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February 28, 2006
GO2-06-031

10 CFR 50.46(a)(3)(ii)
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

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397 INDEPENDENT SPENT FUEL STORAGE INSTALLATION, DOCKET NO. 72-35 2005 ANNUAL OPERATING REPORT**

Dear Sir or Madam:

Enclosed is the annual operating report for Columbia Generating Station for calendar year 2005. This report is submitted pursuant to 10 CFR 50.46, 10 CFR 50.59, 10 CFR 72.48, Regulatory Guide 1.16, Guidelines for Managing NRC Commitment Changes (NEI 99-04), and Licensee Controlled Specification 1.7.8.

If you have any questions or desire additional information pertaining to this report, please contact Mr. GV Cullen at (509) 377-6105.

Respectfully,

D.K. Atkinson

DK Atkinson
Vice President, Nuclear Generation
Mail Drop PE08

Enclosure: Columbia Generating Station 2005 Annual Operating Report

cc: BS Mallett - NRC RIV
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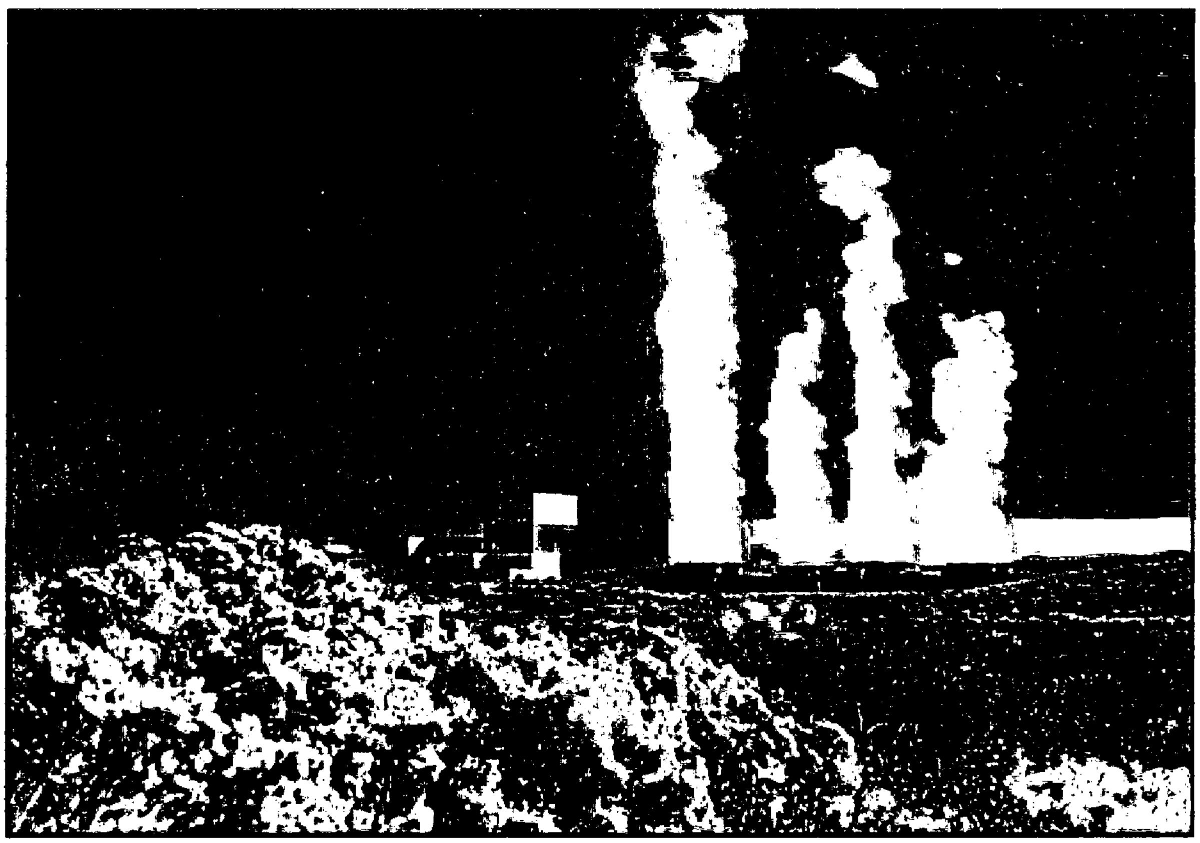
Columbia Generating Station

2005 ANNUAL OPERATING REPORT



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COLUMBIA GENERATING STATION

2005 ANNUAL OPERATING REPORT

DOCKET NO. 50-397

DOCKET NO. 72-35

FACILITY OPERATING LICENSE NO. NPF-21

Energy Northwest
P.O. Box 968
Richland, Washington 99352

Columbia Generating Station
2005 Annual Operating Report

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1.0 Reporting Requirements

The reports in this document are provided pursuant to: 1) the requirements of Licensee Controlled Specification (LCS) 1.7.8, "Sealed Source Contamination;" 2) the requirements of 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors;" 3) the requirements of 10 CFR 50.59, "Changes, Tests, and Experiments;" 4) the requirements of 10 CFR 72.48, "Changes, Tests, and Experiments;" 5) the guidance contained in Regulatory Guide 1.16, "Reporting of Operating Information-Appendix A Technical Specifications," Revision 4, August 1975; and 6) the guidance contained in NEI 99-04, "Guidelines for Managing NRC Commitment Changes," Revision 0, July 1999.

Licensee Controlled Specification 1.7.8 requires a report to be submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

Regulation 10 CFR 50.46(a)(3)(ii) requires that, in part, for each (non-significant) change to or for each error discovered in an acceptable Emergency Core Cooling System (ECCS) performance evaluation model or in the application of such a model that affects the temperature calculation, the applicant or licensee report the nature of the change or error and the estimated effect on the limiting ECCS analysis to the Commission at least annually as specified in 10 CFR 50.4.

Regulation 10 CFR 50.59(d)(2) requires that licensees submit, as specified in 10 CFR 50.4, a report containing a brief description of any changes, tests, and experiments, including a summary of the evaluation of each. This report must be submitted at intervals not to exceed 24 months.

Regulation 10 CFR 72.48(d)(2) requires that licensees submit, as specified in 10 CFR 72.4, a report containing a brief description of any changes, tests, and experiments, including a summary of the evaluation of each. This report must be submitted at intervals not to exceed 24 months.

Regulatory Guide 1.16 states that routine operating reports covering the operation of the unit during the previous calendar year should be submitted prior to March 1 of each year. Each annual operating report should include:

- A narrative summary of operating experience during the report period relating to safe operation of the facility, including safety-related maintenance not covered elsewhere.

- For each outage or forced reduction in power of over 20 percent of design power level where the reduction extends for more than four hours:

  (a) The proximate cause and the system and major component involved (if the outage or forced reduction in power involved equipment malfunction).

  (b) A brief discussion of (or reference to reports of) any reportable occurrences pertaining to the outage or power reduction.

  (c) Corrective action taken to reduce the probability of recurrence, if appropriate.

  (d) Operating time lost as a result of the outage or power reduction.

  (e) A description of major safety-related corrective maintenance performed during the outage or power reduction, including the system and component involved and identification of the critical path activity dictating the length of the outage or power reduction.

  (f) A report of any single release of radioactivity or single radiation exposure specifically associated with the outage which accounts for more than ten percent of the allowable annual values.

- A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/year and their associated man-rem exposure according to work and job functions. (Columbia Generating Station [Columbia] License Amendment 190 eliminated the requirement to report this information.)

- Indications of failed fuel resulting from irradiated fuel examinations, including eddy current tests, ultrasonic tests, or visual examinations completed during the report period.

"Guidelines for Managing NRC Commitment Changes," NEI 99-04, is an NRC-endorsed method for licensees to follow when managing or changing NRC commitments. For commitment changes that meet certain criteria, the guidance specifies that the NRC staff be notified of the changes either annually or along with Final Safety Analyses Report (FSAR) updates required by 10 CFR 50.71(e).

2.0 Summary of Plant Operations

The summary of plant operations is provided in accordance with Regulatory Guide 1.16, Revision 4, Section C.1.b.(1).

The year began with Columbia at 100% power. On May 7, the station began the scheduled refueling and maintenance outage (R-17). The outage ended 35 days later when the operators synchronized the generator to the grid early June 11. On June 15, the reactor automatically scrammed due to the closure of the turbine throttle valves. Following completion of the maintenance activities that included the replacement of digital electro-hydraulic (DEH) system cards and the replacement of the Division I standby service water pump (SW-P-1A), operators restarted the reactor and synchronized to the grid on June 22. On June 23, the reactor automatically scrammed from approximately 23% power due to loss of feedwater flow to the reactor pressure vessel. Following completion of the maintenance activities that included a modification to the RCIC system, operators restarted the reactor and synchronized to the grid on July 2. The station returned to full power on July 3, and operated continuously for the remainder of the calendar year.

Planned reductions in power were made routinely during the year for equipment maintenance, surveillance testing, and control rod manipulations.

3.0 Outages and Forced Reductions in Power

The information about the outages or forced reductions in power is provided in accordance with Regulatory Guide 1.16, Section C.1.b.(2).

April 12, 2005 (approximately 9 hours at reduced power)
On April 10, following a control rod sequence exchange and main turbine bypass valve testing, operators stopped the power ascension at 91% due to condensate feedwater heat exchanger (feedwater heater) trips. On the afternoon of April 12, the operators reduced power to 72% to facilitate the feedwater heater restoration. Full power operation was resumed April 13.

The feedwater heater trips were due to a human performance error. A procedure step was not executed properly resulting in the heater trip. Corrective actions taken included counseling personnel and revising the procedure.

May 7 - June 11, 2005 - R 17 (35 day refueling outage)
Columbia began the 17th Refueling and Maintenance Outage at 00:01 on May 7, 2005. Activities completed include replacing 28 control rod drive mechanisms, nine control rod blades, and 280 fuel bundles, installing 20 jet pump clamps, rebuilding three main steam isolation valves (MSIVs), completing the 10 year in-service inspections, replacing two rows of blades on the high pressure turbine, performing extensive transformer yard corrective maintenance, and replacing or refurbishing more than 1,000 valves. Emergent work activities completed included maintenance associated with reactor feedwater check valves, the reactor vessel steam dryer repairs, ECCS pump bolting, main generator hydrogen cooler flange gasket leak, and MSIV connector wiring.

Columbia returned to power production when the generator was synchronized to the Bonneville Power Administration (BPA) grid at 00:04 on June 11.

June 15 - 22, 2005 (approximately 174 generator off-line hours)
On the afternoon of June 15, the reactor automatically scrammed due to closure of the turbine throttle valves. **[LER-2005-003-00]** Although the trouble shooting did not identify the specific cause of the DEH system malfunction, the three circuit cards that provide control signals to the turbine throttle valves were replaced. These cards were the most likely source of DEH system malfunction.

At the time of the scram, the plant was in a 72-hour shutdown limiting condition for operation (LCO) due to the standby service water pump (SW-P-1A) degraded performance. The replacement of SW-P-1A extended the duration of the forced outage by four days.

Following reactor startup on June 21, the generator was synchronized to the grid the next evening.

The dominant root cause for the SW-P-1A degradation was the failure of the preventive maintenance (PM) program. The lack of sufficient program rigor to establish and implement adequate PM bases allowed the plant to adopt and accept a condition monitoring driven maintenance program resulting in no preventive maintenance being performed on the SW pumps. In determining the extent of the condition, Energy Northwest determined that two other standby service water pumps were susceptible to the same problems. The SW-P-1B was replaced and engineering personnel have determined that the high pressure core spray (HPCS) service water pump (HPCS-P-2) will be inspected at the next opportunity.

June 23 - July 2, 2005 (approximately 216 generator off-line hours)
On the afternoon of June 23, the reactor automatically scrammed from approximately 23% power due to a human performance error while restoring the feedwater pump low suction pressure trip. **[LER-2005-004-00]** During the scram recovery, the reactor core isolation cooling (RCIC) system tripped twice when operators attempted to restart the system for level control.

The RCIC system problem required additional system maintenance and testing and eventually a modification to the low suction pressure trip. In addition, a pre-existing oil leak on the start-up transformer was included in the outage scope. These two conditions, along with a focus on addressing human performance issues, extended the outage from two days to eight days.

Immediate actions taken in response to the human performance errors included a stand-down with Maintenance and Operations supervision, management readiness reviews of all routine and emergent work, and additional oversight of all control room and selected activities. The readiness reviews and additional oversight were terminated following plant restart. A time delay relay was added to the low suction pressure trip logic to resolve the problem encountered with the RCIC system.

On the afternoon of July 2, the generator was synchronized to the grid and full power operation was resumed on the afternoon of July 3.

August 13, 2005 (approximately 8 hours at reduced power)
On the morning of August 13, operators reduced reactor power to about 70% to recover a low pressure feedwater heater and to perform maintenance on feedwater heater 6B level control drain valve. The heaters were returned to service and full power operation was resumed that evening.

November 4 - 7, 2005 (approximately 57 hours at reduced power)
Late November 4, operators began reducing reactor power to 60% to locate and repair main condenser tube leaks. The search for the leaks was unsuccessful. The operators restored equipment to the operating configuration and returned the reactor to 100% power operation on the evening of November 7.

November 8 - 9, 2005 (approximately 32 hours at reduced power)
On the morning of November 8, operators reduced power to about 75% to perform maintenance on two condensate filter demineralizers. Maintenance was completed and full power operation was resumed on the evening of November 9.

4.0 Sealed Source Contamination

There were no incidents of sealed source contamination during 2005 that required reporting in accordance with Columbia Licensee Controlled Specification 1.7.8.

5.0 Fuel Performance

The information relative to fuel integrity is provided in compliance with Regulatory Guide 1.16, Section C.1.b.(4), and FSAR Section 4.2.4.3, "Post-Irradiation Surveillance."

Fuel Integrity
No fuel failures were identified during calendar year 2005 (Cycle 17 and 18). This conclusion was based on readings of offgas radioactivity from the pre-treatment process radiation monitoring system.

The sum-of-six readings have stayed considerably below 300 microCi/sec, the INPO threshold for fuel failures. The values for the Xe-133/Xe-135 and Xe-138/Xe-133 activity ratios have been within the range for an intact core.

Fuel Corrosion Update
Columbia did not experience any fuel defects or gross cladding anomalies during 2005. Accordingly, fuel inspections were not required by FSAR commitments. However, both SVEA-96 and ATRIUM-10 fuels were inspected in May, during the R17 outage (following the completion of Cycle 17). These inspections were in response to the Energy Northwest implementation, in recent years, of several new water chemistry programs. These programs include noble metals addition, iron and zinc injection, and hydrogen water chemistry injection. The R17 inspections indicated there was no abnormal corrosion with either the SVEA-96 fuel that was loaded for Cycles 16 and 17 or the ATRIUM-10 fuel that was loaded for Cycle 17. A SVEA-96 fuel assembly residing in the core during Cycles 15, 16 and 17 was also visually inspected. There was no indication of any further enhanced corrosion beyond that experienced by similar bundles during Cycle 15. Energy Northwest determined there were no issues with regard to the continued operation of Cycle 18 with both ATRIUM-10 and SVEA-96 fuel.

6.0 10 CFR 50.46 Changes or Errors in ECCS LOCA Analysis Models

The information relative to non-significant changes and errors in ECCS cooling performance models is provided in compliance with 10 CFR 50.46.

Westinghouse methodology was used to license SVEA-96 fuel in the Columbia core. No errors were discovered in the Westinghouse ECCS loss of coolant accident (LOCA) analysis model and no revisions were made to the Columbia LOCA Analysis Report during 2005.

AREVA used the Framatome-ANP (FANP) methodology to license ATRIUM-10 fuel in the Columbia core. Energy Northwest reported the errors in the application of the approved methodology to the NRC in Reference 1. The report included the estimated impact of the errors on peak cladding temperature (PCT) for Cycle 17 for single-loop and two-loop recirculation pump operation scenarios.

The report also included the estimated impacts on Cycle 18 PCT due to the error in the lower plenum flow calculation. The new Cycle 18 analysis, completed in June 2005 and documented in Reference 2, corrected the error and confirmed the final PCTs for both single-loop and two-loop operations reported in Reference 1.

Since June, no errors were discovered in the AREVA ECCS loss of coolant accident (LOCA) analysis model and no revisions were made to the Columbia LOCA Analysis Report.

References

1. Letter, Energy Northwest to NRC, GO2-05-108, "Report of Significant Changes in Cycle 17 and 18 Emergency Core Cooling System (ECCS) Evaluation Analysis," June 16, 2005.

2. AREVA Report EMF-3172(P), Revision 1, "Columbia Generating Station LOCA-ECCS Analysis MAPLHGR Limit for ATRIUM™-10 Fuel," June 2005 (Proprietary).

7.0 10 CFR 50.59 Changes, Tests, and Experiments

This section contains the summary of the evaluations for activities implemented during 2005 that were assessed pursuant to 10 CFR 50.59 requirements.

Energy Northwest implemented the revised 10 CFR 50.59 rule in August 2001. Seven evaluations were performed for activities implemented in 2005, under the revised rule. One change, implemented in 2005, was evaluated under the old rule, in 2000. Accordingly, the term *unreviewed safety question* still applies to the evaluation that was approved under the old rule.

Energy Northwest evaluated each change summarized in the following sections and determined that no change required NRC approval, represented an unreviewed safety question, or required a change to the Technical Specifications.

PLANT MODIFICATION RECORD 93-0037-17s (Safety Evaluation 00-0023)
This plant modification provided for the installation of permanent platforms and step ladders in the day tank rooms for each of the emergency diesel generators (DG). In 2005, the platforms and step ladders for the DG-1 and DG-2 rooms were installed to complete this project. The platforms replaced temporary scaffolding installed to facilitate access to the day tank measuring ports.

<u>Evaluation Summary</u>
The installed platforms and step ladders perform a passive function of allowing access to the measuring ports. The platforms and step ladders were designed and installed in accordance with seismic requirements to ensure that they would not have any adverse impacts or interactions with important to safety structures,

systems, or components (SSC). The evaluation concluded that the plant modification did not represent an *unreviewed safety question*.

FSAR CHANGE LDCN-FSAR-05-005 (Evaluation 5059-05-0001) A new methodology was used to calculate the decay heat of irradiated fuel bundles in the spent fuel pool.

Evaluation Summary
The ORIGEN-ARP methodology will be used in lieu of the Branch Technical Position ASB 9-2 and the original version of the ORIGEN computer code that were used to calculate decay heat loads in the spent fuel pool. The NRC accepted the use of ORIGEN2 for calculating decay heat in spent fuel pool applications through issuance of a Safety Evaluation Report (SER). Additionally, Energy Northwest determined that ORIGEN-ARP provides results that are essentially the same as ORIGEN2. The SER restriction to include a 2% uncertainty factor in power measurement has been inherently included in the error terms derived for ORIGEN-ARP. The 50.59 evaluation concluded that the change in methodology did not constitute a departure from a method of evaluation currently described in the FSAR that required prior NRC approval.

Subsequent to implementing the change in the methodology during R17, the NRC issued a Severity Level IV non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000397/2005009-002, Failure to Obtain Prior Approval Prior to Implementing a New Methodology. Energy Northwest will seek NRC approval for the change in the methodology to ORIGEN-ARP to calculate the decay heat loads.

FSAR CHANGE LDCN-FSAR-05-005 (Evaluation 5059-05-0004) The evaluation associated with this portion of the FSAR change allowed administrative controls for establishing temperature limits for reactor building closed cooling water (RCC) and SW.

Evaluation Summary
The allowed heat load in the spent fuel pool was increased above previous values. During normal operations, including assumed LOCA initial conditions, the heat load is limited to approximately 10.1 million BTU/hr. Administrative controls will be used to establish temperature limits on the RCC and SW cooling supply to the fuel pool cooling (FPC) heat exchangers. Administrative controls will also be used to allow transfer of fuel bundles to the spent fuel pool 24 hours after shutdown as long as the times and temperatures are supported by analysis. This will allow startup following a refueling outage with sole reliance on SW cooling until a time when the decay heat load has dropped to a level that can be accommodated by RCC, while maintaining spent fuel pool temperatures within the acceptance criteria (such as long term reactor building temperature and humidity response and equipment qualifications). Timing of operator actions to

align SW to FPC heat exchangers and align emergency makeup of SW to the spent fuel pool are not adversely affected. An increase in cooling water temperature above the administrative limit will not make an equipment malfunction more likely since actions can be taken to mitigate the increase prior to the spent fuel pool temperature exceeding the applicable acceptance criterion. Operating SW in lieu of RCC does not represent more than a minimal decrease in system reliability since SW was designed and licensed as the long-term cooling source to the FPC system following a loss of the RCC system. Energy Northwest determined prior NRC approval was not required.

PLANT DESIGN CHANGE 0000002074 (Evaluation 5059-04-0005) This change replaced the thermal dispersion level switches in the HPCS system pump suction line from the condensate storage and transfer (CST) tanks (HPCS-LS-3A and 3B) with pressure switches and time delay relays. The switches transfer the suction source from the CST tanks to the suppression pool in response to a CST pipe break.

<u>Evaluation Summary</u>
The replacement of the thermal dispersion type level switches with pressure switches and time delay relays will continue to accomplish the function of detection and isolation of a CST tank suction pipe break. The suction transfer logic from the CST tanks to the suppression pool remains the same. No accidents or malfunctions with different results are created by this design change. Therefore, Energy Northwest determined that prior NRC approval was not required.

PLANT DESIGN CHANGE 0000003719 (Evaluation 5059-05-0007) This change allowed use of wireless communication devices with administrative control in the main control room and the radwaste control room.

<u>Evaluation Summary</u>
The basis for acceptance of wireless portable transceivers in nuclear power plants is contained in Regulatory Guide 1.180 and other industry standards and topical references. Inherent in this basis are attributes of electromagnetic compatibility (EMC) of plant equipment. Attributes of EMC when evaluated to the existing plant layout and emissions background, to new communication equipment selected and applied with the use of administrative controls, contributes to an overall assurance that safety related structures, systems and components are designed to accommodate the effects of and are applied so they may be compatible with the environmental conditions associated with nuclear plant service conditions. Establishing electromagnetic compatibility involves (1) assessment of plant emissions background and equipment susceptibility, (2) plant design, equipment specification and installation, along with (3) administrative controls, operator practice and training all used to limit impacts of electromagnetic effects on plant instrument and control systems. Based on

analysis of site background emissions at a minimum exclusion distance, the wireless communications device maximum radiated power level selected are conservative when compared to the Regulatory Guide 1.180 recommended field strength limit for portable transceivers. The instrument grounding system provides added assurance that equipment in the plant remains in accordance with EMI/RFI control practices and ensures the use of these wireless communication devices will not physically alter any equipment, system performance, or operator actions and the current FSAR analyses remain bounding. Therefore, Energy Northwest determined that prior NRC approval was not required.

FSAR CHANGE LDCN-FSAR-04-010 and LCS CHANGE LDCN-LCS-04-007
(Evaluation 5059-05-0006, Revision 1)
The refuel bridge monorail hoist and auxiliary hoist overload (HOL) cutoff setpoints were raised from the 535 lbs maximum to 975 lbs maximum. The increased HOL cutoff setpoints allow these auxiliary hoists to support in-vessel maintenance, including control rod blade exchange using the Multi Lift Tool and jet pump mixer removal and replacement.

<u>Evaluation Summary</u>
The evaluation determined that the increase in the refuel bridge HOL cutoff setpoints from 535 lbs (maximum) to 975 lbs (maximum) is within the design functions of those auxiliary hoists as currently described by the FSAR and is bounded by accident analyses presented in the FSAR. The change is not adverse to the safe operation of Columbia, to the public, or to the environment, when used as described for in-vessel maintenance. Therefore, Energy Northwest determined that prior NRC approval was not required.

LCS CHANGE LDCN-LCS-02-004 (Evaluation 5059-05-0002)
The interval for the channel calibration of the main steam safety relief valve (SRV) position indication and the turbine overspeed protection, pursuant to LCS Surveillance Requirements (SR) 1.3.3.1.2 and SR 1.3.7.6.2, were extended from 18 to 24 months to accommodate a 24-month fuel cycle. The snubber functional test interval, as specified in the Augmented Inservice Inspection and Testing Program (LCS Bases SR 1.7.3.1) was also extended from 18 to 24 months.

The SRV position indication provides control room operators valve status information during accident situations. The SRV position indication is classified as Regulatory Guide 1.97, Category 2, Type D.

The purpose of the Turbine Overspeed Protection System is to prevent an overspeed condition which could lead to the destruction of turbine components and generation of high-speed missiles that could damage safety related components, equipment or structures.

Snubbers are passive devices used for supporting piping systems. The safety related snubbers ensure the structural integrity of safety related systems is maintained during and following a seismic or other event that initiates dynamic loads.

The evaluations of the extension of the SR intervals were performed using the guidance provided in Generic Letter (GL) 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-month Fuel Cycle." The GL 91-04 assigns three criteria that must be met in order to extend the SR interval.

Evaluation Summary
No postulated accidents, transients, or malfunctions are impacted by the increase in the surveillance test intervals. Therefore, the change will not increase the frequency or probability of occurrence of an accident or malfunction. The change did not affect the capability of any important to safety SSC to perform the intended safety functions. The change does not create the potential for a new type of event or adverse impact on the fission product boundaries. The change has no impact on the methodologies used to establish the design bases for the station. Therefore, Energy Northwest determined that prior NRC approval was not required.

LCS CHANGE LDCN-LCS-02-019 (Evaluation 5059-04-0010)
The interval for performance of the system functional test of the protective relays (primary and backup) for the 6.9 kV reactor coolant recirculation (RRC) pumps pursuant to LCS SR 1.8.10.2 was extended from 18 to 24 months. The relays (primary and backup) protect the primary containment electrical penetrations for the 6.9-kV RRC pumps. The RRC pumps must be de-energized for the primary overcurrent protection relay channel calibration. To schedule the system functional test during refueling outages, the interval was changed to 24 months. The backup overcurrent protection relay functional test can be performed while the 6.9-kV RRC circuits are energized. However, to maintain consistency with the primary relays, the backup relay interval was also extended.

These protective relays limit the fault current heating experienced by the penetration conductors to levels below the conductor ratings thereby preventing a challenge to primary containment integrity. The periodic system functional surveillance requirement for the 6.9-kV overcurrent relays demonstrates the operability of the primary and backup overcurrent protective relays. Although these overcurrent protective devices are not considered to function in any design basis accident or transient, evaluation of the extension of the SR interval was performed using the guidance provided in Generic Letter (GL) 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-month Fuel Cycle." The GL 91-04 assigns three criteria that must be met in order to extend the SR interval.

Evaluation Summary
No postulated accidents, transients, or malfunctions are impacted by the increase in the surveillance test interval. Therefore, the change will not increase the frequency or probability of occurrence of an accident or malfunction. The change did not affect the capability of any important to safety SSC to perform the intended safety functions. The change does not create the potential for a new type of event or adverse impact on the fission product boundaries. The change has no impact on the methodologies used to establish the design bases for the station.

8.0 10 CFR 72.48 Changes, Tests, and Experiments

There were no activities implemented during 2005 that required reporting pursuant to 10 CFR 72.48 requirements.

9.0 Regulatory Commitment Changes (NEI 99-04 Process)

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

Non-Safety Related Instrument Air (RCC-216797-00)
In response to NCV 50-397/03-07-01, Energy Northwest discussed plans for local leak rate testing (LLRT) of the main steam isolation valves (MSIV). In that letter to the NRC, Energy Northwest stated,

> "Energy Northwest is currently planning to conduct MSIV LLRT testing with only safety related air sources providing seating pressure to the valves during our next refueling outage."
> (Reference letter GO2-04-072, dated April 16, 2004, RL Webring to NRC)

The new commitment is:

> Energy Northwest is currently planning to conduct future MSIV LLRT testing with only safety related air sources providing seating pressure to the outboard MSIVs, and that non-safety related air will be re-established to the inboard MSIVs after valve closure. The current MSIV LLRT test plan may be modified prior to the next refueling outage as a result of action taken by the BWROG MSIV Testing Committee.

The commitment change clarifies the statement about using non-safety related air and provides additional details regarding impact of the BWROG (Boiling Water Reactor Owners Group) MSIV Committee actions on future testing plans.

As written, the original commitment (GO2-04-072) implies that non-safety related air would not be used when testing all MSIVs. The letter failed to mention that during an LLRT, the inboard MSIV accumulator pressure must be increased above 63 psig (after valve closure pressure) to offset the opening force created by test pressure under the inboard valve disc. More than 44 psig is needed to offset a 25 psig line test pressure. Based on this, Columbia MSIV LLRT testing procedures will specify that once the inboard MSIV has been closed, non-safety related air would be re-established to the accumulator prior to the LLRT.

This revision also adds that the testing commitment is subject to change based on the BWROG MSIV Testing committee actions.