ENERGY NORTHWEST

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GO2-12-014

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 2010 – 2011 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, two existing commitments have been changed.

If you have any questions or desire additional information pertaining to this report, please contact Zachary K. Dunham, Licensing Supervisor, at (509) 377-4735.

Respectfully,

BJ Sawatzke
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
WA Horin – Winston & Strawn

IE47
NMSS26
NRR
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 2010 and 2011 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.

5059-11-0001 "EC9525 REMOVE SMOKE DETECTOR POWER SUPPLY FOR ETL ACTUATION"

Brief Description

Removal of automatic remote intake isolation on fire dampers WMA-FD-1,-2 via smoke detectors WOA-SMD-1A,1B. The proposed change involves switching control room air makeup isolation upon smoke (non-plant building structure fire or range fire) from automatic to manual.

Summary of Evaluation

The removal of the automatic isolation of the remote air intake by duct-type smoke detectors WOA-SMD-1A & -1B will cause a net decrease in the likelihood of a spurious isolation of the remote air intake during a seismic event. Increasing the likelihood of having the remote air intakes available will have a positive effect on control room habitability since the smoke detectors are not seismically qualified.

The smoke detectors will still provide an alarm in the control room. The thermal links are still available for detection of hot temperatures from a nearby fire and actuate the fusible link to automatically close the fire rated dampers. If fires external to the Plant result in smoke intake, the duct smoke detectors will still alarm and the Control Room Operators can manually close the intake valves and operate the Control Room HVAC in the recirculation mode without filtration. The smoke detectors or fire dampers are not credited for any chapter 15 accident analyses and current FSAR analyses remain bounding.

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 activity 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.

This activity does not require prior NRC approval.

5059-11-0002 "EC8715 CYCLE 21 CORE DESIGN"

Brief Description

The Cycle 21 reload design is analyzed using GE Hitachi (GEH)/Global Nuclear Fuel (GNF) Methodology. The proposed change is an update to the approved GEH/GNF methodology or changes due to error corrections to computer codes associated with the approved methodology.

Summary of Evaluation

The changes identified in the approved GEH/GNF methodology as applied to the Cycle 21 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 SERs.

5059-11-0005 "EC9989 ACCEPT "AS-IS" UNDERSIZED AIR ACTUATORS SECONDARY CONTAINMENT ISOLATION VALVES FDR-V-220 AND FDR-V-222

Brief Description

The activity is changing the design of FDR-V-220 and FDR-V-222 valve actuators FDR-AO-220 and FDR-AO-222 from 33.375 inches to 29.250 inches in length, reducing the "full open" stroke length from 3.5 inches to ~70% open with a 2.1 inches stroke length.

Summary of Evaluation

These air operators were installed in the present configuration during startup and this activity establishes this condition as the design condition. The likelihood of occurrence of malfunctions will not increase over what was previously evaluated because the change is an advantageous change in stroke length for the "fail closed" failure mode of spring force to close on loss of air and the valve operator speed is not affected by a change in valve stem length. Applied spring force to close the longer stroke valve will close the shorter stroke valve, so the spring function is not adversely affected by the shorter stroke length because it is within the design stroke length of the original design. This activity does not have the potential to increase the likelihood of malfunctions in FDR-V-220 and FDR-V-222 because possible valve positions are within the full stroke capability and design for both valves. The change in valve stem travel distance and the elapsed stroke time does not change the safety function to close within stroke time limits and remain closed to meet the surveillance test criteria. The shorter stroke length keeps the disc in the flow stream, leaving the valve at approximately 70% open and minimally increasing the differential pressure across the disc. The change in stroke length does not result in more than a minimal increase in the consequence of an accident previously evaluated in the FSAR because the flow erosion of the valve gate is possible but not likely, and this erosion would not allow leakage of a quantity large enough to affect offsite dose if liquid were released because the driving differential pressure (dP) is sub-atmospheric. This activity does not have the potential to increase the consequences of a malfunction because the change in stroke length does not impact the dose calculation on evaluated malfunctions. NRC prior approval is not required for this activity.

5059-11-0006 "EC10208 DEH CONTROL SYSTEM ENHANCEMENTS FOR R20"
Brief Summary
The logic in the Control PLC (Programmable logic controller) of Digital Electro-Hydraulic (DEH) Control System will be revised to send a trip signal to the Trip PLC to trip the main turbine when both throttle valves on a single steam chest go closed. This modification will allow the Control Tricon to sense the mispositioned throttle valves and send a trip signal to the Trip Tricon to allow the protective actions to complete and place the turbine in a safe condition.

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 Turbine Trip transient (FSAR 15.2.3). The current FSAR analyses remain bounding. 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.

This activity does not require prior NRC approval.

10 CFR 72.48 Changes, Tests, and Experiments

No changes, tests or experiments were conducted during 2010 and 2011 pursuant to 10 CFR 72.48 requirements.

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-110787-00)
The change proposed is to perform inspections of the Columbia River intake structure for the plant's makeup water on a basis determined by the Preventative Maintenance Optimization Program (PMO) and past history of inspections rather than on an annual

basis. The original frequency was based on the inspection frequency already in place at the time the commitment was made. Additionally, refueling outages at the time of the commitment were on an annual frequency.

GL 89-13 Item I required licensees to implement and maintain an ongoing program of surveillance and control techniques to significantly reduce the incidence of flow blockage problems as a result of bio-fouling. A program acceptable to the NRC was included as Enclosure 1 to the generic letter. With respect to surveillance activities, Enclosure 1 recommended inspection of the intake structure once per refueling cycle for macroscopic biological fouling organisms, sediment and corrosion. Any fouling accumulations should be removed. However, for those licensees not wishing to implement the guidance in Enclosure 1, the GL does allow deviations and states, "It should be noted that Enclosure 1 is provided as guidance for an acceptable program. An equally effective program to preclude bio-fouling would also be acceptable."

Columbia currently does and will continue to inspect the intake structure of the Service Water System on an annual basis which meets the requirements of the GL. This change to the Tower Makeup (TMU) inspection frequency does not affect any current commitments of frequency or quality of inspections of the Standby Service Water System (SW).

Originally, the TMU intake structure was surveyed annually which aligned with the refueling schedule at the time. Columbia will continue to perform the underwater inspection of the TMU pump pit; however, results to date indicate there is no need to perform annual inspections or even to perform inspections every two years. Based on operating history, an inspection frequency which is based upon the PMO program is sufficient to preclude bio-fouling of the TMU and SW systems. Therefore, the requirements of the GL continue to be met with the revised frequency of the TMU inspection.

The original commitment on page 3 of Energy Northwest Letter GO2-90-017, dated February 5, 1990, stated :

> "WNP-2 will annually inspect the service water spray ponds for macroscopic biological fouling organisms in addition to the annual inspections that are already being performed on the Standby Service Water pump intake screens and at the Columbia River intake structure for the plant's makeup water. Any fouling accumulations will be removed."

The commitment has been revised to say:

> "WNP-2 will annually inspect the service water spray ponds for macroscopic biological fouling organisms in addition to the annual inspections that are already being performed on the Standby Service Water pump intake screens. Inspections at the Columbia River intake structure for the plant's makeup water will be performed on a schedule established in accordance with the Preventative Maintenance Optimization Program (PPM 1.5.13). Any fouling accumulations will be removed."

In response to Generic Letter 88-01 (RCC-212971-00)
The original commitment description:

> Energy Northwest hereby requests NRC approval to adopt the revised austenitic stainless steel piping weld inspection schedule criteria contained in Electric Power Research Institute proprietary report TR-113932 (BWRVIP-75) in lieu of our present commitments to GL 88-01.

This commitment as requested by letter G02-01-019, "Columbia Generating Station, Operating License NPF-21; Request for Adoption of BWRVIP-75 Weld Examination Schedule", dated February 9, 2001, and as approved by NRC letter GI2-01-054, "Approval of Boiling Water Reactor Vessel and Internals Project BWRVIP-75 Weld Examination Schedule for the Columbia Generating Station (TAC NO. MB1357)", dated April 9, 2001, was a revision to the original commitment to implement the inspection requirements of GL 88-01.

The commitment has been revised as follows:

> An augmented in-service inspection of all piping and components which are considered susceptible to Intergranular Stress Corrosion Cracking (IGSCC) will be performed. The program will follow the guidelines set forth in "BWR Vessel Internals Project: Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75-A)", EPRI, Palo Alto, CA and BWRVIP: 2005, 1012621.

To use BWRVIP-75 in lieu of GL 88-01 licensees had to address the open items from the original NRC Safety Evaluation (SE) to the NRC's satisfaction. This was accomplished by Columbia in the above letters G02-01-019 and GI2-01-054. BWRVIP-75-A addressed the open items from the original SE. The NRC reviewed the revised BWRVIP-75-A and found that it was acceptable (ML060760028).