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10 CFR 50.59(d)(2)
10 CFR 72.48(d)(2)

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397 INDEPENDENT SPENT FUEL STORAGE INSTALLATION, DOCKET NO. 72-35 BIENNIAL COMMITMENT CHANGES AND 50.59/72.48 REPORT**

Dear Sir or Madam:

Enclosed is the Columbia Generating Station 2008 – 2009 Commitment Changes and 50.59/72.48 Report. This report is submitted pursuant to 10 CFR 50.59(d)(2), 10 CFR 72.48(d)(2), and Guidelines for Managing NRC Commitment Changes (NEI 99-04). There are no commitments being made to the NRC by this letter, however, five existing commitments have been changed.

If you have any questions or desire additional information pertaining to this report, please contact Mr. MC Humphreys at (509) 377-4025.

Respectfully,

D.K. Atkinson for
D.K. ATKINSON
WS Oxenford
Vice President, Nuclear Generation & Chief Nuclear Officer

Attachment: Biennial Commitment Changes and 50.59/72.48 Report

cc: NRC Region IV Administrator
NRC NRR Project Manager
NRC Senior Resident Inspector/988C
RN Sherman – BPA/1399
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IE07

NMSS

10 CFR 50.59 Changes, Tests, and Experiments

This section contains a brief description of any changes, tests, and experiments, including the summary of the evaluations for activities implemented during 2008 and 2009 that were assessed pursuant to 10 CFR 50.59 requirements.

Energy Northwest evaluated the changes summarized below and determined prior NRC approval was not required.

5059-08-0001 "Accept As Is Increased Cask Heat Load During ISFSI Cask Loading Operations"

Brief Description
The heat load assumed in the secondary containment drawdown analysis is increased due to the presence of an ISFSI cask on the refuel floor.

The cask heat load input parameter assumed in the secondary containment drawdown analysis for the plant 10 CFR 50 license is being aligned with the bounding value from CoC Amendment 2 implementation in the 10 CFR 72 license. The cask heat load is changing from 23 kW to 28 kW due to an input parameter change from another license.

Summary of Evaluation
The increase in drawdown time due to the change in heat load in a dry cask on the refuel floor does not cause any increase in the consequences of an accident as previously evaluated because the dose evaluation uses a parameter value of a drawdown time that bounds the drawdown result. No change is needed to the consequences evaluation. The methodology for the reactor building drawdown model changes the heat load modeled in the building spaces, but the increase in heat load does not compromise the standby gas treatment system's ability to draw down the secondary containment, and still meets the drawdown time of 20 minutes stated in the FSAR. The change of this element of this analysis methodology yields a result that is essentially the same as the analysis of record.

5059-08-0003 PDC 6396 "Upgrade Feedwater Turbine Controls with Electric Actuators"

Brief Description
This change will replace the existing digital (Lovejoy) reactor feedwater turbine control system and its electro-hydraulic valve positioner with a redundant Invensys/Triconex programmable logic controller (PLC) based system that drives an electric linear actuator and its new valve control linkage. The new system will perform all the functions of the existing system but with improved reliability and fault tolerance. The existing hydraulic servos, linkages, and associated filters, valves and piping will be removed and new electric actuators with associated linkages and redundant position feedback will be installed. The Operator Control Stations (OCS) on panel H13-P603 will be replaced by 2 touch screens (Human Machine Interface) on board A (H13 P840). The new cabinets will also be installed in the Main Control Room and Turbine Building elevation 441' to house the new equipment.

The existing reactor feedwater turbine governor valves and speed controls have required numerous downpowers, shutdowns and have become unreliable. Hydraulics has exhibited stick/slip, resulting in speed "hunting." Electronics have exhibited changes in setpoint in Manual Demand (MDEM) mode.

Summary of Evaluation
The feedwater system provides a reliable source of high purity feedwater during both normal operation and anticipated transient conditions. The system is designed with sufficient capacity to provide for 110% of the feedwater flow at rated load. This provides sufficient margin to provide flow under anticipated transient conditions. The feedwater system is not required for safe shutdown of the reactor.

The feedwater flow is regulated by controlling the speed of the turbine-driven feedwater pumps to deliver the required flow to the reactor vessel. The turbine speed controls regulate the turbine speed based upon input from the feedwater level control system (FWLCS).

No FSAR described accident analyses are adversely impacted by this modification. The new system retains all of the control functions of the existing system. There are no new control functions that directly interface with any safety SSCs. The function of the replacement reactor feedwater turbine control system will not change the operating or design parameters of any other plant system. The new reactor feedwater turbine control system has been designed as a highly reliable system. This design is achieved through implementation of a combination of using highly reliable components and application of fault tolerant design that meets the guidelines of Electric Power Research Institute (EPRI) TR-102348 for digital upgrades.

A Failure Modes and Effects Analysis (FMEA) was performed for the new hardware and software to verify that a single failure of the new system does not increase the frequency or consequences of normal operating transients and Anticipated Operational Occurrences (AOO) such as reactor scram, loss of feedwater or feedwater level control system failure.

This evaluation has shown that no increase in frequency of occurrence or consequences of an accident or malfunction of an SSC important to safety previously evaluated in the FSAR will occur. The proposed activity does not result in a design basis limit for a fission product barrier being altered or exceeded. This design change maintains the FSAR design function of the feedwater system and does not create an accident of a different type or a malfunction of an SSC important to safety with a different result than previously evaluated in the FSAR.

5059-08-0004 "Accept As Is Determination For HPCS-V-12 Leakage"
Brief Description
This evaluation addresses the "accept-as-is" determination of PERA 204-0825-01 for HPCS-V-12. HPCS-V-12 is the four inch minimum flow valve for the high pressure core spray pump HPCS-P-1. The valve is a motor-operated split-disc gate valve. It was modified to prevent pressure locking. The modification provided a 0.5" line from the valve bonnet to piping downstream of the valve. When HPCS-P-1 is running, the pump

pressure unseats the upstream disc, permitting the leakage into the bonnet, through the 0.5" line, to the upstream pipe. The upstream pipe directs the leakage to the suppression pool. The leakage is approximately 15 gpm.

PER 204-0825 documented valve leakage that was attributed to its modification to prevent pressure locking. 5059SCREEN-08-0166 determined that this leakage was potentially adverse to the functions of HPCS and the suppression pool.

Summary of Evaluation
When HPCS-P-1 is running, the minimum flow valve, HPCS-V-12, leaks approximately 15 gpm past its seat, with its leakage directed to the suppression pool. The leakage is the result of a 0.5 inch bonnet drain that was installed to prevent the valve's pressure locking. The leakage corresponds to a rate of level change of less than 0.3 inches per hour in the suppression pool. In modes 1, 2 and 3, suppression pool level is normally maintained within a 1.5" bandwidth, which is within the 4" bandwidth required by Technical Specifications (TS). The leakage was documented in PER 204-0825.

The HPCS minimum flow valve has no role in the initiation of any FSAR-described accident, so its leakage has no effect on the frequency of occurrences of accidents. The leakage past the valve seat causes a minor increase of inventory to the suppression pool during system testing and RPV injection. The slow rate of increase, together with procedural guidance regarding the leakage, and redundant level instrumentation, ensures that the operator is aware of level change, and has ample time to respond to the change and ensure compliance with TS limits.

During surveillance testing of HPCS, when the system is aligned in a Condensate Storage Tank (CST) to CST mode, there will be a slight depletion of CST inventory. However, in this alignment, HPCS is not operable. RCIC, while aligned to take suction from the CST, does not credit its inventory; the suppression pool is the credited source of water. The evaluation determined that there is not a more-than minimal likelihood of the occurrence of a malfunction of an SSC important to safety.

During HPCS injection to the reactor pressure vessel (RPV), the leakage past HPCS-V-12 results in a corresponding reduction of pump flow to the RPV. However, there is adequate margin in the performance of HPCS-P-1 to ensure that TS flow requirements are met and, accordingly, the system will perform its safety function. The leakage will not increase the consequence of any accident or any malfunction of an SSC.

The leakage of HPCS-V-12 has no effect on any SSC that interfaces with HPCS, the suppression pool, or the CST and, thus, does not create the possibility of a new type of accident, nor create a malfunction with a result that is different than previously evaluated. All HPCS and suppression pool functions are maintained, ensuring that design basis limits are not exceeded for all fission product barriers.

The accept-as-is disposition of the HPCS-V-12 leakage does not entail the departure from any existing method of evaluation used in safety analyses, or in establishing the design bases.

5059-09-0001 Advanced PDC 6553 "Change To Computer Code Used To Perform Criticality Analysis"
Brief Description
The proposed change is to use the computer code MCNP Version 4A to perform the criticality analyses for the new fuel vault and spent fuel racks for the GE14 fuel design. Columbia will be loading GE14 fuel for the first time at Cycle 20. This is a different code than has been used previously for these analyses at Columbia. Specifically, the analyses were performed with MCNP-4A using the ENDF/B-V cross-sections. Previously, the analyses for other fuel types were performed with PDQ and KENO as documented in FSAR 9.1.

The GE14 fuel design is analyzed, in part, by GE-Hitachi (GEH). The GEH methodology for performing criticality analyses uses the MCNP code and not KENO or PDQ.

Summary of Evaluation
This change will utilize the MCNP Version 4A (MCNP-4A) code to perform the criticality analyses for the new fuel vault and spent fuel pool for the GE14 fuel design. GE14 is being loaded for the first time at Columbia at Cycle 20. Only question 8 is applicable to this change. This change is not considered a departure from a method of evaluation described in the FSAR since its use is (a) based on sound engineering practice, (b) appropriate for the intended application and (c) within the limitations of the applicable Safety Evaluation Report (SER). MCNP is a standard and accepted tool for spent fuel pool criticality analysis. The NRC has previously reviewed and approved the use of MCNP for criticality safety applications at nuclear power plants as documented in SERs for the following facilities: Fitzpatrick, Davis-Besse Unit 1, H. B. Robinson Unit 2, Turkey Point Units 3 and 4, and Arkansas Nuclear One Unit 2.

5059-09-0002 PDC 6553 "Cycle 20 Core Design"
Brief Description
The Cycle 20 reload design is analyzed using Global Nuclear Fuel (GNF) Methodology. The proposed changes are updates to the approved GNF methodology or changes due to error corrections to computer codes associated with the approved methodology.

The Cycle 20 reload design is evaluated using GNF Methodology.

Summary of Evaluation
The changes identified in the approved Global Nuclear Fuel (GNF) methodology as applied to the Cycle 20 reload analysis do not represent a departure from a method of evaluation. The results from these changes are all either conservative or essentially the same. The methodology complies with the associated NRC Safety Evaluation Report (SER).

5059-09-0004 PDC 8484 "DEH Software Modification"
Brief Description
The functional objective of EC-8484 is to implement software changes which were determined to be needed after the review of the Post scram transient data on 08/05/09. These changes will provide system enhancements to improve system reliability of the

Digital Electro-Hydraulic (DEH) Control System for controlling pressure following a turbine trip from 100% power. The focus of this Evaluation will be to address the DEH logic that will be revised so that the pressure controller will remain in Auto mode upon high, or a low failure signal of pressure transmitters (MS-PT-1A, MS-PT-1B & MS-PT-1C). The additional software changes being made were evaluated per 50.59SCREEN-09-0212 and do not require a 50.59 Evaluation.

Following a turbine trip from 100% power resulting in Forced outage 09-04, the turbine bypass valves did not control pressure in automatic during the post scram transient. The pressure transient due to the load reject resulted in throttle pressure increasing above the transmitter high fail setpoint and resulted in generating a failed signal for all three transmitters. The three simultaneous pressure transmitter failure signals caused the pressure controller to transfer to manual (Ref. AR CR 202385) as programmed. From review of the post scram transient response, it has been determined that the preferred action to be for the controller function to remain in automatic, rather than switching to manual. Since the DEH remains in auto, it will go back to modulating bypass valves when pressures inputs return to normal range. Thus, the pressure controller will provide optimized control to respond properly to normal and transient conditions.

Summary of Evaluation
The DEH Control System provides automatic or manual control of turbine-generator speed and load by positioning of the governor valves and throttle valves. DEH also accomplishes the task of controlling reactor pressure during normal plant operation (including startup and shutdown) by positioning the governor valves and the bypass valves.

This modification retains all of the functions of the existing system. This modification does not introduce the possibility of a change in the likelihood of a malfunction because no new failure modes are introduced. There are no new control functions that directly interface with important to safety SSCs. This modification will not change the operating or design parameters of any other plant system.

This modification has no effect on the pressure regulator failure open transient or ATWS event (FSAR 15.1.3 and 15.8.9) since redundancy is not reduced. The change in DEH logic to keep the pressure controller in auto mode upon a high or low failure signal does not affect the analyzed pressure regulator failure closed transient event (FSAR 15.2.1).

This evaluation has shown that no increase in frequency of occurrence or consequences of an accident or malfunction of an SSC important to safety previously evaluated in the FSAR will occur. The proposed activity does not result in a design basis limit for a fission product barrier being altered or exceeded. This design change maintains the FSAR design function of DEH System and does not create an accident of a different type or a malfunction of an SSC important to safety with a different result than previously evaluated in the FSAR.

5059-09-0005 PDC "8514 Modification of RFW Trip Logic"

Brief Description

The RFW-P-1A and RFW-P-1B low suction pressure trip setpoint is being changed to 300 psig. The trip setpoint for RFW-P-1A is now at 331 psig, and RFW-P-1B is currently at 291 psig. The trip setpoint for RFW-P-1A is being reduced to provide more operating margin, and optimize its ability to survive a transient low-suction pressure event. The trip set point for RFW-P-1B is being raised to provide protection for the pump against cavitation for maximum flow conditions (i.e. > 23,000 gpm), when there is a higher required NPSH. The 300 psig exceeds that required to meet minimum NPSH requirements over the full range of operating conditions for the RFW pumps.

The RFW pump low suction pressure trip time delay (COND-RLY-62/28A) setpoint is being increased for RFW-P-1A from 4 seconds to 10 seconds. The time delay setpoint for RFW-P-1B will remain unchanged at 4 seconds. The time delay change to 10 seconds will provide a longer time for system pressure to recover during suction line pressure transients. This will increase the likelihood that RFW-P-1A will continue to operate and perform its design function.

Summary of Evaluation

The proposed change was evaluated, and it was determined that engineering change (EC) 8514 may be implemented without prior NRC approval. The evaluation's conclusions are summarized below.

The evaluation determined that the proposed changes to the low-pressure set point for RFW-P-1A will not result in the increase of frequency of an accident previously evaluated in the FSAR. The FSAR describes two loss-of-feedwater events. The limiting feedwater fault is the loss of reactor feedwater (RFW) flow with a failure to scram. The setpoint and time delay changes to the low-pressure setpoints of RFW-P-1A and 1B have no credible effect on the capability to scram the reactor.

The FSAR also cites a pump failure or a failure in the feedwater controller that may cause the loss of feedwater flow to the reactor pressure vessel (RPV). The FSAR classifies this failure as an incident of moderate frequency that is non-limiting.

The proposed activity will reduce the frequency of loss-of-feedwater events by adding a time-delay to the low-suction pressure trip of RFW-P-1A. The new proposed time delay, 10 seconds, will provide assurance that RFW-P-1A can operate through a system pressure transient, and continue to provide flow to the RPV. It will also improve system response to a presumed failure of the feedwater controller by maintaining RPV level below level 8.

The proposed change in pressure setpoints will also assist in the RFW pump's ability to survive a low suction pressure transient; however, the benefit is lesser than that of the time delay change. The lowered setpoint of RFW-P-1A will provide a slight increase in margin to trip for less severe transients, while the higher trip setpoint of RFW-P-1B would result in a incrementally faster trip which, in turn, will reduce suction flow in the common header to the RFW pumps. This reduced flow will facilitate the recovery of suction pressure, and shorten the duration of the transient.

The change to the setpoints and time delay will not have more than a minimal increase in the likelihood of a malfunction of an RFW pump. Operation of RFW-P-1A pump at maximum flow (~23,000 gpm) under worst-case instrument error may result in cavitation. Cavitations will cause accelerated wear on pump rotating elements. However, this damage is gradual, and would not result in the catastrophic failure of the pump. Any additional wear on the pump due to an incidental transient is expected to be minimal. Degradation to the cavitation-induced wear would be detectable during vibration monitoring and regular maintenance.

The increase of the low pressure setpoint in RFW-P-1B from 291 psig to 300 psig provides more conservative protection to RFW-P-1B, reducing the likelihood of cavitation damage under conditions of maximum flow.

The setpoint and time-delay change for the low-suction pressure trip of the RFW pumps will have no adverse effect on any other important-to-safety structure, system and component (SSC).

The setpoint changes will not result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR. The RFW pumps have no role in the mitigation of radiological consequences of any FSAR-described accident. The setpoint changes enhance the ability of at least one RFW pump to supply water to the RPV after a low-suction pressure transient. This, in turn, provides a higher level of assurance that the core will remain covered at all times and, thus, minimize challenges to emergency core cooling systems (ECCS) systems.

The setpoint changes do not create the possibility for an accident of a different type. The proposed setpoint changes and time delay extension, together, lower the probability that transients in the condensate / feedwater system will cause a complete loss of feedwater. No new failure modes are introduced to other condensate / RFW components, or to any important-to-safety SSC.

The changes to the RFW pump suction pressure setpoints do not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR. Industry experience has shown that the staggering of time-delays for low-suction pressure trips is more effective in preventing unnecessary scrams than staggering the pressure setpoints, as Columbia currently has. The proposed changes are consistent with this experience, and the recommendations of the BWR Owners Group, in their "Recommendations for Reducing Unnecessary Reactor Scrams."

The proposed change to RFW-P-1A - lowering the low-pressure setpoint and the addition of a time delay to the trip of RFW-P-1A adds the potential for brief periods of cavitation in that pump, with resultant wear on rotating components. This wear will not result in the loss of pump function but rather a gradual degradation of rotating elements. This is bounded by the loss-of-feedwater event which, in itself, is a non-limiting Chapter 15 event. For the limiting loss-of-feedwater event, an anticipated transient without scram (ATWS) event, a complete failure to scram is postulated to occur for all reactor protection system (RPS) scram signals. None of the proposed changes will have any adverse effect on the RPS or any other important-to-safety SSC.

The proposed setpoint changes do not result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered. The pressure and time-delay changes to the low-suction pressure setpoint of the RFW pumps do not cause changes in any system operating parameters for RFW, nor any other system. The change enhances the system's ability to provide uninterrupted flow of condensate to the RPV, and maintains a non-safety related capability to supply water to the core.

The proposed changes to RFW pump setpoints do not result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses. The low-suction pressure setpoint of RFW-P-1A and 1B is not in any design bases or safety analyses described in the FSAR.

10 CFR 72.48 Changes, Tests, and Experiments

This section contains a brief description of any changes, tests, and experiments, including the summary of the evaluations for activities implemented during 2008 and 2009 that were assessed pursuant to 10 CFR 72.48 requirements.

Energy Northwest evaluated the change summarized below and determined prior NRC approval was not required.

7248-07-0001 "Accept As Is Reduced Cooling Time And Higher Heat Load Condition For The Amendment 1 Casks"
Brief Description
This activity resolves the condition adverse to quality created by the Independent Spent Fuel Storage Installation (ISFSI) Amendment 1 casks loaded with spent fuel with cooling time values not in conformance with the shielding analysis in the site boundary dose calculation. The corrective action accepts the reduced cooling time and higher heat load condition "as is" for the 15 Amendment 1 casks by performing the offsite dose analysis again with input values that correspond to the conditions that exist and bound the cooling time parameters in the loaded casks.

NEI 96-07 Appendix B Section B4.4 requires that 10 CFR 72.48 be applied to the compensatory actions that address and resolve the non-conformance identified in CR 2-07-08466.

Summary of Evaluation
The ISFSI site boundary dose was computed prior to the original ISFSI license campaign using a fuel cooling time of 13 years, and that value was cited in the UFSAR (72.212 Report). Almost all of the bundles loaded in the 2002 and 2004 cask loading campaigns had been cooled for 5.7 to 10 years rather than 13 years at the time of loading into the 15 casks. This activity resolves the condition adverse to quality created by these Amendment 1 casks loaded with spent fuel with cooling time values not in conformance with the confinement or shielding analyses supporting the site boundary dose calculation. The corrective action accepts the reduced cooling time and higher

heat load condition "as is" by performing the offsite dose analyses again with input values that bound the cooling time parameters in the loaded casks. The accident dose consequence of this activity resulted in essentially the same offsite dose, and therefore this change did not cause more than a minimal increase in consequences of an accident as previously evaluated. However, the dose consequence of both the normal condition (assumed 1% fuel failure) and the off-normal (10% fuel failure) is computed from effluent and direct dose with the applicable malfunctions. The actual consequence values are a small fraction of the limit. While the total dose did increase, it did not exceed the minimal threshold because the use of Amendment 1 casks has been constrained to these 15 casks already loaded. Since all projected Amendment 1 casks have not been loaded, the dose did not exceed the 10% margin to the limit criterion. Therefore, the consequence of a malfunction is considered to be minimal.

Regulatory Commitment Changes (NEI 99-04 Process)

This section reports changes to regulatory commitments consistent with the information pertaining to Regulatory Commitment Changes (RCC) and is included pursuant to NEI 99-04 criteria for reporting.

In response to Generic Letter 89-13 (RCC-110791-00)
This commitment was created to address the NRC's request that licensees conduct a test program to verify the heat transfer capability of all safety related heat exchangers cooled by service water. At Columbia Generating Station (CGS), testing of the heat transfer capabilities of the water-to-water heat exchangers are performed via a separate commitment listed in letter GO2-90-017. Where testing is not possible, the heat exchangers are inspected and cleaned. This information is trended. Thermal performance testing results are monitored and trended.

The commitment change involves cooling water flow and differential temperature tests described in the original commitment. The testing is redundant to that performed for thermal performance testing as committed to in the same Generic Letter 89-13 response. Therefore the commitment is being changed to include the option to discontinue this testing based on good trending results. The original commitment stated:

> WNP-2 will monitor and record cooling water flow and inlet and outlet temperatures for all affected heat exchangers during the modes of operation in which cooling water is flowing through the heat exchangers. The data collection will be performed as special test procedures, one for each of the two loops of service water and also one for HPCS service water. The tests will take the data in the flow balanced condition such that all the loads will be tested with design flow present. Initial testing will be completed by the end of outage R5 (June 1990). The tests will then be performed annually over the next three refueling cycles. Depending upon the trends observed, the periodicity of testing will be revised.

The last sentence has been revised to say:

> Depending upon the trends observed, the periodicity of testing will be revised or testing discontinued.

In response to Generic Letter 89-13 (RCC-110792-00)
This commitment was created to address the NRC's request that licensees conduct a test program to verify the heat transfer capability of all safety related heat exchangers cooled by Service Water. Specifically, this commitment verifies that fouling of the heat exchanger tubes was not progressing to the point that flow was being restricted, causing an increase in differential pressure across the heat exchanger. This type of restriction would be the result of severe fouling. CGS experiences biofouling which creates a think film of material on the tubes. Macrofouling or silting which would be the

most likely mechanism to cause a large differential pressure across a heat exchanger has not been experienced on the safety-relate heat exchangers at CGS. Macrofouling is controlled via chemical addition and silting is minimized by the design of the Service Water System and the low silt levels in the makeup water from the Columbia River. The differential pressure testing would not detect the biofouling that is typically experienced at CGS. Biofouling of safety-related heat exchangers is best detected by thermal performance testing. This testing is performed on a periodic basis on selected safety-related heat exchangers at CGS. The results are trended and monitored. There is no longer a need for differential pressure testing of heat exchangers at CGS. Therefore this commitment has been deleted. The original commitment stated:

> Concurrent with the heating exchanger cooling water flow monitoring, WNP-2 will measure, record, and trend differential pressure across each of the heat exchangers to ensure that tube fouling that could be masked by the capacity of the manual flow throttle valves used for flow balancing each of the loads is not occurring undetected.

In response to Generic Letter 89-13 (RCC-110795-00)
This commitment was created to address the NRC's request that licensees conduct a test program to verify the heat transfer capability of all safety related heat exchangers cooled by service water. The purpose of the original commitment was to verify that fouling of the heat exchanger tubes was not progressing to the point that flow was being restricted, causing an increase in differential pressure across the heat exchanger. This type of restriction would be the result of severe fouling. CGS experiences biofouling which creates a thin film of material on the tubes. Macrofouling or silting which would be the most likely mechanism to cause a large differential pressure across a heat exchanger has not been experienced on the safety-related heat exchangers at CGS. Macrofouling is controlled via chemical addition and silting is minimized by the design of the Service Water System and the low silt levels in the makeup water from the Columbia River. The differential pressure testing would not detect the biofouling that is typically experienced at CGS. Biofouling of safety-related heat exchangers is best detected by thermal performance testing. However, in the case of these small seal coolers, thermal performance testing is not practical. Therefore, RHR-HX-2A and RHR-HX-2B are inspected and cleaned on a periodic basis to ensure that they can perform their design function in support of residual heat removal (RHR) system operation.

RHR-HX-2A and RHR-HX-2B are inspected and cleaned on a four year frequency via PMIDs 10161 and 10162. RHR-HX-2A was last cleaned in 2005 and RHR-HX-2B was cleaned in 2007. There is no longer a need for differential pressure testing of heat exchangers RHR-HX-2A and RHR-HX-2B and the commitment should be changed to eliminate this differential pressure testing and the inspection and cleaning should be referenced instead. The original commitment stated:

> The small water-to-water heat exchangers that cannot be thermally tested through system operation will either be differential pressure tested as per section II.1 above or will be functionally tested per an existing technical specification surveillance. Specifically, the mechanical seal coolers for RHR pumps 2A and 2B will be differential pressure tested.

The first sentence was revised to say:

> The small water-to-water heat exchangers that cannot be thermally tested through system operation will be inspected and cleaned or will be functionally tested per an existing technical specification surveillance.

In response to Generic Letter 89-13 (RCC-110796-00)
This commitment was created to address the NRC's request that licensees conduct a test program to verify the heat transfer capability of all safety related heat exchangers cooled by service water. The purpose of the original commitment was to verify that flow across the air side of air-to-water heat exchangers (HVAC cooling coils) are free from debris which would hinder air flow and interfere with proper heat transfer. Currently CGS performs periodic inspections of the air side of the safety related heat exchangers. The coils are cleaned if necessary. CGS committed to performing an air side inspection of the coils in letter GO2-90-017. This inspection and cleaning satisfies the original concern presented in the Generic Letter 89-13 and fan differential pressure monitoring and trending is no longer necessary. Therefore the commitment can be revised to include the option to discontinue this testing based on good trending results. The original commitment stated:

> For the Air-to-Water heat exchangers (HVAC cooling coils), in addition to the cooling water flow temperature and pressure testing, WNP-2 will measure, record, and trend fan differential pressures (indicative of air flow). Initial testing will be completed by the end of outage R5. The tests will then be performed annually over the next three refueling cycles. Depending upon the trends observed, the periodicity of testing will be revised accordingly.

The last sentence has been revised to say:

> Depending upon the trends observed, the periodicity of testing will be revised accordingly or testing discontinued.

In response to Generic Letter 89-13 (RCC-110798-00)
The hydrogen recombiners referenced in the original commitment are part of the Containment Atmospheric Control System. The hydrogen recombiner function of the system has been deactivated via Engineering Change 4533. The hydrogen recombiner aftercoolers are no longer used and therefore, there is no need for inspection, cleaning, and eddy-current testing of the service water side of these aftercoolers. Therefore this commitment has been deleted. The original commitment stated:

> In addition to the cooling water flow temperature and pressure monitoring, WNP-2 will inspect, clean and eddy-current test the standby service water side of the hydrogen recombiner after coolers on a regularly scheduled basis in lieu of thermal performance testing.