PRESSURIZED WATER REACTOR OWNERS GROUP

PWROG PWR Owners Group

PWROG-14001-NP-A
Revision 1

WESTINGHOUSE NON-PROPRIETARY CLASS 3

PRA Model for the Generation III Westinghouse Shutdown Seal

Risk Management Committee

PA-RMSC-1463 Rev. 1

October 2017

Westinghouse

PWROG-14001-NP-A
Revision 1

PRA Model for the Generation III Westinghouse Shutdown Seal

PA-RMSC-1463 Rev. 1

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October 2017

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NRC FINAL SAFETY EVALUATION

This section contains the following documents:

1. NRC cover letter, *"Final Safety Evaluation for Pressurized Water Reactor Owners Group Topical Report PWROG-14001-P, Revision 1, 'PRA Model for the Generation III Westinghouse Shutdown Seal'"* (CAC NO. MF4397)
2. Final Safety Evaluation, *"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"*

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 23, 2017

Mr. W. Anthony Nowinowski, Program Manager
PWR Owners Group, Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive, Suite 380
Cranberry Township, PA 16066

SUBJECT: FINAL SAFETY EVALUATION FOR PRESSURIZED WATER REACTOR OWNERS GROUP TOPICAL REPORT PWROG-14001-P, REVISION 1, "PRA MODEL FOR THE GENERATION III WESTINGHOUSE SHUTDOWN SEAL" (CAC NO. MF4397)

Dear Mr. Nowinowski:

By letter dated July 3, 2014, as supplemented by letters dated September 29, 2014, March 3, 2015, May 23 and September 8, 2016, January 20 and April 17, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML14190A331, ML14280A117, ML15068A014, ML16309A151, ML16309A045, ML17073A166, and ML17125A082, respectively), 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 NRC staff has found that TR PWROG-14001-P/NP, Revision 1, is acceptable for referencing in licensing applications for nuclear power plants to the extent specified and under the limitations delineated in the TR and in the enclosed final safety evaluation (SE). The final SE defines the basis for our acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in licensing action requests, our review will ensure that the material presented applies to the specific plant involved. Requests for licensing actions that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

W. Nowinowski - 2 -

In accordance with the guidance provided on the NRC website, we request that the PWROG publish approved proprietary and non-proprietary versions of TR PWROG-14001-P/NP, Revision 1, within 3 months of receipt of this letter. The approved versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The approved versions shall include an "-A" (designating approved) following the TR identification symbol.

As an alternative to including the request for additional information (RAIs) and RAI responses behind the title page, if changes to the TR were provided to the NRC staff to support the resolution of RAI responses, and if the NRC staff reviewed and approved those changes as described in the RAI responses, there are two ways that the accepted version can capture the RAIs:

1. The RAIs and RAI responses can be included as an Appendix to the accepted version.
2. The RAIs and RAI responses can be captured in the form of a table (inserted after the final SE) which summarizes the changes as shown in the approved version of the TR. The table should reference the specific RAIs and RAI responses which resulted in any changes, as shown in the accepted version of the TR.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, PWROG will be expected to revise the TR appropriately or justify its continued applicability for subsequent referencing. Licensees referencing this TR would be expected to justify its continued applicability or evaluate their plant using the revised TR.

Sincerely,

Dennis C. Morey, Chief
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure:
Final Safety Evaluation (Non-Proprietary)

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. <u>REGULATORY EVALUATION</u>

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. <u>TECHNICAL EVALUATION</u>

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.

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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).

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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 571°F, 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 [ ]. [ ]. 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. LIMITATIONS AND CONDITIONS

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.

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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).

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Date: August 23, 2017

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- Alvin Robertson for expert consultation regarding impact of RCP seal leakage on plant operations.
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Pacific Gas & Electric	Diablo Canyon 1 & 2 (W)		X
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South Carolina Electric & Gas	V.C. Summer (W)		X
So. Texas Project Nuclear Operating Co.	South Texas Project 1 & 2 (W)		X
Southern Nuclear Operating Co.	Farley 1 & 2 (W)	X	
	Vogtle 1 & 2 (W)	X	
Tennessee Valley Authority	Sequoyah 1 & 2 (W)	X	
	Watts Bar 1 & 2 (W)	X	
Wolf Creek Nuclear Operating Co.	Wolf Creek (W)	X	
Xcel Energy	Prairie Island 1 & 2 (W)		X

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	Tihange 1 & 3 (W)		X
Electricite de France	58 Units		X
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Eletronuclear-Eletrobras	Angra 1 (W)		X
Emirates Nuclear Energy Corporation	Barakah 1 & 2		X
Eskom	Koeberg 1 & 2 (W)		X
Hokkaido	Tomari 1, 2 & 3 (MHI)		X
Japan Atomic Power Company	Tsuruga 2 (MHI)		X
Kansai Electric Co., LTD	Mihama 3 (W)		X
	Ohi 1, 2, 3 & 4 (W & MHI)		X
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EXECUTIVE SUMMARY

Westinghouse has developed a reactor coolant pump (RCP) SHIELD® Passive Thermal Shutdown Seal (SDS) (Generation III) that restricts reactor coolant system (RCS) inventory losses to very low leak rates during a plant event that results in the loss of all RCP seal cooling. The SDS is a thermally actuated, passive device that is installed between the Number 1 seal and the Number 1 seal leak-off line to provide a low leakage seal in the event of a loss of all RCP seal cooling. The installation of the SDS permits plants to respond to a wide range of events with only a turbine-driven or diesel-driven auxiliary feedwater pump available. These events include station blackout, fires that disrupt power supplies, loss of component cooling water system, and loss of service water system. Because there are negligible RCS inventory losses through the RCP seals with the SDS installed, RCS makeup is no longer necessary to achieve a stable state.

The consensus Probabilistic Risk Assessment (PRA) model for performance of Westinghouse RCP seal packages with high-temperature O-rings is the WOG2000 model that has been reviewed and approved by the United States (US) Nuclear Regulatory Commission (NRC) for use in the PRA. Risk-informed applications derived from PRA studies using the WOG2000 model have become an integral aspect of plant operation and include daily plant configuration risk management activities, Mitigating System Performance Index (MSPI) reporting, license amendment requests for changes to the plant Technical Specifications (TS), and regulatory interactions associated with the Significance Determination Process (SDP).

Plants installing the Westinghouse Generation III SDS will likely change their PRA to credit the reduced seal leakage capability of the SDS. The new SDS model can be appended to the current WOG2000 model or other future seal leakage models in the PRA. In order to most efficiently utilize both NRC and utility resources in reviewing PRA models used in risk-informed applications, the PRA model for the SDS was submitted to, and approved for use by, the NRC.

The performance of the SDS has been verified by testing and analysis. The testing has included individual component tests as well as tests of the entire seal assembly under conditions that exceed those observed in RCP seal operating experience.

The PRA model developed for the Generation III SDS is based on the extensive testing and analysis performed on the Generation III SDS design. Integration of the Generation III SDS into the plant PRA model may result in a significant decrease in core damage frequency (CDF). The actual reduction will vary from plant to plant depending on the contribution of the RCP seals to the core damage risk metric.

In addition to the reduction in the CDF risk metric, the installation of the SDS may also permit licensees to modify their coping strategies for station blackout and fires as required by 10 CFR 50.63 and 10 CFR 50.48, respectively, by assuming an RCP seal leak rate of 1 gpm per RCP for the duration of the event. A deterministic model of SDS performance is included for information, but no NRC review or approval of the deterministic model was performed.

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RECORD OF REVISIONS

Revision	Date	Revision Description
0	June 2014	Original Issue
1	July 2014	Added clarification of the relation between the WOG2000 model, future seal models, and the SDS PRA model. Clarified limitations and applicability of the WOG2000 model. Provided additional details of the leak rate calculations in Section 2.3.2.
"-A"	October 2017	Approved version of Topical Report.

RESOLUTION OF FINAL SAFETY EVALUATION TO APPROVED TOPICAL REPORT

Update	Revision Description
1	Addition of: • Final Safety Evaluation • RAI responses (Appendix B)
2	Update of the following text from Paragraph 3 of the Executive Summary, and Paragraph 2 of the Summary and Conclusions based on the conclusions of the final safety evaluation: *"In order to most efficiently utilize both NRC and utility resources in reviewing PRA models used in risk-informed applications, the PRA model for the SDS is being submitted to the NRC for review and approval."* was updated to: *"In order to most efficiently utilize both NRC and utility resources in reviewing PRA models used in risk-informed applications, the PRA model for the SDS was submitted to, and approved for use by, the NRC."*
3	Update of the following text from Paragraph 6 of the Executive Summary: *"A deterministic model of the SDS performance is included for information, but no NRC review or approval of the deterministic model is requested."* was updated to: *"A deterministic model of the SDS performance is included for information, but no NRC review or approval of the deterministic model was performed."*

RESOLUTION OF FINAL SAFETY EVALUATION TO APPROVED TOPICAL REPORT

Update	Revision Description
4	Update of the following text from Paragraph 1 of the Purpose based on the conclusions of the final safety evaluation: *"The NRC is requested to review and approve the PRA model for the Westinghouse SDS described in this report. A deterministic model for the SDS performance is included for information, but NRC review and approval of this model is not requested."* was updated to: *"The NRC has reviewed and approved the PRA model for the Westinghouse SDS described in this report. A deterministic model for the SDS performance is included for information, but NRC review and approval of this model was not performed."*
5	Removal of the following text from Paragraph 1 of Section 2.1: *"However, no NRC review and approval is being requested for the RCP models not used in the U.S."*
6	Addition of second paragraph of Section 2.2 based on OG-14-347 Enclosure 1.
7	Addition of third paragraph of Section 2.2.3 based on OG-14-347 Enclosure 1.
8	Replacement of second paragraph of Section 2.2.3. Second paragraph of Section 2.2.3 was replaced with text from Enclosure 1 of OG-17-114, and modified to match the conclusions of the final safety evaluation.
9	[ ]a,c
10	Reorganization of Section 2.5.1 which includes addition of Sections 2.5.1.1, 2.5.1.2, and 2.5.1.3. Section was reorganized to better display calculation of failure probabilities and variances, and to include calculation of Model 93A Bypass O-ring failure rate calculation based on the conclusions of the final safety evaluation.

RESOLUTION OF FINAL SAFETY EVALUATION TO APPROVED TOPICAL REPORT

Update	Revision Description
11	Addition of the following paragraph to the end of Section 2.5.1 based on the conclusions of the final safety evaluation: *"Additionally, if industry-wide operational experience or post-operational testing demonstrates further successes, it is appropriate to perform Bayesian calculations to update the failure probability/rate derived in this section. Section A.2 includes guidance for performing the Bayesian calculations."*
12	Created Section 2.5.2 for PRA Fault Tree Model discussion that was previously discussed in Section 2.5.1.
13	Addition of the following text to Section 2.5.2 based on the conclusions of the final safety evaluation: *"Additionally, the fault tree file shows all three basic events that would be applicable for the Model 93A RCP. However, the basic event for the failure of the Model 93A shaft sleeve O-ring (ORING-FAILS) may be omitted for plants that do not use the Model 93A RCP."*
14	Modification to Figure 2.5-1 to include the Model 93A RCP Bypass O-ring failure mode based on the conclusions of the final safety evaluation.
15	[ ]^{a,c}
16	Addition of the following text to the end of Paragraph 1 of Section 2.5.5.1 based on the revisions proposed in OG-16-278 Enclosure 1: *"Additionally, the treatment of parametric uncertainty using the NUREG-1855 (Reference 11) process is further described in Section A.1."*
17	[ ]^{a,c}
18	Addition of the following text to the end of Section 2.5.6.4: *"Additional discussion on the treatment of multiple RCPs in the PRA model is provided in Section A.4."*

RESOLUTION OF FINAL SAFETY EVALUATION TO APPROVED TOPICAL REPORT

Update	Revision Description
19	Addition of the following text to the end of Section 2.5.6.5 based on the revisions proposed in OG-17-114 Enclosure 1. *"When evaluating the operator action to trip the RCPs, analysts should consider the relevance of hardware/equipment failures and whether or not the operator action to trip the RCPs is dominant as compared to the hardware/equipment failures. Analysts should refer to the screening criteria provided in the SY supporting requirement of the ASME/ANS PRA Standard (Reference 10) when determining whether or not to include hardware/equipment failures as part of the operator action to trip the RCPs. In addition, any dependency analysis for the operator action to trip the RCPs, and subsequent recovery actions to trip the RCPs by alternate means, should be done in accordance with the HR supporting requirements of the ASME/ANS PRA Standard (Reference 10)."*
20	[ ][a,c]
21	Addition of final safety evaluation limitations to Section 3.
22	Addition of references based on addition of text from RAI responses.
23	Addition of Appendix A based on OG-17-114 Enclosure 1 and OG-16-278 Enclosure 1.
24	Update of Figure 2.3-1 to a higher resolution diagram.

ACRONYMS

ac	Alternating Current
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
B&W	Babcock and Wilcox
CDF	Core Damage Frequency
CE	Combustion Engineering
CFR	Code of Federal Regulations
ECCS	Emergency Core Cooling System
ELAP	Extended Loss of ac Power
gpm	Gallons per Minute
HR	Human Reliability Analysis
LLC	Limited Liability Corporation
MSPI	Mitigating Systems Performance Index
NRC	Nuclear Regulatory Commission
PA	Project Authorization
PRA	Probabilistic Risk Assessment
psia	Pounds per Square Inch Absolute
psig	Pounds per Square Inch Gage
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owners Group
QA	Quality Assurance
RAI	Request for Additional Information
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RMSC	Risk Management Subcommittee
rpm	Revolutions per Minute
SDP	Significance Determination Process
SDS	Shutdown Seal
SOKC	State of Knowledge Correlation
SY	Systems Analysis
TS	Technical Specifications
US	United States
W	Westinghouse

1 INTRODUCTION AND PURPOSE

1.1 INTRODUCTION

The Westinghouse reactor coolant pump (RCP) circulates reactor coolant fluid between the reactor core and the steam generators during normal operation to transfer heat generated in the core to the steam generators. During off-normal and accident conditions, the RCPs may continue to run to provide forced circulation between the core and steam generators for decay heat removal. The Westinghouse RCP seal package consists of a series of three seals which provide a barrier to limit leakage of the reactor coolant system (RCS) fluid. The design includes redundant seal cooling functions, which maintain the controlled leakage to within allowable values. Cooling to the RCP seals is provided by seal injection with thermal barrier cooling available as a backup. RCP cooling by either means alone is sufficient to provide adequate seal cooling so that prompt mitigating actions are not required.

As long as cooling is provided to the RCP seals by either means, the seals function normally and there are no significant RCS inventory losses. If seal injection flow is provided, the flow through the seal package (between 1 and 5 gallons per minute (gpm) per RCP) comes from seal injection (nominally 8 gpm per RCP) with the excess seal injection traveling down the RCP shaft and into the RCS. If seal injection is lost, but thermal barrier cooling is available, the coolant leakage from the RCS into the seal package is cooled so that the leak rate is controlled at the pre-event seal leak rate (between 1 and 5 gpm per RCP), which does not present a significant challenge to core cooling. If all seal cooling is lost, the leakage of reactor coolant through the RCP seals increases as described in WCAP-10541 (Reference 1). Loss of RCS inventory, by means of the RCP seal leakage after a loss of all seal cooling, can result in loss of core cooling if mitigating actions are not completed in a timely manner.

For plants with Westinghouse RCPs, compliance with several of the United States (US) Nuclear Regulatory Commission's (NRC) regulatory requirements is impacted by the performance of the RCP seals when all RCP seal cooling is lost. Most notable of these are the station blackout coping strategies for compliance with 10 CFR 50.63 and the fire coping strategies for compliance with 10 CFR 50.48.

The probabilistic risk assessments (PRAs) for plants with Westinghouse RCPs also show that the performance of the RCP seals under loss of all seal cooling conditions is a contributor to the plant core damage frequency (CDF), both for internal hazard initiating events including fire initiating events and external hazard initiating events such as seismic, external flooding and high winds. A PRA model for Westinghouse RCP seal behavior under loss of all seal cooling conditions, commonly referred to as the WOG2000 model, is provided in WCAP-15603 (Reference 2).

To provide a means for utilities to address potentially risk-significant events, Westinghouse has developed an RCP Shutdown Seal (SDS), the Generation III SHIELD® Passive Thermal Shutdown Seal, which will limit RCS inventory losses to very low leak rates in the event of a loss of seal cooling. The SDS is a thermally actuated, passive device that is integral to the Number 1 seal insert and is located between the Number 1 seal and the Number 1 seal leak-off line to limit RCS inventory losses in the event of a loss of all RCP seal cooling. The PRA model for the Generation III SDS can be appended to the current WOG2000 model in plant-specific PRAs.

The construction of the model also allows it to be appended to any future PRA models of RCP seal leakage that may be developed to replace the WOG2000 model.

A detailed description of the Generation III SDS can be found in TR-FSE-14-1 (Reference 3). TR-FSE-14-1 provides a basis for use of the Generation III SDS in FLEX strategies, and provides detailed information regarding the design of the SDS and associated testing results. The NRC reviewed this technical report and concluded that the Generation III SDS is acceptable for use as a FLEX strategy (Reference 9). Rather than repeating the information discussed in TR-FSE-14-1 in this topical report, data that is relevant to the PRA model is summarized herein, and references are made to TR-FSE-14-1 as a source of more detailed information regarding the design and testing of the Generation III SDS.

The PRA and deterministic models in this report apply to the Generation III SDS design. The PRA and deterministic models described in WCAP-17100-P-A (Reference 7) and WCAP-17100-P Supplement 1 (Reference 8), for the previous SDS designs, are no longer valid since operating experience from those designs has shown that they are not capable of reliable operation in plant environments. Unless otherwise noted, all references to the SDS in this report are referring to the Generation III SDS design.

1.2 PURPOSE

This report documents the new models for RCP seal behavior under loss of all seal cooling conditions. The NRC has reviewed and approved the PRA model for the Westinghouse SDS described in this report. A deterministic model for SDS performance is included for information, but NRC review and approval of this model was not performed.

2 TECHNICAL EVALUATION

2.1 PUMP MODELS AND SUB-MODELS

There are 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. Internationally, there are additional Westinghouse RCP models. These include Model 100D, which also has an 8 inch (nominal diameter) seal, and Models 63, 70, and 93D, which have 7 inch (nominal diameter) seals. The PRA and deterministic models documented in this report apply to the RCP models used in the US.

In addition to the four basic US RCP model designations, Westinghouse RCPs can contain an additional identifier to designate differences in some design aspects. An "S" designation refers to the presence of a spool piece between the reactor pump and the motor that facilitates RCP seal inspection and replacement. In pumps with a spool piece, the RCP motor does not have to be lifted from the pump in order to remove the RCP seal package. The seal package is identical for pumps with or without the spool piece. A "CS" designation refers to a cartridge type Number 2 and Number 3 RCP seal. The Number 2 and Number 3 Cartridge Seals are slightly different in design to allow for easier replacement as an assembly, as opposed to individual components. The remainder of the seal package is identical for pumps with or without the cartridge seal design. In general, the SDS will be identical for all pumps within a basic model designation (i.e., Model 93, Model 93A, Model 93A-1 and Model 100A). Minor differences between the designs of the SDS for different RCP models are described below.

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2.2 SUMMARY OF APPLICABLE DATA

TR-FSE-14-1 (Reference 3) provides a detailed description of the Generation III SDS design, its test program, and testing results. Testing data that has particular relevance to the SDS PRA and deterministic models is summarized in this section. [

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2.2.1 Actuation and Initial Sealing

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2.2.2 Rotating Shaft Sealing

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2.2.3 O-Ring Sealing in the RCP Model 93A

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2.2.4 Endurance Sealing

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2.2.5 Actuation on a Rotating Shaft

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2.2.6 Natural Circulation Test Data

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2.3 ANALYSIS RESULTS

2.3.1 RCP Coast Down and SDS Actuation

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2.3.2 Leakage Following Actuation on a Rotating Shaft

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Figure 2.3-1: SDS Internal Components

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Table 2.3-1 Seal Leak Rates with Damaged SDS

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2.4 CURRENT PRA MODEL

The current consensus PRA model for Westinghouse RCP seal performance following a loss of all seal cooling is the WOG2000 model described in WCAP-15603 (Reference 2). This PRA model considers various combinations of seal failure scenarios involving the Number 1 and Number 2 RCP seals.

Because the SDS terminates flow before the Number 1 seal leak-off line and before the Number 2 RCP seal, successful actuation of the SDS isolates all of the RCS inventory loss pathways for the normally installed RCP seal package. Therefore, the RCP seal PRA model for plants with an SDS installed would be altered as described in Section 2.5.

2.5 PRA MODEL WITH THE SDS

The PRA model for the RCP seals with the SDS must take into account both the probability that the SDS fails to actuate and seal, and the probability that the operators fail to trip the RCPs in a timely manner following a loss of all seal cooling.

2.5.1 SDS Failure to Actuate and Seal

Failure probabilities for components modeled in the PRA are commonly developed using Bayesian analysis techniques, which mathematically combine prior information about an event with other data. The prior distribution represents all that is known or assumed about the failure probability prior to collecting any data. As reported in NUREG/CR-6823 (Reference 4), the information summarized by the prior distribution can be objective, subjective, or both. Operational data and data from a previous but comparable experiment could be used as objective data. Subjective information could involve personal experience and opinions, expert judgment, or design information. One class of prior distributions that is widely used is termed noninformative priors, also referred to as priors of ignorance, or reference priors. These names refer to the situation where very little *a priori* information about a parameter is available in comparison to the information expected to be provided by the data sample.

The Jeffreys noninformative prior distribution is commonly used to estimate the failure probability for rare events.

For demand failures, the Jeffreys noninformative mean for beta distributions is given in NUREG/CR-6928 (Reference 5) by:

$$P = \frac{n + 0.5}{D + 1}$$

Where:

P = Jeffreys noninformative mean

n = the number of failures

D = the number of tests

For time-dependent failures, the Jeffreys noninformative mean for gamma distributions is given in NUREG/CR-6928 (Reference 5) by:

$$\lambda = \frac{n + 0.5}{T}$$

Where:

λ = Jeffreys noninformative mean

n = the number of failures

T = the number of hours

NUREG/CR-6823 (Reference 4) provides the information necessary to calculate the variances associated with the Jeffreys noninformative prior distributions used to estimate the failure probabilities for the SDS.

Additionally, if industry-wide operational experience or post-operational testing demonstrates further successes, it is appropriate to perform Bayesian calculations to update the failure probability/rate derived in this section. Section A.2 includes guidance for performing the Bayesian calculations.

2.5.1.1 Failure to Actuate

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2.5.1.2 Failure to Remain Sealed

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2.5.1.3 Failure of Model 93A RCP O-ring to Remain Sealed

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2.5.2 Fault Tree Model

The fault tree structure for assessing the failure probability of the SDS in the PRA model is shown in Figure 2.5-1. A calculated probability for failure of the SDS to seal as well as the probability of the bypass O-ring to seal, is included based on a typical PRA mission time of 24 hours. [

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Figure 2.5-1: Fault Tree for Failure of the SDS (24 Hour Mission Time)

2.5.3 Operator Action to Trip the RCPs

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Table 2.5-1 Operator Action Times to Trip RCPs Following Loss of Seal Cooling		

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Table 2.5-2 [ ]a,c

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2.5.4 PRA Model Implementation

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Figure 2.5-2: PRA Model for Loss of Seal Cooling with SDS and the WOG2000 Model

2.5.5 Discussion of Uncertainties

The ASME/ANS PRA Standard (References 6 and 10) requires that PRA model uncertainties be identified and characterized so that they may be investigated in the course of risk-informed decision making. Guidance for the treatment of uncertainty in the PRA model and applications is included in NUREG-1855 (Reference 11). Uncertainties that are relevant to the PRA model for the SDS are described below.

2.5.5.1 Parametric Uncertainties

PRA models for the SDS should consider both aleatory (parameter) and modeling uncertainty. Parametric uncertainties refer to the uncertainty in the values for the failure probabilities used in the PRA model. As discussed in Section 2.5.1 [

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2.5.5.2 Performance Uncertainties

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2.5.5.3 Material Uncertainties

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2.5.5.4 Other Modeling Uncertainties

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2.5.6 Additional PRA Modeling Considerations

2.5.6.1 Level 2 PRA

Core damage accident sequences for various initiating events in existing Level 1 PRAs may be binned as low pressure core damage sequences for consideration in the Level 2 PRA as a result of RCP seal leakage. Consideration of the thermal-hydraulic response of the plant for core damage accident sequences with successful actuation of the SDS may result in these accident sequences being re-binned as high pressure core damage sequences. An assessment of the binning of the Level 1 core damage sequences is required when installation of the SDS is credited in the plant-specific PRA.

2.5.6.2 RCS Cooldown and Depressurization

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2.5.6.3 Seal Survivability Assessments

The mechanical seal packages installed in Westinghouse RCPs can be procured from multiple vendors. Therefore, licensees with Westinghouse RCPs that have procured mechanical seals and high temperature O-rings from other vendors should verify that the assessments of Westinghouse O-ring and mechanical seal survivability provided in this report are applicable to the behavior of their seal packages following a loss of seal cooling prior to using those assessments as a basis for RCP seal behavior in the PRA.

2.5.6.4 Common Cause Failures

TR-FSE-14-1 (Reference 3) documents the extensive testing and analysis performed to ensure functionality of the SDS and discusses quality assurance processes that govern its manufacture. [

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Additional discussion on the treatment of multiple RCPs in the PRA model is provided in Section A.4.

2.5.6.5 Operator Action for Tripping RCPs

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When evaluating the operator action to trip the RCPs, analysts should consider the relevance of hardware/equipment failures and whether or not the operator action to trip the RCPs is dominant as compared to the hardware/equipment failures. Analysts should refer to the screening criteria provided in the SY supporting requirement of the ASME/ANS PRA Standard (Reference 10) when determining whether or not to include hardware/equipment failures as part of the operator action to trip the RCPs. In addition, any dependency analysis for the operator action to trip the RCPs, and subsequent recovery actions to trip the RCPs by alternate means, should be done in accordance with the HR supporting requirement of the ASME/ANS PRA Standard (Reference 10).

2.5.6.6 Inadvertent Actuation

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The resulting 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 other failure modes discussed in Section 2.5 are also on a per-pump basis. Consistent with SY-A15 of the ASME/ANS Standard (Reference 10), the calculated value for inadvertent actuation can be screened from the fault tree model since the value is greater than two orders of magnitude lower than the proposed failure frequency of the SDS. Therefore, the inadvertent actuation failure mode should not be included in the fault tree model for the SDS.

2.6 DETERMINISTIC MODEL

The current model used in deterministic analyses for RCP seal performance following a loss of all RCP seal cooling is based on WCAP-10541 (Reference 1). This model is used in determining the acceptability of coping strategies for 10 CFR 50.48(a) and 50.48(b) (fire protection) and 10 CFR 50.63 (station blackout).

For plants installing the SDS, the fire protection and station blackout analyses can be changed to model a 1 gpm per RCP leak rate, consistent with the design specification and testing for the SDS, provided that the scenario under consideration is within the SDS design specification and qualification testing. [

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The Westinghouse SDS is not designed as an RCS pressure boundary component. The SDS has no RCS pressure boundary function in its normal non-actuated state. [

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3 LIMITATIONS

The limitations of the Generation III SDS PRA model described in this report are:

- The SDS must be installed in all RCPs in the plant.
- The plant's RCP seals are manufactured by Westinghouse and constructed of silicon nitride to ensure the following:
  - Westinghouse survivability assessments and testing of the O-rings in the seal package are valid.
  - If the plant's RCP seals are not manufactured by Westinghouse, a basis for the use of the SDS model in the PRA should be developed by the licensee.
  - Licensees must confirm that the flow rates in the WOG2000 model are appropriate for use in their plant-specific PRA model, or develop the appropriate plant-specific flow rates.
- The RCP oil lift pumps should not be in operation after RCP startup (the analysis of RCP coast down discussed in Section 2.3.1 assumes that the oil lift pumps are secured).
- The assumed environmental conditions for the SDS for any accident or transient, for which the SDS is credited as a reliable means of controlling the RCS inventory loss through the RCP seals to less than 1 gpm per RCP, are within the limiting analysis, qualification, and testing parameters for the SDS documented in TR-FSE-14-1 (Reference 3).
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- If the [ ]^{a,c}, an analysis must be performed to demonstrate that the SDS remains at a temperature below its maximum qualified limit of [ ]^{a,c}. *Note that this limitation was added based on Final Safety Evaluation.*
- If, on a plant specific basis, the Generation III SDS PRA model is used in a manner different than described in this approved topical report, 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.
- The Final Safety Evaluation does not approve the installation of the SDS. 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. *Note that this limitation was added based on Final Safety Evaluation.*
- If industry-wide operational experience demonstrates further successes, additional Bayesian calculations to update the relevant failure probability may be performed. *Note that this limitation was added based on Final Safety Evaluation.*
- Utilities with RCP model 93A installed shall incorporate the SDS Bypass failure mode as described in Section 2.5.1.3. *Note that this limitation was added based on Final Safety Evaluation.*

- Utilities shall develop plant-specific Human Error Probabilities for: [ ]a,c and these shall be factored into the PRA model of record. This information shall be provided in the risk-informed licensing application submittals. *Note that this limitation was added based on Final Safety Evaluation.*
- The failure probabilities are conditional and will not automatically apply if failures are observed from industry-wide operating experience or post-operational testing. *Note that this limitation was added based on Final Safety Evaluation.*
- It is the licensees responsibility to review all conditions and limitations stated in the final safety evaluation.

The limitations of the Generation III SDS deterministic model described in this report are:

- The SDS must be installed in all RCPs in the plant.
- The RCP oil lift pumps should not be in operation after RCP startup (the analysis of RCP coast down discussed in Section 2.3.1 assumes that the oil lift pumps are secured).
- The assumed environmental conditions for the SDS for any accident or transient, for which the SDS is credited as a reliable means of controlling the RCS inventory loss through the RCP seals to less than 1 gpm per RCP, are within the limiting analysis, qualification, and testing parameters for the SDS documented in TR-FSE-14-1 (Reference 3).
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4 SUMMARY AND CONCLUSIONS

Westinghouse has developed an RCP SDS that restricts reactor coolant system inventory losses to very low leak rates when a plant event results in the loss of all RCP seal cooling. The SDS is a thermally actuated, passive device that is installed between the Number 1 seal and the Number 1 seal leak-off line to provide a leak-tight seal in the event of a loss of all RCP seal cooling. The installation of the SDS will permit plants to respond to a wide range of events with only a turbine-driven or diesel-driven auxiliary feedwater pump available. The possible events include station blackout and ELAP, fires that disrupt power supplies, loss of component cooling water system, and loss of service water system.

The consensus PRA model for Westinghouse RCP seal performance is the WOG2000 model, which has been approved by the US NRC. The new SDS model can be appended to the current WOG2000 model or other future seal leakage models in the PRA. Plants installing the Westinghouse SDS will likely change their PRA to credit the reduced seal leakage capability of the SDS. In order to most efficiency utilize NRC and utility resources in reviewing PRA models used for risk-informed applications, the PRA model for the SDS was submitted to, and approved for use by, the NRC.

The performance of the SDS has been verified by a large amount of testing and analysis to confirm that it meets very stringent design goals. The testing has included individual component tests as well as tests of the entire seal assembly under conditions that exceed those observed in RCP seal operating experience.

The PRA model developed for the SDS is based on the extensive testing and analysis performed on the SDS design. Integration of the SDS within the plant PRA model may result in a significant decrease in core damage frequency (CDF). The actual reduction will vary from plant to plant depending on the contribution of the RCP seals to the core damage risk metric.

In addition to the reduction in PRA risk metrics, the installation of the SDS may also permit licensees to modify their coping strategies for station blackout and fires as required by 10 CFR 50.63 and 10 CFR 50.48, respectively, by assuming an RCP seal leak rate of 1 gpm per RCP for the duration of the event.

Both the deterministic and the PRA models for the SDS are applicable for a time period of up to 168 hours at nominal hot standby conditions and are also applicable to any scenario that includes operator actions for RCS cooldown and depressurization.

5 REFERENCES

1. WCAP-10541, Revision 2, "Reactor Coolant Pump Seal Performance Following a Loss of All AC Power," Westinghouse Electric Company, November 1986.
2. WCAP-15603, Revision 1-A, "WOG 2000 Reactor Coolant Pump Seal Leakage Model for Westinghouse PWRs," Westinghouse Electric Company, June 2003.
3. TR-FSE-14-1-P, Revision 1, "Use of Westinghouse SHIELD® Passive Shutdown Seal for FLEX Strategies," March 2014.
4. NUREG/CR-6823, "Handbook of Parameter Estimation for Probabilistic Risk Assessment," U.S. Nuclear Regulatory Commission, September 2003.
5. NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," Nuclear Regulatory Commission, February 2007.
6. ASME/ANS RA-Sb-2013, "Standard for Level 1 / Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," September 2013.
7. WCAP-17100-P-A, Revision 1, "PRA Model for the Westinghouse Shut Down Seal," August 2011.
8. WCAP-17100-P Supplement 1, Revision 0, "PRA Model for the Westinghouse Shut Down Seal – Supplemental Information for All Domestic Reactor Coolant Pump Models," December 2012.
9. NRC Acceptance Letter to TR-FSE-14-1, ADAMS Accession No. ML14132A128, May 2014.
10. 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.
11. NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," March 2017.
12. NUREG/CR-6771, "Debris Induced Loss of ECCS Recirculation on PWR Core Damage Frequency," August 2002.

APPENDIX A : ADDITIONAL INFORMATION

A.1 TREATMENT OF PARAMETRIC UNCERTAINTY IN RISK-INFORMED APPLICATIONS USING NUREG-1855

NUREG-1855 (Reference 11) provides guidance on how to treat uncertainties associated with PRAs used by a licensee or applicant to support a risk-informed application to the NRC. The focus of NUREG-1855 (Reference 11) is the treatment of uncertainties related to the lack of knowledge about, or confidence in, the system or model (i.e., epistemic uncertainty). Specifically, parameter uncertainty relates to the uncertainty in the computation of the input parameter values used to quantify the frequencies and probabilities of the events in the PRA logic model. As part of the risk-informed decision making process, the NRC requires that the numerical results of the PRA, including their associated quantitative uncertainty, are compared with the appropriate acceptance guidelines.

NUREG-1855 (Reference 11) provides a seven stage regulatory process for treating uncertainty within the context of a PRA and risk-informed application. The overview of the process is illustrated below. Stage A determines the applicability of NUREG-1855 (Reference 11) to the process. For the purposes of this discussion, risk-informed applications are those where the PRA is utilized to demonstrate acceptable risk metrics. Risk significant events that credit the SDS for limiting risk impact should utilize NUREG-1855 (Reference 11) and that licensees (in stages B through F) should utilize the NUREG-1855 process to demonstrate whether the risk-informed application results in an acceptable risk to the public.

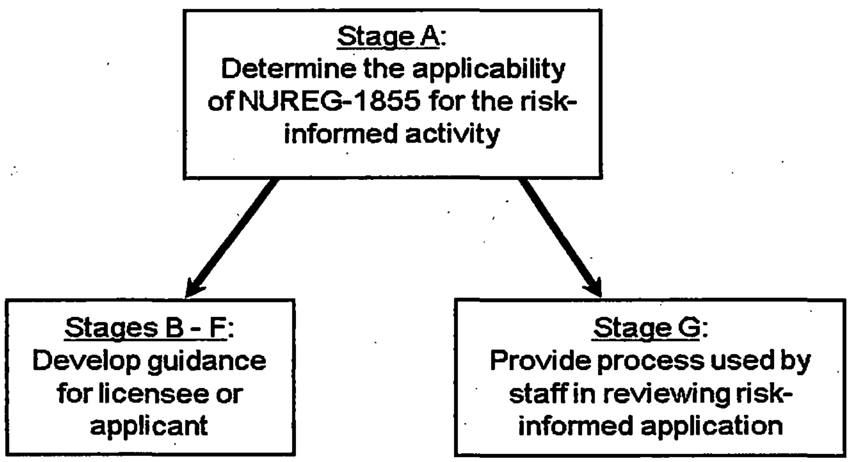

Figure A-1: Risk-Informed Review Process

The uncertainties inherent in specifying the SDS model parameters results in the treatment of uncertainty being an important aspect of the associated risk informed applications. The specific activities recommended by the licensee or utility to support risk informed applications include:

- Stage B: Understanding the risk-informed application and determine the scope of the PRA needed to support the application. Based on the application under consideration, this task will identify those models and parameters which require regulatory review.
- Stage C: Evaluating the completeness uncertainties and determining if bounding analyses are acceptable for the missing scope items.
- Stage D: Evaluating the parameter uncertainties.

- Stage E: Evaluating model uncertainties to determine their impact on the applicable acceptance guidelines.
- Stage F: Developing strategies to address key uncertainties in the application.

While NUREG-1855 (Reference 11) focuses on epistemic uncertainties, the framework of the document supports the risk-informed regulatory guidance that requires that the risk impact of all relevant uncertainties (aleatory and epistemic) be treated in a risk-informed application. In treating epistemic uncertainties, the model should consider completeness uncertainty, parametric uncertainty, and model uncertainty.

For many parameters (e.g., component failure probabilities or failure rates), the uncertainty is characterized as probability distributions that represent the degree of belief in the value of the parameter. In order to assess the influence of the parameter uncertainty of the PRA inputs on the PRA results it is important to: (1) characterize the uncertainty in the parameters used in the various PRA inputs, (2) propagate that uncertainty through the analysis that calculates the risk metrics, while also accounting for state-of-knowledge-correlation (SOKC) and its potential effects on the results, and (3) compare the results with the acceptance guidelines. For the Generation III SDS SHIELD® PRA model, the above three items are achieved by completing the following:

1. <u>Characterizing the uncertainty in the quantification of the parameters in the basic events and other inputs of the PRA mode</u>
   For the failure rates derived in Section 2.5.1, appropriate bounds of uncertainty are given for each failure rate. The mean values of the failure rates as well as a classified distribution and associated uncertainty parameters are provided for each derived failure rate.
2. <u>Quantifying the risk metrics, accounting for parameter uncertainty and the state-of-knowledge correlation</u>
   A typical way to account for both the parameter uncertainty and to properly treat the state-of-knowledge correlation (SOKC) is to use type codes in the PRA model. However, due to the uniqueness of RCP seal modeling, there is no need for the grouping of basic events that use the same failure rate data that is derived in Section 2.5.1, since the loss of seal cooling event is considered to affect all RCPs and if an RCP is predicted to fail, all RCPs are assumed to fail. Quantifying the model with the appropriate characterization of basic events will provide the quantitative PRA model results.
3. <u>Compare the application risk results with the application acceptance guidelines</u>
   This step involves comparing the risk results with the application acceptance guidelines. This comparison reveals if and how the acceptance guidelines are met and, if needed for the decision, how the uncertainty of the risk metric estimates, arising from the propagation of the uncertainty in the parameter values of the PRA inputs, impacts the comparison.
   In addition to the treatment of parameter uncertainty in the SDS model, the application should also consider uncertainties in the PRA that may be introduced by the modeling of the RCP seal. NUREG-1855 (Reference 11) explicitly includes RCP seal models as being subject to model uncertainty. Typical model uncertainties in RCP seal development may include assumed RCP seal failure modes, RCP seal leakage rates and their impact on assigning LOCA end states and the associated risk significant operator actions. These features are typically addressed via sensitivity studies. In the NRC review of

model uncertainty, the NRC expects the licensee to demonstrate an adequate understanding of (1) the part of the PRA that is affected, (2) the modeling approach or assumption used, (3) the impact on the PRA, and (4) whether there is an associated conservative bias. Model uncertainty is typically addressed via sensitivity studies.

To assess the significance of the Generation III SDS SHIELD® on the risk metric results for a specific application, a method for assessing the parametric dependence on the risk metric results is discussed below:

1. Quantify the model and review the risk metric results without taking credit for the SDS.
   If the risk results meet the acceptance criteria and no credit for the SDS is needed, then there is no need to further evaluate the affect the Generation III SDS SHIELD® has on the PRA model.
2. Quantify the model and review the risk metric results taking credit for the SDS at the 95th percentile failure rate values.
3. Quantify the model and review the risk metric results taking credit for the SDS at the 5th percentile failure rate values.

A.2 INCLUSION OF OPERATING EXPERIENCE TO UPDATE FAILURE PROBABILITIES

When operating experience is gained for the Generation III SDS SHIELD®, it is appropriate to Bayesian update the failure probabilities derived in Section 2.5.1. The following section provides the appropriate equations that would be used to perform the Bayesian update. The section is broken out into two parts, a description of the equations used when the prior distribution is a gamma distribution (hourly failure rate), and when the prior distribution is a beta distribution (demand). The prior Alpha and Beta parameters are derived in Section 2.5.1.

Update Process for Failure Rate Parameter:

Prior Distribution = Gamma (Hourly Failure Rate)

$$\alpha_{posterior} = \alpha_{prior} + n_{failures} \tag{1}$$

$$\beta_{posterior} = \beta_{prior} + n_{exposures} \tag{2}$$

$$\sigma^2_{posterior} = \frac{\alpha_{posterior}}{(\beta_{posterior})^2} \tag{3}$$

$$\mu_{posterior} = \frac{\alpha_{posterior}}{\beta_{posterior}} \tag{4}$$

Where α, and β are the uncertainty parameters of the distribution, σ^2 is the variance of the distribution, and μ is the mean of the distribution.

Update Process for Demand Failure Parameter:

Prior Distribution – Beta (Demands)

$$\alpha_{posterior} = \alpha_{prior} + n_{failures} \tag{5}$$

$$\beta_{posterior} = \beta_{prior} + n_{exposures} - n_{failures} \tag{6}$$

$$\sigma^2_{posterior} = \frac{\alpha_{posterior} * \beta_{posterior}}{(\alpha_{posterior} + \beta_{posterior} + 1)(\alpha_{posterior} + \beta_{posterior})^2} \tag{7}$$

$$\mu_{posterior} = \frac{\alpha_{posterior}}{(\alpha_{posterior} + \beta_{posterior})} \tag{8}$$

Where α, and β are the uncertainty parameters of the distribution, σ^2 is the variance of the distribution, and μ is the mean of the distribution.

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A.3 FORMAL PROCESS FOR IDENTIFYING A FAILURE OF THE GENERATION III SDS SHIELD®

The Regulation that is applicable to a defect which could create a substantial safety hazard is 10 CFR 21 "Reporting of Defects and Noncompliance." 10 CF 21 requires the supplier of basic components to facilities or activities licensed by the NRC, to report to the NRC any deviation or failure to comply with the Atomic Energy Act of 1954, as amended, or any applicable rule, regulation, order or license of the NRC that could create a substantial safety hazard, were it to remain uncorrected.

Westinghouse has a procedure that implements the requirements of 10 CFR 21.

If a failure of the Generation III SDS SHIELD® to actuate during testing were to occur, the Westinghouse Part 21 process would be implemented. The failure to actuate would be evaluated and after the cause of the failure was determined, Westinghouse would issue a letter to the affected customers that discusses the issue. Additionally, Westinghouse would notify the NRC of the issue, if required by 10 CFR 21.

A.4 TREATMENT OF MULTIPLE REACTOR COOLANT PUMPS (RCPS) IN PRA MODELS

Section 2.5 contains a PRA model for the Generation III SHIELD® SDS in which a loss of all seal cooling scenario is postulated and analyzed. Similar to the NRC approved WOG2000 PRA model (WCAP-15603, Reference 2), it is assumed that all RCPs with the same Generation III SDS SHIELD® assembly installed would respond the same to a loss of all seal cooling scenario (e.g., fail or succeed). This assumption provides a conservative failure probability for the failure of all Generation III SDS SHIELD® assemblies failing during a loss of all seal cooling scenario. Mathematically this can be shown as follows:

During a loss of all seal cooling event, all Generation III SHIELD® SDS assemblies will fail. Therefore, the failure probability can be calculated as follows (for a two-loop plant).

For Dependent Events:

$$P(A \cap B) = P(A) * P(B|A)$$

Assuming the Generation III SHIELD® SDS failures are dependent, and that if one fails all fail, results in the following:

$$P(B|A) = 1$$

$$P(A \cap B) = P(A)$$

A more realistic model for failure of all RCPs would consider P(B|A) as a common cause parameter. A conservative estimate for a common cause parameter would be less (if not much less) than one. Given this, it can be shown that the realistic model for the failure of all RCPs is lower than the failure parameters used to model the failure of all RCPs.

Furthermore, it is not unreasonable to expect some degree of randomness in the failures of the installed Generation III SDS SHIELD® assemblies. Thus, not all Generation III SDS SHIELD® assemblies would be expected with 100% certainty to fail. This assumption provides a bounding leak rate and simplifies the analysis. This treatment represents an epistemic uncertainty. Therefore epistemic uncertainty assessment associated with the plant PRA model should consider sensitivity cases to estimate the impact of an SDS failure with a lower bound leak rate (i.e., one RCP failure, no dependence).

A.5 INADVERTENT ACTUATION EVALUATION

This section contains a detailed evaluation and discussion on the probability of an inadvertent actuation of the SDS. Inadvertent actuation of the SDS while the RCP is in operation will damage the SDS and prevent the SDS from performing its function, should it be needed following a loss of seal cooling. As a consequence of the significance of the event, and the low likelihood of detection given premature actuation, considerable effort has gone into including design features to prevent inadvertent actuation. In order to quantify the likelihood of inadvertent actuation of the SDS an expert panel comprised of senior Westinghouse subject matter experts in design, field services, plant operations, and probabilistic risk assessment was convened to perform a failure modes and effects analysis of the SDS and to develop an estimate of the probability of an inadvertent actuation using proven PRA methodologies. The following discussion describes the events that could lead to an inadvertent actuation of the SDS, the expected consequences, and the assigned probabilities.

Fault Tree Model for Inadvertent Actuation

To systematically assess the frequency of inadvertent actuation of the SDS, a fault tree model was developed. Subsequently, numerical probabilities were assigned to the fault tree basic events and the fault tree was quantified to yield a frequency of inadvertent actuation.

The fault tree model structure and numerical assignments of associated basic event probability were based on the engineering judgment of a group of experienced Westinghouse engineers who considered their own background as well as input from other subject matter experts. The full fault tree structure is shown in Figure A-2 and includes the following possible causes of inadvertent actuation:

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Inadvertent actuation is defined as any failure of the SDS or SDS components that would result in the actuation of the SDS at seal leak-off temperature below 235°F. Seal leak-off temperatures above 235°F typically result from a loss of seal cooling event, for which the SDS is designed to actuate. Therefore, by definition, if the seal leak-off temperature exceeds this value and the SDS actuates, actuation of the SDS is not inadvertent. Therefore, by definition, inadvertent actuation is only considered possible when the SDS actuates without a prior loss of all seal cooling accident to the affected RCP. If a loss of all seal cooling accident occurs, and the SDS actuates prior to reaching its actuation temperature, this occurrence is not considered inadvertent. This type of failure is considered a random failure of the SDS.

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Figure A-2: Inadvertent Actuation Fault Tree

Fault Tree Quantification

The assignment of numerical values to the basic events in the fault tree was performed by the expert panel consensus process. The assignment of numerical values considered the SDS design and quality control, as well as, extensive testing and analysis that have been performed to prevent inadvertent actuation of the SDS as described in TR-FSE-14-1 (Reference 3). The guidelines for the assignment of numerical values to the basic events in the fault tree and the basis for the assignment is described below.

A set of criteria was developed for translating engineering judgment into quantifiable values. The criteria are similar to the guidelines for probability assignments used in NUREG/CR-6771 (Reference 12). The Table A-1 criteria were provided to expert panel members for use in assigning probabilities to physical events such as component failures of relevance to the inadvertent actuation failure mode.

Table A-1: Guidelines for Assigning Conditional Probabilities to Physical Events

Value	Description
1.0E-01	The indicated outcome is UNLIKELY but it cannot be supported by analysis or testing. It is a credible outcome, considering uncertainties, such that it could occur during the plant lifetime.
5.0E-02	The indicated outcome is HIGHLY UNLIKELY. Analyses or testing have not been done directly or cannot rule it out completely. However, arguments in favor of this outcome are not supported by available testing or analysis.
1.0E-02	The indicated outcome is VERY UNLIKELY. Available testing or analyses show that the outcome did not occur. Consideration of these uncertainties might lead to this outcome but no analytical or experimental support can be found.
1.0E-03	The indicated outcome is EXTREMELY UNLIKELY. Available testing or analyses show that the outcome did not occur and consideration of the uncertainties cannot reasonably be found to lead to this outcome even when no analytical or experimental support can be found.
1.0E-04	The indicated outcome is ALMOST IMPOSSIBLE. No analysis or testing is available to support this result even considering relevant uncertainties. It has credibility only if a number of unsupported (but not demonstrably incorrect) assumptions are made. All available analysis and experiments support alternate outcomes.

Some of the fault tree entries require an estimation of the probability that an out-of-specification item would not be detected in the Westinghouse or supplier quality program. It was assumed that all components will be procured and shipped under the Westinghouse Quality Assurance Program and that all suppliers would have an approved quality program that is acceptable under the Westinghouse Quality Assurance Program. The Table A-2 criteria were provided to expert panel members for use in assigning probabilities to non-compliances under the vendor and Westinghouse Quality programs that would contribute to events that could impact SDS inadvertent actuation.

Table A-2: Guidelines for Assigning Conditional Probabilities to Quality Program Events	
Value	**Description**
1.0E-01	The indicated outcome is UNLIKELY. Quality program is based on observations by Westinghouse that are not completely independent from supplier quality checks or a supplier on the Westinghouse Quality Supplier List with whom issues have occurred in the past.
1.0E-02	The indicated outcome is VERY UNLIKELY. Quality program is based on testing or observations by Westinghouse that are not completely independent from supplier quality checks or a supplier on the Westinghouse Quality Supplier List with whom little (or no) experience has accumulated.
1.0E-03	The indicated outcome is EXTREMELY UNLIKELY. Quality program is based on independent testing by Westinghouse or a supplier on the Westinghouse Quality Supplier List with whom experience has accumulated.
1.0E-04	The indicated outcome is ALMOST IMPOSSIBLE. Quality program is based on no human interactions or decisions.

The criteria above provide a basis for qualifying the fault tree based on engineering judgment by an expert panel to provide an estimate of the yearly frequency of inadvertent actuation of the SDS during normal operation on a per pump-year basis.

This section provides the numerical value assigned to each of the basic events identified in the fault tree along with the basis for that assignment as developed by the expert panel.

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]a,c

Conclusions:

The resulting frequency of inadvertent actuation, based on the consensus judgment process, is [ ]a,c. The resulting 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 other failure modes discussed in Section 2.5 are also on a per-pump basis. Consistent with SY-A15 of the ASME/ANS Standard, the calculated value for inadvertent actuation can be screened from the fault tree model since the value is greater than two orders of magnitude lower than the proposed failure frequency of the SDS. Therefore, the inadvertent actuation failure mode should not be included in the fault tree model for the SDS.

APPENDIX B PWROG CORRESPONDENCE AND RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION

This section contains the following documents:

RAI Responses:

1. 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)"*, dated September 29, 2014, (ML14280A117)
2. Enclosure (1) to OG-14-347, *"Status of Open Items from TR-FSE-14-1-P, Revision 1."*
   Note that this enclosure was considered entirely proprietary and therefore a non-proprietary enclosure is not included in this document.
3. Enclosure (2) to OG-14-347, *"PWROG Slides from August 28, 2014 Meeting."*
   Note that this enclosure was considered entirely proprietary and therefore a non-proprietary enclosure is not included in this document.
4. 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)"*, dated March 3, 2015, (ML15068A014)
5. Enclosure to OG-15-80, *"Response to NRC Request for Additional Information on PWROG-14001-P, Revision 1, 'PRA Model for the Generation III Westinghouse Shutdown Seal'"*
   Note that this enclosure was considered entirely proprietary and therefore a non-proprietary enclosure is not included in this document.
6. Affidavit CAW-15-4112, Enclosure to OG-15-80, *"Application for Withholding Proprietary Information from Public Disclosure, 'Response to NRC request for Additional Information on PWROG-14001-P, Revision 1, 'PRA Model for the Generation III Westinghouse Shutdown Seal' (Proprietary)"*
7. 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)"*, dated May 23, 2016, (ML16309A151)
8. LTR-RES-16-82, Enclosure to OG-16-188, *"Generation III SHIELD® Shutdown Seal post-Operational Test Plan,"* dated May 12, 2016.
   Note that this enclosure was considered entirely proprietary and therefore a non-proprietary enclosure is not included in this document.
9. Affidavit CAW-16-4424, Enclosure to OG-16-188, *"Application for Withholding Proprietary Information from Public Disclosure, 'Generation III SHIELD® Shutdown Seal Post-Operational Test Plan' (Proprietary)"*
10. 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"*, dated September 8, 2016, (ML16309A045)

11. LTR-RAM-16-76, Enclosure to OG-16-278, *"Responses to RAI-APHB-11, RAI-APHB-12 and Revisions to Topical Report PWROG-14001-P/PWROG-14001-NP, Revision 1, 'PRA Model for the Generation III Westinghouse Shutdown Seal,' that Address Additional NRC Information Requests*", dated September 8, 2016.

Note that this enclosure was considered entirely proprietary and therefore a non-proprietary enclosure is not included in this document.

12. Affidavit CAW-16-4457, Enclosure to OG-16-278, *"Application for Withholding Proprietary Information from Public Disclosure, 'Responses to RAI-APHB-11, RAI-APHB-12 and Revisions to Topical Report PWROG-14001-P/PWROG-14001-NP, Revision 1, 'PRA Model for the Generation III Westinghouse Shutdown Seal,' that Address Additional NRC Information Request'" (Proprietary)*
13. Letter OG-17-114, *"Submittal of Revisions to PWROG-14001-P/NP, Revision 1, 'PRA Model for the Generation III Westinghouse Shutdown Seal,' per PA-RMSC-1463"*, dated April 17, 2017, (ML17125A082)
14. LTR-RAM-17-17, Enclosure to OG-17-114, *"Revisions to Topical Report PWROG-14001-P/PWROG-14001-NP, Revision 1, 'PRA Model for the Generation III Westinghouse Shutdown Seal.'"*, dated April 12, 2017.
15. Affidavit CAW-17-4569, Enclosure to OG-17-114, *"Revisions to Topical Report PWROG-14001-P/PWROG-14001-NP, Revision 1, 'PRA Model for the Generation III Westinghouse Shutdown Seal'" (Proprietary)*

PWROG
PWR Owners Group

Program Management Office
1000 Westinghouse Drive
Cranberry Township, PA 16066

PWROG-14001-P/NP Revision 1
Project Number 694

September 29, 2014

OG-14-347

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Subject: PWR Owners Group
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)

During the August 28, 2014 meeting between the Pressurized Water Reactor Owner's Group (PWROG) and the Nuclear Regulatory Commission (NRC) to discuss the Probabilistic Risk Assessment (PRA) model for the Generation III Westinghouse SHIELD® Passive Shutdown Seal (SDS), which is documented in Topical Report (TR) PWROG-14001, the NRC staff requested the following:

1. Provide clarification relative to whether a post operational test of the Generation III (Gen III) SDS is considered part of the qualification test program that is discussed in TR-FSE-14-1-P, Revision 1.

2. Provide the status of the two open items discussed in TR-FSE-14-1-P, Revision 1, and provide a determination of whether those open items are also applicable to TR PWROG-14001.

3. Provide a reference linking TR-FSE-14-1-P, Revision 1 to TR PWROG-14001.

With respect to item (1), a post operational test of the Gen III SDS is planned for October 2015 following the removal of the first installed Gen III SDS in an RCP in Beaver Valley Unit 2. As stated regarding the post operational test on page D-15 of TR-FSE-14-1-P, Revision 1, "this test is not required for qualification, but indicates intent to verify successful product rollout." Thus, the post operational test is not a requirement for qualification of the Gen III SDS. This is consistent with PWROG statements made

1 **SHIELD** is a trademark or registered trademark of Westinghouse Electric Company LLC, its affiliates and/or its subsidiaries in the United States of America and may be registered in other countries throughout the world. All rights reserved. Unauthorized use is strictly prohibited. Other names may be trademarks of their respective owners.

during the August 28, 2014 meeting that the post operational test is considered to be confirmatory, and should not be required for the NRC to issue a Final Safety Evaluation for TR PWROG-14001 or to allow credit for the Gen III SDS in risk-informed applications.

With respect to item (2), due to the proprietary nature of the open items and associated technical details, the closure of the open items that are contained in TR-FSE-14-1-P, Revision 1 and the applicability of those items to the PRA model of TR PWROG-14001 is addressed in Enclosure 1. Following receipt of the Final Safety Evaluation for TR PWROG-14001, Revision 1, the NRC approved TR will be issued as TR PWROG-14001, Revision 2, and will incorporate the changes identified in Enclosure 1.

With respect to item (3), TR-FSE-14-1-P, Revision 1 (ML14084A496) is referenced in TR PWROG-14001, Revision 1. TR-FSE-14-1-P, Revision 1 documents the design and qualification test information that forms the basis for the PRA model provided in TR PWROG-14001, Revision 1.

Sincerely yours,

Norman J. Stringfellow

Jack Stringfellow, Chairman
PWR Owners Group

TZ:NJS:rfn

cc: PWROG Risk Management Committee
PWROG Licensing Committee
PWROG PMO
J. Gresham, Westinghouse
B. Howard, Westinghouse
B. Kunkel, Westinghouse
S. Davis, Westinghouse
R. Lutz, Westinghouse
P. Hijeck, Westinghouse
S. Baier, Westinghouse
M. Skocik, Westinghouse
A. Mahan, Westinghouse
C. Zozula, Westinghouse
R. Schneider, Westinghouse
M. LaPresti, Westinghouse
D. Weaver, Westinghouse

Enclosure 1: Status of Open Items from TR-FSE-14-1-P, Revision 1
Enclosure 2: PWROG Slides from August 28, 2014 Meeting

Pages B-5 through B-26 are not included in this document since they are proprietary.

Program Management Office
1000 Westinghouse Drive
Cranberry Township, PA 16066

PWROG-14001-P/NP Revision 1
Project Number 694

March 3, 2015

OG-15-80

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Subject: PWR Owners Group
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)

References:

1. "PWR Owners Group Submittal of PWROG-14001-P/NP, Revision 1, "PRA Model for the Generation III Westinghouse Shutdown Seal," OG-14-211, dated July 3, 2014.

2. NRC Letter, Jonathan Rowley of NRR to Anthony Nowinowski of the PWR Owners Group Program Management Office, "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" (TAC NO. ME4397), February 6, 2015

By letter dated February 6, 2015, the NRC staff transmitted requests for additional information (RAIs) to the Pressurized Water Reactor Owner's Group (PWROG) regarding the Probabilistic Risk Assessment (PRA) model for the Generation III Westinghouse SHIELD® Passive Shutdown Seal (SDS), which is documented in Topical Report (TR) PWROG-14001. The enclosure to this letter documents the PWROG responses to the NRC RAIs.

This submittal contains information proprietary to Westinghouse Electric Company LLC; it is supported by affidavits signed by Westinghouse, owner of the information. The affidavits set forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b) (4) of Section 2.390 of the Commission's regulations.

Accordingly, it is respectfully requested that this information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the information listed above or supporting Westinghouse affidavit should reference CAW-15-4112 and should be addressed to Mr. J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Correspondence related to this transmittal should be addressed to:

Mr. W. Anthony Nowinowski, Program Manager
PWR Owners Group, Program Management Office
Westinghouse Electric Company
Mail Stop ECE 5-16
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355

Sincerely yours,

Norman J. Stringfellow

Jack Stringfellow, Chairman
PWR Owners Group

TZ:NJS:rfn

Enclosure 1: Response to NRC Request for Additional Information on PWROG-14001-P, Revision 1, "PRA Model for the Generation III Westinghouse Shutdown Seal." (Proprietary)

Enclosure 2: Application for Withholding, CAW-15-4112 (Non-proprietary) with accompanying affidavit, Proprietary Information Notice and Copyright Notice for Response to NRC Request for Additional Information on PWROG-14001-P, Revision 1, "PRA Model for the Generation III Westinghouse Shutdown Seal."

If you have any questions, please do not hesitate to contact me at (205) 992-7037 or Mr. W. Anthony Nowinowski, Program Manager of the PWR Owners Group, Program Management Office at (412) 374-6855.

cc: PWROG Risk Management Committee
PWROG Licensing Committee
PWROG PMO
J. Gresham, Westinghouse
D. Weaver, Westinghouse
S. Baier, Westinghouse
M. Lucci, Westinghouse
P. Hijeck, Westinghouse
M. Skocik, Westinghouse
A. Mahan, Westinghouse
C. Zozula, Westinghouse
R. Schneider, Westinghouse
M. LaPresti, Westinghouse
J. Andrachek, Westinghouse

Pages B-30 through B-42 are not included in this document since they are proprietary.

Westinghouse

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CAW-15-4112

March 2, 2015

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Enclosure to OG-15-80, "Response to NRC Request for Additional Information on PWROG-14001-P, Revision 1, 'PRA Model for the Generation III Westinghouse Shutdown Seal'" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-15-4112 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

The subject document was prepared and classified as Westinghouse Proprietary Class 2. Westinghouse requests that the document be considered proprietary in its entirety. As such, a non-proprietary version will not be issued.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Pressurized Water Reactor Owners Group (PWROG).

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-15-4112, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

Very truly yours,

J.A. Gresham

James A. Gresham, Manager

Regulatory Compliance

CAW-15-4112

March 2, 2015

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, James A. Gresham, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

James A. Gresham, Manager
Regulatory Compliance

(1) I am Manager, Regulatory Compliance, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is contained in Enclosure to OG-15-80, "Response to NRC Request for Additional Information on PWROG-14001-P, Revision 1, 'PRA Model for the Generation III Westinghouse Shutdown Seal'" (Proprietary), for submittal to the Commission, being transmitted by PWROG letter OG-15-80 and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse for the Pressurized Water Reactor Owners Group is expected to be applicable in other licensee submittals in response to certain NRC requirements for justification of models describing the behavior of the Westinghouse Shutdown Seal in risk-informed regulatory applications and deterministic licensing analyses.

(a) This information is part of that which will enable Westinghouse or Pressurized Water Reactor Owners Group participants in this program to:

(i) Develop plant specific Probabilistic Risk Assessment and deterministic models to describe the behavior of the Westinghouse shutdown seal for postulated plant events that result in a loss of all RCP seal cooling.

(ii) Expeditiously modify risk-informed regulatory applications and gain any required NRC approval of those changes.

(b) Further this information has substantial commercial value as follows:

(i) Plants will install the shutdown seal for its benefits in risk-informed applications and licensing analyses. Westinghouse plans to sell the risk assessment and licensing basis models as the basis for installing the shutdown seal.

(ii) Westinghouse can sell support and defense of the results and conclusions of the subject PWROG report.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide a product that provides similar benefits and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith is the proprietary version of a document furnished to the NRC in connection with requests for generic and/or plant-specific review and approval. The document is to be considered proprietary in its entirety.

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PWROG
PWR Owners Group

Program Management Office
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066

Project 694

May 23rd, 2016

OG-16-188

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject: PWR Owners Group, Risk Management Committee
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)

References:

1. PWR Owners Group Submittal of PWROG-14001-P/NP, Revision 1, "PRA Model for the Generation III Westinghouse Shutdown Seal," OG-14-211, dated July 3, 2014.

On May 17th, 2016 the PWROG met with the NRC at the White Flint Offices to discuss the test plan that the PWROG has developed in support of obtaining a Safety Evaluation on PWROG-14001-P, Revision 1 (Reference 1). The PWROG agreed at the meeting to provide a draft copy of the testing plan to the NRC so that an RAI (on PWROG-14001-P) could be generated which formally requests additional testing so that the current failure probabilities in the topical report may be approved.

The purpose of this letter is to transmit the Pressurized Water Reactor Owners Group (PWROG) draft test plan proposal in support of PWROG-14001-P, Revision 1. This draft plan is being provided to the NRC for information only. The draft proposed test plan is attached as Enclosure 1.

Enclosure 1 contains information proprietary to Westinghouse; therefore Enclosure 2 contains the Westinghouse affidavit requesting that the proprietary information not be disclosed. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations. This document is being provided for information only and no follow-up action is required.

It is respectfully requested that this information which is proprietary to Westinghouse, be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the information identified above or the supporting Westinghouse affidavit should be directed to: Mr. J. A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania, 16066.

Correspondence related to this transmittal should be addressed to:

Mr. W. Anthony Nowinowski, Program Manager
PWR Owners Group, Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive
Suite 380
Cranberry Township, Pennsylvania 16066

If you have any questions, please do not hesitate to contact me at (205) 992-7037 or Mr. W. Anthony Nowinowski, Program Manager of the PWR Owners Group, Program Management Office at (412) 374-6855.

Sincerely yours,

Norman J. Stringfellow

Jack Stringfellow
Chairman and Chief Operating Officer
PWR Owners Group

NJS:WAN

Enclosure 1: LTR-RES-16-82.pdf (Proprietary)

Enclosure 2: CAW-16-4424.pdf (Requesting the proprietary information not be disclosed)

cc: PWROG PMO
PWROG Management Committee
PWROG Risk Management Committee
PWROG Licensing Committee
R. Linthicum, PWROG
J. Stringfellow, PWROG
D. Mirizio, PWROG
M. Higby, Westinghouse
M. Lucci, Westinghouse
T. Becker, Westinghouse

U.S. Nuclear Regulatory Commission
OG-16-188

May 23rd, 2016
Page 3

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DISTRIBUTION NOTICE

Pages B-54 through B-61 are not included in this document since they are proprietary.

Westinghouse Non-Proprietary Class 3

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CAW-16-4424

May 23, 2016

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-RES-16-82, "Generation III **SHIELD®** Shutdown Seal Post-Operational Test Plan" (Proprietary)

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC (Westinghouse), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-16-4424 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

The subject document was prepared and classified as Westinghouse Proprietary Class 2. Westinghouse requests that the document be considered proprietary in its entirety. As such, a non-proprietary version will not be issued.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Pressurized Water Reactor Owners Group (PWROG).

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-16-4424 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

J A Gresham

James A. Gresham, Manager
Regulatory Compliance

CAW-16-4424
May 23, 2016

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, James A. Gresham, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

James A. Gresham, Manager
Regulatory Compliance

(1) I am Manager, Regulatory Compliance, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-RES-16-82, "Generation III **SHIELD®** Shutdown Seal Post-Operational Test Plan" (Proprietary), for submittal to the Commission, being transmitted by PWROG Letter OG-16-188 and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with a proposed post-operational test plan for the Generation III **SHIELD®** shutdown seal, and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to support the PWROG in resolving any issues the NRC may have regarding providing a safety evaluation for the Generation III SDS **SHIELD®** PRA Model Topical Report PWROG-14001-P.

(b) Further, this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of marketing the Generation III **SHIELD**® shutdown seal.

(ii) Westinghouse can sell support and defense of industry guidelines and acceptance criteria for plant-specific applications.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

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Program Management Office
1000 Westinghouse Drive, Suite 380
Cranberry Township, PA 16066

Project 694

September 8, 2016

OG-16-278

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject: PWR Owners Group, Risk Management Committee
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

References:

1. PWR Owners Group Submittal of PWROG-14001-P/NP, Revision 1, "PRA Model for the Generation III Westinghouse Shutdown Seal," OG-14-211, dated July 3, 2014.
2. NRC Letter for Request for Additional Information: PWROG Topical Report PWROG-14001-P/NP, Revision 1 "PRA Model for the Generation III Westinghouse Shutdown Seal," dated August 3, 2016 (TAC NO. MF4397)

On July 3, 2014, in accordance with the Nuclear Regulatory Commission (NRC) Topical Report (TR) program for review and acceptance, the Pressurized Water Reactor Owners Group (PWROG) requested formal NRC review and approval of PWROG-14001-P/NP, Revision 1 (reference 1). Upon NRC review of the TR, the NRC staff has determined that additional information is needed to complete the review per letter dated August 3, 2016 (reference 2). Enclosed please find the PWROG response to the NRC Request for Additional Information (RAI) Questions and text clarifications that will be included in the PWROG-14001-P/NP Topical Report which is all included in the RAI responses. The Non-Proprietary version of the responses to the RAI Questions will be communicated in a separate letter.

Enclosed are:

- Three copies of Response to NRC Request for Additional Information (RAI) Questions regarding Pressurized Water Reactor (PWR) Owners Group PWROG-14001-P/NP, Revision 1 "PRA Model for the Generation III Westinghouse Shutdown Seal," and the proposed clarifications to PWROG-14001-P/NP, Revision 1 (Proprietary). (Enclosure 1)

Also enclosed is Westinghouse letter CAW-16-4457 (Enclosure 2), the accompanying affidavit, Proprietary Information Notice, and Copyright Notice.

The enclosed RAI Response (Enclosure 1) contains information proprietary to Westinghouse Electric Company LLC; it is supported by an affidavit signed by Westinghouse, owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b) (4) of Section 2.390 of the Commission's regulations.

It is respectfully requested that this information which is proprietary to Westinghouse, be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the information identified above or the supporting Westinghouse affidavit should be directed to: Mr. J. A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania, 16066.

Correspondence related to this transmittal should be addressed to:

Mr. W. Anthony Nowinowski, Program Manager
PWR Owners Group, Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive
Suite 380
Cranberry Township, Pennsylvania 16066

If you have any questions, please do not hesitate to contact me at (205) 992-7037 or Mr. W. Anthony Nowinowski, Program Manager of the PWR Owners Group, Program Management Office at (412) 374-6855.

Sincerely yours,

Norman J. Stringfellow

Jack Stringfellow
Chairman and Chief Operating Officer
PWR Owners Group

NJS:WAN

Enclosure 1: PWROG Request For Additional Information Response, LTR-RAM-16-76 (Proprietary)

Enclosure 2: Affidavit for Withholding, CAW-16-4457 (Non-Proprietary)

cc: PWROG PMO
PWROG Management Committee
PWROG Risk Management Committee
PWROG Licensing Committee
R. Linthicum, PWROG
J. Stringfellow, PWROG
D. Mirizio, PWROG
M. Higby, Westinghouse
M. Lucci, Westinghouse
T. Becker, Westinghouse
J. Rowley, US NRC

Pages B-72 through B-96 are not included in this document since they are proprietary.

Westinghouse Non-Proprietary Class 3

Westinghouse

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CAW-16-4457

September 8, 2016

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-RAM-16-76, Revision 0, "Responses to RAI-APHB-11, RAI-APHB-12 and Revisions to Topical Report PWROG-14001-P/PWROG-14001-NP, Revision 1, 'PRA Model for the Generation III Westinghouse Shutdown Seal,' that Address Additional NRC Information Requests" (Proprietary)

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC (Westinghouse), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-16-4457 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Pressurized Water Reactor Owners Group (PWROG).

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-16-4457 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

JA Gresham
James A. Gresham, Manager
Regulatory Compliance

CAW-16-4457
September 8, 2016

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, James A. Gresham, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

James A. Gresham, Manager
Regulatory Compliance

(1) I am Manager, Regulatory Compliance, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-RAM-16-76, "Responses to RAI-APHB-11, RAI-APHB-12 and Revisions to Topical Report PWROG-14001-P/PWROG-14001-NP, Revision 1, 'PRA Model for the Generation III Westinghouse Shutdown Seal,' that Address Additional NRC Information Requests" (Proprietary), for submittal to the Commission, being transmitted by PWROG Letter OG-16-278 and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with Topical Report PWROG-14001-P, and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to support the PWROG in resolving any issues the NRC may have regarding providing a safety evaluation for the Topical Report PWROG-14001-P

(b) Further, this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of marketing the Generation III **SHIELD®** shutdown seal.

(ii) Westinghouse can sell support and defense of industry guidelines and acceptance criteria for plant-specific applications.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith is the proprietary version of a document, furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

COPYRIGHT NOTICE

PWROG
PWR Owners Group

Program Management Office
1000 Westinghouse Drive, Suite 380
Cranberry Township, PA 16066

Project 694

April 17, 2017

OG-17-114

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject: PWR Owners Group, Risk Management Committee
Submittal of Revisions to PWROG-14001-P/NP, Revision 1, "PRA Model for the Generation III Westinghouse Shutdown Seal," per PA-RMSC-1463

References:

1. PWR Owners Group Submittal of PWROG-14001-P/NP, Revision 1, "PRA Model for the Generation III Westinghouse Shutdown Seal," OG-14-211, dated July 3, 2014.
2. NRC Letter for Transmittal of the Draft Safety Evaluation for Pressurized Water Reactor Owners Group Topical Report PWROG-14001-P, Revision 1, "PRA Model for the Generation III Westinghouse Shutdown Seal," dated December 23, 2016 (TAC NO. MF4397)
3. PWR Transmittal of Comments on the Draft Safety Evaluation Regarding PWROG-14001-P/NP, Revision 1, "PRA Model for the Generation III Westinghouse Shutdown Seal," OG-17-24, dated January 20th, 2017.

The Advisory Committee on Reactor Safety (ACRS) provided a number of inquiries on topical report PWROG-14001-P/NP at the February 6th, 2017 subcommittee meeting. Additional information was requested on the following:

- Inadvertent Actuation of the Generation III Shutdown Seal (SDS)
- Human Reliability Analysis (HRA) Dependency on the Operator Action to trip the Reactor Coolant Pumps (RCPs)
- RCP Shaft Sleeve O-ring Failure

Enclosure 1 provides the additional text that will be added to the approved version of the topical report which addresses the ACRS subcommittee inquiries. The revisions will be included in the approved version of the topical report after the Final Safety Evaluation (FSE) is issued.

Enclosed are:

- Three copies of Revisions to Topical Report PWROG-14001-P/PWROG-14001-NP, Revision 1, "PRA Model for the Generation III Westinghouse Shutdown Seal," LTR-RAM-17-17 (Proprietary and Non-Proprietary). (Enclosure 1)

Also enclosed is Westinghouse letter CAW-17-4569 (Enclosure 2), the accompanying affidavit, Proprietary Information Notice, and Copyright Notice.

The enclosed revisions (Enclosure 1) contain information proprietary to Westinghouse Electric Company LLC; it is supported by an affidavit signed by Westinghouse, owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b) (4) of Section 2.390 of the Commission's regulations.

It is respectfully requested that this information which is proprietary to Westinghouse, be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the information identified above or the supporting Westinghouse affidavit should be directed to: Mr. J. A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania, 16066.

Correspondence related to this transmittal should be addressed to:

Mr. W. Anthony Nowinowski, Program Manager
PWR Owners Group, Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive
Suite 380
Cranberry Township, Pennsylvania 16066

If you have any questions, please do not hesitate to contact me at (205) 992-7037 or Mr. W. Anthony Nowinowski, Program Manager of the PWR Owners Group, Program Management Office at (412) 374-6855.

Sincerely yours,

Norman J. Stringfellow

Jack Stringfellow
Chairman and Chief Operating Officer
PWR Owners Group

U.S. Nuclear Regulatory Commission April 17, 2017
OG-17-114 Page 3

NJS:WAN

Enclosure 1: PWROG Revisions to PWROG-14001-P/NP, LTR-RAM-17-17 (Proprietary and Non-Proprietary)
Enclosure 2: Affidavit for Withholding, CAW-17-4569 (Non-Proprietary)

cc: PWROG PMO
PWROG Management Committee
PWROG Risk Management Committee
PWROG Licensing Committee
R. Linthicum, PWROG
J. Stringfellow, PWROG
D. Mirizio, PWROG
M. Higby, Westinghouse
M. Lucci, Westinghouse
M. Degonish, Westinghouse
J. Andrachek, Westinghouse
M. LaPresti, Westinghouse
T. Becker, Westinghouse
B. Benney, US NRC

Revisions to Topical Report PWROG-14001-P/PWROG-14001-NP, Revision 1, "PRA Model for the Generation III Westinghouse Shutdown Seal."

April 12, 2017

Author:
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Matthew M. Degonish
Risk Applications and Methods I

Verifier:
*Electronically Approved**
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Melissa A. Lucci
Risk Applications and Methods I

***Electronically approved records are authenticated in the Electronic Document Management System.**

Inadvertent Actuation

The following text will be added to Subsection 2.5.5.6 of TR PWROG-14001-P/PWROG-14001-NP (Reference 1).

The design of the SDS assembly provides additional defenses against inadvertent actuation of the SDS. This is accomplished by providing margins between the design specifications and the expected operational conditions as well as assuring that catastrophic component failure does not lead to inadvertent actuation. Consensus judgment from an expert panel comprised of senior Westinghouse subject matter experts in design, field services, plant operations, and probabilistic risk assessment was used to develop an estimate of the probability of an inadvertent actuation using proven PRA methodologies. The resulting frequency of inadvertent actuation, based on the consensus expert judgment process, is [ ]^{a,c}. A detailed evaluation and additional discussion on inadvertent actuation is provided in Appendix A.

The following text will be added as Appendix A in TR PWROG-14001-P/PWROG-14001-NP (Reference 1).

This appendix contains a detailed evaluation and discussion on the probability of an inadvertent actuation of the SDS. Inadvertent actuation of the SDS while the RCP is in operation will damage the SDS and prevent the SDS from performing its function, should it be needed following a loss of seal cooling. As a consequence of the significance of the event, and the low likelihood of detection given premature actuation, considerable effort has gone into including design features to prevent inadvertent actuation. In order to quantify the likelihood of inadvertent actuation of the SDS an expert panel comprised of senior Westinghouse subject matter experts in design, field services, plant operations, and probabilistic risk assessment was convened to perform a failure modes and effects analysis of the SDS and to develop an estimate of the probability of an inadvertent actuation using proven PRA methodologies. The following discussion describes the events that could lead to an inadvertent actuation of the SDS, the expected consequences, and the assigned probabilities.

Fault Tree Model for Inadvertent Actuation

To systematically assess the frequency of inadvertent actuation of the SDS, a fault tree model was developed. Subsequently, numerical probabilities were assigned to the fault tree basic events and the fault tree was quantified to yield a frequency of inadvertent actuation.

The fault tree model structure and numerical assignments of associated basic event probability were based on the engineering judgment of a group of experienced Westinghouse engineers who considered their own background as well as input from other subject matter experts. The full fault tree structure is shown in Figure A-1 and includes the following possible causes of inadvertent actuation:

- [ ]^{a,c}
- [ ]^{a,c}
- [ ]^{a,c}
- [ ]^{a,c}

Inadvertent actuation is defined as any failure of the SDS or SDS components that would result in the actuation of the SDS at seal leak-off temperatures below [ ]^{a,c}. Seal leak-off temperatures above [ ]^{a,c} are indicative of loss of all seal cooling conditions for which the SDS is designed to actuate. Therefore, by definition, if the seal leak-off temperature exceeds this value and the SDS actuates, actuation of the SDS is not inadvertent. Therefore, by definition, inadvertent actuation is only considered possible when the SDS actuates without a prior loss of all seal cooling accident to the affected RCP. If a loss of all seal cooling accident occurs, and the SDS actuates prior to reaching its actuation temperature, this occurrence is not considered inadvertent. This type of failure is considered a random failure of the SDS.

[

]^{a,c}

Figure A-1: Inadvertent Actuation Fault Tree

Fault Tree Quantification

The assignment of numerical values to the basic events in the fault tree was performed by the expert panel consensus process. The assignment of numerical values considered the SDS design and quality control, as well as, extensive testing and analysis that have been performed to prevent inadvertent actuation of the SDS as described in TR-FSE-14-1-P, Revision 1 (Reference 3). The guidelines for the assignment of numerical values to the basic events in the fault tree and the basis for the assignment is described below.

A set of criteria was developed for translating engineering judgment into quantifiable values. The criteria are similar to the guidelines for probability assignment used in NUREG/CR-6771. The Table A-1 criteria were provided to expert panel members for use in assigning probabilities to physical events such as component failures of relevance to the inadvertent actuation failure mode.

Table A-1: Guidelines for Assigning Conditional Probabilities to Physical Events

Value	***Description***
1.0E-01	*The indicated outcome is UNLIKELY but it cannot be supported by analysis or testing. It is a credible outcome, considering uncertainties, such that it could occur during the plant lifetime.*
5.0E-02	*The indicated outcome is HIGHLY UNLIKELY. Analyses or testing have not been done directly or cannot rule it out completely. However, arguments in favour of this outcome are not supported by available testing or analysis.*
1.0E-02	*The indicated outcome is VERY UNLIKELY. Available testing or analyses show that the outcome did not occur. Consideration of these uncertainties might lead to this outcome but no analytical or experimental support can be found.*
1.0E-03	*The indicated outcome is EXTREMELY UNLIKELY. Available testing or analyses show that the outcome did not occur and consideration of the uncertainties cannot reasonably be found to lead to this outcome event when no analytical or experimental support can be found.*
1.0E-04	*The indicated outcome is ALMOST IMPOSSIBLE. No analysis or testing is available to support this result even considering relevant uncertainties. It has credibility only if a number of unsupported (but not demonstrably incorrect) assumptions are made. All available analysis and experiments support alternate outcomes.*

Some of the fault tree entries require an estimation of the probability that an out-of-specification item would not be detected in the Westinghouse or supplier quality program. It was assumed that all components will be procured and shipped under the Westinghouse Quality Assurance Program and that all suppliers would have an approved quality program that is acceptable under the Westinghouse Quality Assurance Program. The Table A-2 criteria were provided to expert panel members for use in assigning probabilities to non-compliances under the vendor and Westinghouse Quality control programs that would contribute to events that could impact SDS inadvertent actuation.

Table A-2: Guidelines for Assigning Conditional Probabilities to Quality Program Events

Value	***Description***
1.0E-01	*The indicated outcome is UNLIKELY. Quality program is based on observations by Westinghouse that are not completely independent from supplier quality checks or a supplier on the Westinghouse Quality Supplier List with whom issues have occurred in the past.*
1.0E-02	*The indicated outcome is VERY UNLIKELY. Quality program is based on testing or observations by Westinghouse that are not completely independent from supplier quality checks or a supplier on the Westinghouse Quality Supplier List with whom little (or no) experience has accumulated.*
1.0E-03	*The indicated outcome is EXTREMELY UNLIKELY. Quality program is based on independent testing by Westinghouse or a supplier on the Westinghouse Quality Supplier List with whom experience has accumulated.*
1.0E-04	*The indicated outcome is ALMOST IMPOSSIBLE. Quality program is based on no human interactions or decisions.*

The criteria above provide a basis for qualifying the fault tree based on engineering judgment by an expert panel to provide an estimate of the yearly frequency of inadvertent actuation of the SDS during normal operation on a per pump-year basis.

This section provides the numerical value assigned to each of the basic events identified in the fault tree along with the basis for that assignment as developed by the expert panel.

[ ]^{a,c}

[ ]^{a,c}

[ ]^{a,c}

[ ]^{a,c}

[ ]a,c

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[ ]^{a,c}

<u>*Conclusions*</u>

The resulting frequency of inadvertent actuation, based on the consensus judgment process, is [ ]^{a,c}. The resulting 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 other failure modes discussed in Section 2.5 are also on a per-pump basis. Consistent with SY-A15 of the ASME/ANS Standard, the calculated value for inadvertent actuation can be screened from the fault tree model since the value is greater than two orders of magnitude lower than the proposed failure frequency of the SDS. Therefore, the inadvertent actuation failure mode should not be included in the fault tree model for the SDS.

Update to Discussion on HRA Dependency for Operator Action to Trip the RCPs

In discussion of the PRA model for the SDS, it was noted that there was a need to expand the discussion on the operator action to trip the RCPs. The current text focuses on the modeling of an uncomplicated single operator action to trip the RCPs whose consequence would change dependent on the time the action was completed. The discussion did not consider the potential that the action could fail due to hardware/equipment failures. Should the operator attempt to trip the RCPs and fail due to hardware failures, the operator may then potentially recover by manually opening the reactor pump trip breakers. The model implicitly assumes that failure of the operator to take the initial action in the required time cannot be recovered. The following information will be added to the current text in Subsection 2.5.5.5 of TR PWROG-14001-P/PWROG-14001-NP (Reference 1).

When evaluating the operator action to trip the RCPs, analysts should consider the relevance of hardware/equipment failures and whether or not the operator action to trip the RCPs is dominant as compared to the hardware/equipment failures. Analysts should refer to the screening criteria provided in the SY supporting requirement of the ASME/ANS PRA Standard when determining whether or not to include hardware/equipment failures as part of the operator action to trip the RCPs. In addition, any dependency analysis for the operator action to trip the RCPs, and subsequent recovery actions to trip the RCPs by alternate means, should be done in accordance with the HR supporting requirement of the ASME/ANS PRA Standard.

Update to Discussion on O-ring Failure Mode in the Model 93A RCP

In the discussion of the PRA model for the SDS, it was noted in Subsection 2.5.1 of PWROG-14001-P/PWROG-14001-NP (Reference 1) that "In construction of the PRA model, [ ]^{a,c}." The following discussion expands on the basis for not developing a model 93A RCP seal specific PRA model. The following information will be added to the end of Subsection 2.2.3 of TR PWROG-14001-P/PWROG-14001-NP (Reference 1).

The Model 93A RCP SDS configuration includes a [ ]^{a,c} This implementation differs from the other Westinghouse RCP designs that use the SDS to [ ]^{a,c}. However, while the Model 93A RCP has [ ]^{a,c}. The use of the [ ]^{a,c} was largely based on a review of the test program data, and the inherent conservatism regarding how the baseline failure probabilities were determined. These considerations are presented below.

The overall approach of the SDS endurance qualification program was to efficiently demonstrate the high reliability of the SDS function for an [ ]^{a,c}. For purposes of qualification testing, the most limiting [ ]^{a,c} were identified. Of particular note was the SDS endurance testing performed on the Westinghouse model 93A-1 pump configuration. This configuration has the [ ]^{a,c}. Another important feature of the SDS qualification testing was that the testing was [ ]^{a,c}. Specifically, the seal ring endurance tests typically lasted between [ ]^{a,c}. Furthermore, it was assumed for purposes of the PRA, that the SDS [ ]^{a,c}. In most situations, loss of seal cooling events will be followed by a prompt plant shutdown and a rapid cooldown which would reduce the RCS pressure and temperature to [ ]^{a,c},

significantly reducing the challenge to the SDS within hours of the initiating event. Thus, the SDS seal failure probability [ ]^{a,c} given a loss of seal cooling can be considered bounding.

As discussed above, the Model 93A RCP includes an [ ]^{a,c}. Separate qualification tests were performed for these O-rings. O-rings can significantly degrade under exposure to high temperatures. The reliability of the O-ring was separately investigated in two test programs. One test program included [ ]^{a,c}. Each test involved two O-rings. These tests exposed the [ ]^{a,c}. The endurance test was continued for [ ]^{a,c}. The pressure profile was intended to bound the maximum pressure/pressurizer heat losses. Following each endurance test, a [ ]^{a,c}. O-ring failure was noted by the [ ]^{a,c}. The O-ring failure pressure was recorded and a pressure "survivability margin" was determined. The [ ]^{a,c} temperature environment was maintained during [ ]^{a,c}. These tests demonstrated the ability of the seal to survive harsh conditions for a timespan of a [ ]^{a,c} with sufficient residual strength to maintain sealing capability to pressures above [ ]^{a,c}. Following this test program, Westinghouse conducted an additional [ ]^{a,c}. As with the earlier test program, the pressure profile included a [ ]^{a,c}. The O-rings maintained their sealing capability throughout both test programs. In total, O-ring experiments included a total exposure time of approximately [ ]^{a,c}.

Analyses of the model 93A seal heatup following actuation of the SDS indicates that the reduced RCS leakage into the seal results in a relatively slow seal heatup process, requiring approximately [ ]^{a,c}. It is recognized that the ERGs instruct the operator to cooldown the RCS shortly after the onset of the loss of seal cooling. RCS cooldown will likely commence well in advance of the sealing ring temperature reaching its maximum asymptotic temperature of [ ]^{a,c}. It is estimated that when a realistic cooldown is credited, the effective O-ring exposure temperature would be generally bounded by [ ]^{a,c}. Applying an Arrhenius based acceleration factor based on the properties of the O-ring material it can be estimated that the effective duration of the seals at the anticipated post SDS actuation condition to be closer to [ ]^{a,c}. These considerations result in an effective [ ]^{a,c}. Therefore, failure of the [ ]^{a,c}. Given the fact that both the seal ring and O-ring failures are based on bounding data, it was judged that creating a second seal model to capture this failure mode was unnecessary. As a result of the bounding nature of the SDS failure probability, particularly as it applies to the Model 93A with [ ]^{a,c}, no adjustment to the baseline PRA value for failure to seal was warranted. Furthermore, it was judged that this approach reduces model complexity, maintains consistency among the PRAs, and simplifies model implementation.

References:

1. PWROG-14001-P/NP, Revision 1, "PRA Model for the Generation III Westinghouse Shutdown Seal," July 2014.

Westinghouse Non-Proprietary Class 3

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CAW-17-4569

April 11, 2017

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-RAM-17-17 P-Attachment, Revision 0, "Revisions to Topical Report PWROG-14001-P/PWROG-14001-NP, Revision 1, 'PRA Model for the Generation III Westinghouse Shutdown Seal' " (Proprietary)

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC ("Westinghouse"), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-17-4569 signed by the owner of the proprietary information, Westinghouse. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Pressurized Water Reactor Owners Group (PWROG).

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-17-4569 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

J. A. Gresham

James A. Gresham, Manager
Regulatory Compliance

CAW-17-4569

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, James A. Gresham, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse") and declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

Executed on: 4/11/17

J A Gresham
James A. Gresham, Manager
Regulatory Compliance

(1) I am Manager, Regulatory Compliance, Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, (e.g., by optimization or improved marketability).

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-RAM-17-17 P-Attachment, Revision 0, "Revisions to Topical Report PWROG-14001-P/PWROG-14001-NP, Revision 1, 'PRA Model for the Generation III Westinghouse Shutdown Seal' " (Proprietary), for submittal to the Commission, being transmitted by PWROG letter OG-17-114. The proprietary information as submitted by Westinghouse is that associated with requests for review and approval of Topical Report PWROG-14001 and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to support the Final Safety Evaluation for Topical Report PWROG-14001.

(b) Further, this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of marketing the Generation III shutdown seal.

(ii) Westinghouse can sell support and defense of industry guidelines and acceptance criteria for plant-specific applications.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and non-proprietary versions of a document, furnished to the NRC associated with requests for review and approval of Topical Report PWROG-14001and may be used only for that purpose.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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