5, Limitations and Conditions). The NRC staff's findings regarding the treatment of failure data is documented in Section 4.10 of this SE.

4.6 Statistical Analysis of SDS Bypass Failure

The PWROG used time-dependent qualification test successes of RCP O-ring material to characterize the potential failure of the O-ring between the RCP shaft and the shaft sleeve (unique to RCP Model 93A). Failure of this O-ring would represent **[** **]** in the RCP Model 93A, as discussed in Sections 2.2.3 and 2.5.2. In its letter dated April 17, 2017, the applicant stated that the Model 93A RCP has two possible failure modes: **[** **]**. Although failure of the **[** **]**, the applicant indicated that the reliability of the sealing function by the Generation III SDS can be treated the same for all Westinghouse RCP models.

The applicant explained that O-rings can degrade under exposure to high temperatures and therefore the reliability of the O-ring has been investigated in two testing programs demonstrating that the Westinghouse supplied shaft sleeve O-ring is capable of withstanding the temperatures and pressure that are expected during a loss of seal cooling event. **[** **]**. Considering a 24-hour mission time, the failure probability is approximately **[** **]**. Therefore, the applicant stated that failure of the O-ring is an insignificant contributor as it is less than two percent of the total failure probability of the Generation III SDS.

The NRC staff noted that the applicant has not provided sufficient justification and the information needed to support that the **[** **]**. **[**The applicant also has not provided the technical bases and the information needed to substantiate the derivation of the proposed **[** **]**. Furthermore, the NRC staff finds that the **[** **]** mode does not meet the criteria for exclusion delineated by the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-Sa-2009, "Addenda to ASME/ANS RA-

S-2008 Standard for Level 1 / Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," February 2009 (Reference 15) as endorsed by Regulatory Guide 1.200, Rev. 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 16), because the proposed failure probability is more than one percent of the total failure probability of the Generation III SDS.

The NRC staff noted that the O-ring testing was performed and the total exposure time of the tests was approximately **[ ]** with no failures. By using the Jeffrey's non-informative prior distribution, an hourly failure rate of **[ ]** is estimated for the failure of the O-ring by the Bayesian update calculation. For a typical PRA mission time of 24 hours, the mean failure probability for the gamma distribution of the O-ring is calculated as **[ ]**. The variance of this gamma distribution is calculated as **[ ]**. The NRC staff noted the O-ring failure rate calculation is consistent with the calculation of the SDS failure to remain sealed because the total exposure time of the test is used in the Bayesian update. Therefore, the NRC staff finds that licensees with RCP Model 93A installed should incorporate the SDS Bypass failure mode with a gamma distribution with a mean failure rate of **[ ]** in its PRA model consistent with the ASME/ANS PRA Standard (Item 4 in Section 5, Limitations and Conditions). On a plant-specific basis, licensees may propose an alternative treatment for modeling the O-ring failure in risk-informed licensing actions provided adequate justification is also submitted. Subsequently, if industry-wide operational experience (i.e., in-service events) or post-operational testing demonstrates further successes, the licensee may perform a Bayesian update of the failure probability (Item 3 in Section 5, Limitations and Conditions). The NRC staff's findings regarding the treatment of failure data is documented in Section 4.10 of this SE.

4.7 Parametric Uncertainty and the Common Cause Failure Treatment of Multiple Reactor Coolant Pumps

4.7.1 Uncertainty

The applicant indicated that PRA models for the SDS should consider both parameter and modeling uncertainty. Parametric uncertainties refer to the uncertainty in the values for the failure probability distributions used in the PRA model. Model uncertainties refer to the uncertainties associated with the models for specific events or phenomena.

The NRC staff noted that as discussed in Section 2.5.1, the **[ ]**. The NRC staff noted that the 5th percentile and 95th percentile, **[ ]**, are reasonable to address the uncertainty associated with the failure to actuate. In performing uncertainty assessments, the distribution may be propagated through the plant-specific PRA model.

The time-dependent Generation III SDS failure to remain sealed is characterized as a gamma distribution with a mean failure rate of **[ ]**. The NRC staff noted that the 5th percentile and 95th percentile, **[ ]**, are reasonable to address the uncertainty associated with the failure to remain sealed. In performing uncertainty assessments, the distribution may be propagated through the plant-specific PRA model.

The time-dependent Generation III SDS bypass failure is characterized as a gamma distribution with a mean failure rate of **[ ]**. The NRC staff noted that the 5th percentile and 95th percentile, **[ ]**, are reasonable to address the uncertainty associated with the bypass failure. In performing uncertainty assessments, the distribution may be propagated through the plant-specific PRA model.

4.7.2 Common Cause Failure

For modeling simplicity, licensees may choose to calculate a single failure probability to collectively represent all shutdown seals installed in the plant based on the SDS failure probabilities discussed in Section 2.5.1 of PWROG-14001-P, Revision 1. Alternatively, licensees may choose to model each Generation III SDS installed at their plant with a separate basic event. Regardless of the modeling approach taken, the PRA model presented in the PWROG-14001-P, Revision 1 assumes that the failure of one SDS results in a failure of all SDSs. While conservative, this approach allows incorporation with the currently-accepted RCP WOG 2000 model. Thus, the NRC staff agrees with and accepts this simplified approach in the plants that use Westinghouse Generation III SDS.

This simplified approach of treating the consequences of RCP seal failures collectively (i.e., for all RCPs, as opposed to an individual RCP) limits the Generation III SDS model's usability for plants that are in the progress of installing the SDS on each of the RCPs (i.e., some RCPs have the Generation III SDS and some RCPs do not). Based on the acceptance of the simplified approach, the NRC staff finds that the Generation III SDS PRA model is not appropriate for plants that operate with a mixture of types of RCP seal packages and thus, the Generation III PRA model cannot be credited at these plants unless the Generation III SDS are installed in each of the RCPs. This is consistent with one of the limitations of the Generation III SDS PRA model indicated in PWROG-14001-P, Revision 1 (Reference 1) which states that "The SDS must be installed in all RCPs in the plant."

4.8 Inadvertent Actuation of SDS

The applicant indicated that the SDS has been designed and tested to ensure that inadvertent actuation would not occur. There are a number of design and test considerations that support its conclusion that an inadvertent actuation is extremely unlikely. First, the applicant indicated in Section 5.1.3 of TR-FSE-14-1, Revision 1 that the actuator for each individual SDS is examined as part of the acceptance testing prior to installation **[ ]**. Second, the applicant also explained in Section 5.1.3 of TR-FSE-14-1, Revision 1 that the SDS is designed with **[ ]** that prevents movement of the actuator until sufficient force is generated by the actuator to **[ ]**. Third, the applicant stated in Section 7.3.2 of TR-FSE-14-1, Revision 1 that vibration and seismic testing of the SDS were included as part of the qualification testing and such testing verified that inadvertent actuation did not occur under seismic loading representing the safe shutdown earthquake spectra or under vibration loading equivalent to nine years of RCP operation.

As discussed in Appendix D of TR-FSE-14-1-P, Revision 1, **[**

]. The NRC staff noted that inspection would likely be the only method that can identify whether an inadvertent actuation has occurred. As part of RAI-APHB-12, the NRC staff requested that the applicant provide information about inspections that will be performed for the Generation III SDS assemblies when they are in-service, during normal maintenance, or during replacement. The applicant's response and the NRC staff's evaluation are discussed in Section 4.10 of this SE.

By letter dated April 17, 2017, the applicant estimated the mean inadvertent actuation rate to be [ ]. The resulting site-wide inadvertent actuation frequency is dependent on the number of RCPs (i.e., the number of RCS loops) and the time the RCPs will operate.

The applicant stated that the failure rate was developed by a panel comprised of subject matter experts in RCP design, plant operations, and PRA. The panel considered a range of possible scenarios that would lead to an inadvertent actuation, which were modeled through fault tree models. The applicant concluded that the inadvertent actuation failure mode can be screened from the fault tree model since the value is more than two orders of magnitude lower than the proposed total failure frequency of the SDS consistent with SY-A15 of the ASME/ANS PRA Standard (Reference 15).

The NRC staff has endorsed the ASME/ANS PRA Standard (Reference 15) as an acceptable approach to demonstrate the technically acceptability of a PRA in Regulatory Guide 1.200 (Ref. 16). The ASME/ANS PRA Standard (Reference 15), DA-D2 gives guidance regarding the use of expert judgment. The NRC staff noted that in its letter dated April 17, 2017, the applicant described the process and the criteria that its expert panel used in the development of the failure probability of inadvertent actuation. Therefore, while the NRC staff finds it reasonable to conclude that the use of expert judgment is consistent with the provision in the ASME/ANS PRA Standard, the NRC staff questioned portions of the applicant's implementation of the expert elicitation process and did not endorse the licensee's specific result or associated fault tree. However, the NRC staff also noted that the possible causes of inadvertent actuation and associated possible contributors to those causes, as identified by the applicant's panel of experts, provided helpful information for understanding the potential failure mechanisms that may lead to inadvertent actuation.

Furthermore, the NRC staff noted that the ASME/ANS PRA Standard (Reference 15), SY-A15(b) provides guidance on when failure modes can be eliminated. Considering the information provided by the licensee, including the design feature of an anti-inadvertent actuation pin, qualification testing, and acceptance testing discussed above, in combination with the detailed evaluation of potential causes of inadvertent actuation, it is reasonable for the NRC staff to conclude that the inadvertent actuation failure mode is expected to contribute less than one percent of the total failure rate of the SDS. Additionally, the NRC staff noted that the effect of inadvertent actuation is that the RCP seal leakage would not be reduced. The effect to the system operation due to the Generation III SDS's failure to actuate or failure to remain sealed is also that the RCP seal leakage would not be reduced. Therefore, it can be concluded that the effects due to these failure modes are the same.

Based on its review, the NRC staff finds that the inadvertent actuation need not be modeled because the inadvertent actuation failure probability is sufficiently small that it can be excluded from the PRA model in accordance with the provisions in ASME/ANS PRA Standard. However, if additional data is received in the future which indicates that the inadvertent actuation failure mode is more likely than expected, the need to model inadvertent actuation should be reconsidered consistent with the ASME/ANS PRA Standard. To address uncertainty associated with the inadvertent actuation failure mode, the NRC staff has imposed additional requirements for inspection and reporting which are discussed in Section 4.10 Performance Monitoring.

4.9 Trip of RCPs and Use of the PRA Model

The thermal-hydraulic response of the RCPs during a loss of seal cooling was performed to determine the available time for operators to trip the RCPs, the RCPs to coast down, and the SDS to actuate. Operator response times for a typical seal leak-off flow rate of 2.5 gpm and an upper bound seal leak-off flow rate of 5 gpm are presented in Section 2.5.2 of PWROG-14001-P/NP, Revision 1. The applicant explained that the coast down calculations are based on engineering models and were compared with operating experience. The engineering models predict coast down times **[** **]**, depending on the pump model. The SDS actuation time was calculated by the applicant for a thermal-hydraulic model that simulates the time dependent behavior of the No. 1 seal during a loss of seal cooling event and has been benchmarked against seal test data and operating experience.

In Section 2.5.3 of PWROG-14001-P/NP, Revision 1, the applicant discussed an event tree that will add the PRA model for the SDS to the current WOG 2000 model. The NRC staff noted that different RCP leakage rates would result depending on whether the RCPs are tripped: **[** **]**. The plant-specific human error probabilities (HEPs) should be incorporated in the plant-specific PRA evaluation. Furthermore, the NRC staff noted that when modeling operator action to trip an RCP, consideration should be given to the associated support systems and hardware failures (e.g., circuit breaker failures), as well as dependency of human failure events consistent with the provisions delineated in the ASME/ANS PRA Standard. Therefore, the NRC staff has found that HEPs based on plant-specific conditions should be developed for **[** **]**. Furthermore, this information shall be provided in risk-informed licensing application submittals (Item 5 in Section 5, Limitations and Conditions).

Implementing the model in the PRA will require use of several standard PRA evaluations including model logic changes, new HEP development, quantification, and documentation. In accordance with Regulatory Guide 1.200, Rev. 2, (Reference 16), all PRA model changes need to be evaluated before the PRA is used to support any risk-informed application, and any change that is a PRA upgrade should be peer reviewed prior to developing the application. The NRC staff will review the implementation of the Generation III SDS PRA models into the PRA in support of risk-informed applications as applicable and according to its guidelines.

PWROG-14001-P/NP, Revision 1, indicates that ASME/ANS RA-Sb-2013, “Standard for Level 1 / Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," September 2013 (Reference 17), is one of the references. The NRC staff noted that ASME/ANS RA-Sb-2013 is not endorsed by the NRC. In RAI APHB-10, the NRC staff requested the applicant identify any Generation III SDS analyses that utilized provisions in ASME/ANS RA-Sb-2013 that are different from those, as endorsed by the NRC staff, in ASME/ANS RA-Sa-2009.

In its response dated March 3, 2015, the applicant stated that there are no Generation III SDS analyses that utilize provisions in ASME/ANS RA-Sb-2013 that are different from those that are endorsed by the NRC staff in ASME/ANS RA-Sa-2009. Based on its review, the NRC staff finds it acceptable that the PRA model documented in PWROG-14001-P, Revision 1, meets the requirements of ASME/ANS RA-Sa-2009 endorsed by the NRC staff.

4.10 Performance Monitoring

The Generation III SDS has been designed and tested to simulate the in-service conditions that may be experienced during the nine-year design life of the SDS. These tests, as documented in Section 7 of TR-FSE-14-1, Revision 1, were designed to account for possible adverse environmental effects on the performance of the Generation III SDS.

Section 7.3.6 of TR-FSE-14-1, Revision 1, states “the final step in the qualification program will be to conduct a static actuation test on an SDS after one cycle of full-power operation in a RCP.” It is not clear to the NRC staff whether a post-operational test of a Generation III SDS that has been in operation for one cycle of full-power would provide sufficient bases supporting the proposed demand failure probability. In RAI APHB-4, the NRC staff requested that the applicant explain the basis for testing after one cycle of full-power operation and the status of this final step. In its response, the applicant stated that the post operational test is not intended for qualification but considered to be a confirmatory test. The applicant also stated that since the test is intended as a confirmation of the design, analysis, and qualification testing that is documented in TR-FSE-14-1, Revision 1, for the Generation III SDS, it is not necessary to test each RCP model. Rather, a single RCP model was planned to be tested as a representative model. The applicant also stated that a single SDS was planned to be removed from a Model 93A RCP and to be tested by Westinghouse in October 2015.

By letter dated October 13, 2015 (Reference 15), Westinghouse indicated that the performance of the Generation III SDS in the post-operational test was consistent with its designed function. Specifically, when the reactor coolant temperature reached 297 degrees Fahrenheit, the shutdown seal actuated and the leakage rate dropped to below 0.01 gpm. After actuation, the sealed condition was maintained to demonstrate a stable, and low level of leakage was maintained. Westinghouse concluded that the seal performance in this post operational test met all relevant acceptance criteria.

The NRC staff noted that since the qualification tests described in TR-FSE-14-1, Revision 1, exposed the Generation III SDS to environmental conditions more severe than the conditions observed in the previous operating experience for the RCP seals, it is reasonable to conclude that the post-operational test can be considered as confirmatory. The NRC staff also noted that in its response to RAI EPNB-1, the applicant provided the justifications that Westinghouse RCP

Models 93, 93A, 93-A-1, and 100A are similar for the design of the Generation III SDS. The NRC staff's review of RAI EPNB-1 was documented in Section 4.3 of this SE. Therefore, the NRC staff finds it acceptable to test one model of the RCP as a representative. The NRC staff, however, concluded that additional post-operational testing is necessary to assure that the proposed failure probabilities remain acceptable throughout the design life of the Generation III SDS.

The NRC staff noted that the applicant developed qualification testing that was intended to bound a nine-year service life. However, standby components, such as the Generation III SDS, may experience a standby failure probability that would increase over time. The NRC staff also noted that the applicant has not provided performance monitoring strategies that will be utilized to verify the assumptions and analyses supporting the associated failure probability of the Generation III SDS. Without additional operating experience, whether originated from testing, inspections, or in-service events, the NRC staff could not conclude that the PWROG has provided sufficient technical bases supporting the proposed demand failure probability throughout the lifetime of the seal. Periodic actuation tests such as those described in the ASME in-service testing programs are generally applied to stand-by equipment to provide confidence that the demand failure probability remains constant. However, periodic actuation testing is not feasible because such tests are destructive for the Generation III SDS and require the component to be removed from service. Further, operating experience from the actuation of the Generation III SDS while in service is not expected because the probability of a loss of all RCP seal cooling is low.

The NRC staff discussed the topic of gathering operating experience for performance monitoring with the applicant at several public meetings. As a result, the NRC staff issued two RAIs, which were intended to request the additional information needed in the areas of inspection and post-operational test to resolve RAI APHB-4, in a letter dated August 3, 2016 (Reference 18). The applicant responded by letter dated September 8, 2016 (Reference 5).

In RAI APHB-12, the NRC staff requested that the applicant provide information about inspections that will be performed for the Generation III SDS assemblies when they are in-service, during normal maintenance, or during replacement.

In its response dated September 8, 2016, the applicant identified some of the inspection criteria for the Generation III SDS. The inspections include ensuring that the SDS has not inadvertently actuated and verifying **[** **]**. Furthermore, the applicant stated that licensees should inspect **[** **]** portions of the SDS for scores, scratches, raised metal **[** **]**, pits, or chips. These inspections are expected to be performed during seal replacement. The applicant also stated that if a Generation III SDS failure occurs, a root cause assessment will be performed to determine the cause of the failure. The determination of the failure classification and results of the RCA will be discussed with the NRC. The NRC staff finds that with these inspection criteria, licensees are capable of identifying inadvertent actuation and unexpected debris, wear, or corrosion products.

Based on its review, the NRC staff finds the response acceptable because the applicant provided inspection criteria that would ensure inadvertent actuation and unexpected debris be accounted for as operating experience for the Generation III SDS. In addition, given the large amount of risk reduction afforded by the Generation III SDS PRA model, the NRC staff finds that even after successful post-operational tests, licensees shall continue performance monitoring through in-service events that include inspections (Item 6 in Section 5, Limitations and Conditions).

In RAI APHB-11, the NRC staff requested the applicant provide a post-service testing plan for the Generation III SDS assemblies to address the lack of operating experience for the Generation III SDS in actual in-service conditions. In its response dated September 8, 2016, the applicant proposed that up to three additional post-operational tests be planned with a minimum of two additional (three total) post-operational tests being performed. Testing will be conducted on complete shutdown seal assemblies including the No. 1 seal insert, ring set, and thermal actuator. The applicant stated that the testing plan is as follows:

- The first post-operational test conducted in October 2015 after 1.5 years of operation will be credited to the total number of tests;
- The second post-operational test will be performed on a shutdown seal that has experienced approximately 4 years of operation;
- The third post-operational test will be performed on a shutdown seal that has experienced approximately 6 years of operation; and
- The fourth post-operational test will be performed on a shutdown seal that has experienced approximately 8-9 years of operation, but only if such service life is achieved prior to December 31, 2025.

The NRC staff noted that the tests are designed to be performed progressively up to the nine-year design life in which each tested Generation III SDS assembly would experience longer in-service duration. However, the NRC staff noted that the applicant has not identified the timing in which the second and third post-operational tests will be performed. In order to ensure that the post-operational test data is collected in a timely manner, the NRC staff concluded that the second post-operational test shall be performed no later than 2020 and the third post-operational test shall be performed no later than 2023. The NRC staff also recognized Generation III SDS will not generally be put in service for its full nine-year design life because the replacement of the Generation III SDS would depend on plant-specific outage schedules as well as the replacement schedule of the normally installed No. 1 seal insert. Therefore, the NRC staff finds it reasonable that the fourth post-operational test would be performed only if a Generation III SDS with such service life is available (Item 7 in Section 5, Limitations and Conditions). Furthermore, by testing the assemblies, the NRC staff noted that the test results would demonstrate performance of the entire package to actuate and seal.

The applicant also stated that if a Generation III SDS failure occurs during the post-operational test, an RCA will be performed to determine the cause of the failure. The applicant indicated in Reference 4 that the RCA may reach one of three conclusions: 1) design deficiency, 2) test facility failure, or 3) random failure. The applicant stated that if the conclusion of the RCA is design deficiency, the NRC staff will be notified. Upon notification, the NRC staff will assess the results of the RCA and determine whether approval on this TR remains valid. The applicant also indicated that if the conclusion of the RCA is due to a test facility failure, consideration will be given to repeat a post-operational test if it failed. However, as discussed in Section 4.1 of this SE, the NRC staff indicated that the Generations I and II SDS failure highlighted the need for post-operational testing. Therefore, the NRC staff finds that if any of the tests failed due to test facility failure, such post-operational test shall be repeated to obtain a valid result (Item 8 in Section 5, Limitations and Conditions).

The applicant further stated that if the conclusion of the RCA is that the failure is a random failure, the licensee can update the failure probabilities using a Bayesian update. The NRC staff finds that the continued use of the standard Bayesian update, in the event of a Generation III SDS failure, may not be acceptable. Additionally, the applicant did not clearly specify if the NRC staff would be notified of a random failure during a post-operational test.

The NRC staff noted that as part of the response to RAI APHB-11 and APHB-12, the applicant indicated that NRC staff will be notified if the post-operational testing fails due to a design deficiency or if a Generation III SDS failure occurs in-service. Given the large amount of risk reduction for some scenarios afforded by the Generation III SDS PRA model and the very limited data on which the failure probabilities are based, the NRC staff finds that notification of the NRC staff should also be provided within 60 days for any industry-wide operating experience that indicates a failure (e.g., inadvertent actuation, failure to actuate, failure to remain sealed, bypass failure, etc.) of the Generation III SDS. Further, the NRC staff shall be informed of the proposed actions that will be taken in response to the failure and the justification for those proposed actions. In addition, the NRC staff noted that failure of the Generation III SDS may invalidate some of the assumptions in the models as well as the qualification process. In some instances, the failure may indicate that the qualification tests do not appropriately represent the actual in-service conditions. Therefore, a Bayesian update for any of the failure probabilities should not be done prior to providing an acceptable justification to the NRC staff which indicates that the Generation III SDS qualification and PRA modeling bases remain valid. The NRC will review the justifications and determine the appropriate risk value to be applied in the future. (Item 9 in Section 5, Limitations and Conditions).

Based on the applicant's responses to RAIs APHB-11 and APHB-12 as well as the information submitted by letter dated May 23, 2016 (Reference 4), supplemented by any tests and notifications identified in Items 6 to 9 in Section 5, Limitations and Conditions, that are not included in the submitted responses, the NRC staff concludes that the applicant has provided performance monitoring strategies that would verify and support the proposed failure probabilities throughout the lifetime of the seal.

4.11 Documentation Requirements

The RCP seal leakage model, including any related bases and analyses, used by licensees must be documented in the licensee-controlled PRA documentation in accordance with licensees' procedures applicable to PRA-related documents. This documentation must include the licensee's evaluation of and determination that the plant-specific procedures and conditions support the applicability of the PRA model used.

If, on a plant-specific basis, the Generation III SDS model is used in a manner different than described in PWROG-14001-P/NP, Revision 1, as modified by the conditions and limitations imposed by this SE, or if it is used for plant-specific conditions and procedures that are different than typically assumed for Westinghouse plants, then the licensee must provide a description and justification for that model, including its supporting analyses and related bases, in their risk-informed licensing applications that rely on this model (Item 10 in Section 5, Limitations and Conditions).

5. <u>LIMITATIONS AND CONDITIONS</u>

The NRC staff has found that the models and parameters presented in PWROG-14001-P, Revision 1, as supplemented, are acceptable for use in plant-specific PRAs and in support of risk-informed applications. Several issues were raised and dispositioned in RAI responses and in a draft test plan developed by the applicant during the review and not dispositioned in the TR. Therefore, the NRC staff finds the approach and methodology in the TR acceptable for use provided that they are supplemented and used in accordance with the following limitations, conditions, and modifications:

1. This SE does not approve the installation of the Generation III SDS.

2. If the **[ ]**, an analysis must be performed to demonstrate that the SDS remains at a temperature below its maximum qualified limit of **[ ]**.

3. If industry-wide operational experience (i.e., in-service events) or the post-operational testing discussed in Item 6, Section 5, Limitations and Conditions) demonstrates further successes, the licensee may perform additional Bayesian calculations to update the relevant failure probability.

4. Licensees with RCP Model 93A installed shall incorporate the SDS Bypass failure mode with a gamma distribution with a mean failure rate of **[ ]**, consistent with the ASME/ANS PRA Standard.

5. Licensees shall develop plant-specific Human Error Probabilities for: **[ ]** and these shall be factored into the licensee's PRA model-of-record. This information shall be provided in the risk-informed licensing application submittals.

6. Licensees shall monitor the performance of the Generation III SDS by using post-operational test data and industry-wide operating experience, which includes in-service events (e.g., as indicated by loss of RCP seal cooling or inspection during RCP seal replacement). Performance monitoring through in-service events will continue after the post-operational tests discussed in Item 6, Section 5, Limitations and Conditions have been completed.

7. The PWROG shall perform a minimum of two additional post-operational tests in which one of the post-operational tests shall be performed no later than 2020 for a Generation III SDS assembly that has experienced approximately four years of operation and another post-operational test shall be performed no later than 2023 for a Generation III SDS assembly that has experienced approximately six years of operation. Furthermore, the PWROG shall perform one additional post-operational test if a Generation III SDS that has experienced approximately eight to nine years of operation will be available prior to December 31, 2025.

8. If any of the post-operational tests are invalidated due to a test facility failure, that post-operational test shall be repeated to obtain a valid result.

9. The failure probabilities accepted for use in this safety evaluation are conditional and will not automatically apply if failures are observed from industry-wide operating experience or post-operational testing. If any industry-wide operating experience or post-operational test data indicates a failure (e.g., inadvertent actuation, failure to actuate, failure to remain sealed, bypass failure, etc.) of the Generation III SDS, the NRC staff shall be notified within 60 days. The results of the failure, including any proposed actions that will be taken in response to the failure and the justification for those proposed actions, will be discussed with the NRC staff. Justifications shall be provided to the NRC staff which indicate that the Generation III SDS qualification and PRA modeling bases remain valid if the proposed actions include Bayesian updating based on failure data. The NRC will review the justifications and determine the appropriate risk value to be applied in the future.

10. If, on a plant-specific basis, the Generation III SDS PRA model is used in a manner different than described in PWROG-14001-P/NP, Revision 1, as modified by the conditions, limitations, and modifications imposed by this SE, or if it is used for plant-specific conditions and procedures that are different than typically assumed for Westinghouse plants, then the licensee must provide a description and justification for that model, including its supporting analyses and related bases, in their risk-informed licensing applications that rely on this model.

6. CONCLUSIONS

The NRC staff has found that the PRA models for the Generation III SDS in Section 2.5 of PWROG-14001-P, Revision 1, are acceptable for use because they appropriately reflect the failure modes and scenarios of the SDS during normal, abnormal, and accident conditions. With the limitations listed in Section 3 of PWROG-14001-P, Revision 1, and Section 5 of this SE, the PRA models for Generation III SDS may be referenced in plant-specific PRAs.

7. REFERENCES

1. PWROG-14001-P/NP, Revision 1, "PRA Model for the Westinghouse Shutdown Seal," PWR Owners Group, July 3, 2014 (ADAMS Accession No. ML14190A331).

2. PWROG Letter OG-14-347, "Response to NRC Staff Questions in the August 28, 2014 Meeting Regarding the Submittal of PWROG-14001, 'PRA Model for the Generation III Westinghouse Shutdown Seal' (PA-RMSC-0499R4)," September 29, 2014 (ADAMS Accession No. ML14280A117).

3. PWROG Letter OG-15-80, "Response to NRC Staff Request for Additional Information for the Submittal of PWROG-14001-P/NP, Revision 1, 'PRA Model for the Generation III Westinghouse Shutdown Seal' (PA-RMSC-0499R4)," March 3, 2015 (ADAMS Accession No. ML15068A014).

4. PWROG Letter OG-16-188, "PWR Owners Group Transmittal of the Draft Proposed Test Plan for the Generation III SHIELD SDS in Support of PWROG-14001-P Revision 1,

for Information Only (PA-RMSC-1463),” May 23, 2016 (ADAMS Accession No. ML16309A151).

5. PWROG Letter OG-16-278, "Submittal of Request for Additional Information Response Regarding PWROG-14001-P/NP, Revision 1, ‘PRA Model for the Generation III Westinghouse Shutdown Seal’ per PA-RMSC-1463," September 8, 2016 (ADAMS Accession No. ML16309A045).

6. PWROG Letter OG-17-24, “Transmittal of Comments on the Draft Safety Evaluation Regarding PWROG-14001-P/NP, Revision 1, ‘PRA Model for the Generation III Westinghouse Shutdown Seal,’ per PA-RMSC-1463,” January 20, 2017 (ADAM Accession No. ML17073A166).

7. PWROG Letter OG-17-114, “Submittal of Revision to PWROG-14001-P/NP, Revision 1, ‘PRA Model for the Generation III Westinghouse Shutdown Seal,’ per PA-RMSC-1463,” April 17, 2017 (ADAM Accession No. ML17125A082).

8. TR-FSE-14-1-P, Revision 1, "Use of Westinghouse SHIELD® Passive Shutdown Seal for FLEX Strategies," Westinghouse Electric Company, March 2014 (ADAMS Accession No. ML14084A498).

9. NRC Letter, Endorsement Letter for Westinghouse Electric Company Technical Report TR-FSE-14-1-P, Revision 1 and TR-FSE-14-1-NP, Revision 1, "Use of Westinghouse Shield Passive Shutdown Seal for FLEX Strategies," May 28, 2014 (ADAMS Accession No. ML14132A128).

10. WCAP-15603, Revision 1-A, "WOG 2000 Reactor Coolant Pump Seal Leakage Model for Westinghouse PWRs," Westinghouse Electric Company, June 2003 (ADAMS Accession No. ML032040132).

11. WCAP-17100-NP, Revision 1, "PRA Model for the Westinghouse Shut Down Seal," Westinghouse Electric Company, February 2010 (ADAMS Accession No. ML101020570).

12. WCAP-17100-NP, Supplement 1, Revision 0, "PRA Model for the Westinghouse Shut Down Seal - Supplemental Information for All Domestic Reactor Coolant Pump Models," Westinghouse Electric Company, December 2012 (ADAMS Accession No. ML13023A243).

13. Westinghouse Letter LTR-NRC-15-78, “Submittal of ‘Successful Post-Operational Testing of the Generation III SHIELD® Passive Thermal Shutdown Seal’ (Non-Proprietary),” October 13, 2015, (ADAMS Accession No. ML15288A372).

14. NRC Letter, "Request for Additional Information RE: Pressurized Water Reactor Owners Group Topical Report PWROG-14001-P/NP, Revision 1, ‘PRA Model for the Generation III Westinghouse Shutdown Seal,’" February 6, 2015 (ADAMS Accession No. ML14363A506).

15. ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," February 2009.

16. Regulatory Guide 1.200, Rev. 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” March 2009.

17. ASME/ANS RA-Sb-2013, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," September 2013.

18. NRC Letter, "Request for Additional Information Regarding: Pressurized Water Reactor Owners Group Topical Report PWROG-14001-P/NP, Revision 1, ‘PRA Model for the Generation III Westinghouse Shutdown Seal,’" August 3, 2016 (ADAMS Accession No. ML16188A417).

Principal Contributors: Ching Ng, Division of Risk Assessment
Sara Lyons, Division of Risk Assessment
Robert Wolfgang, Division of Engineering

Date: August 23, 2017