ENERGY NORTHWEST

P.O. Box 968 ■ Richland, Washington 99352-0968

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Docket No. 50-397

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Subject: **COLUMBIA GENERATING STATION, OPERATING LICENSE NPF-21 ANNUAL OPERATING REPORT 2000**

The annual operating report for calendar year 2000 is attached. If you have any questions or desire additional information pertaining to this report, please contact either me or PJ Inserra at (509) 377-4147.

Respectfully,

RL Webring
Vice President, Operations Support/PIO
Mail Drop PE08

Attachment

cc: EW Merschoff, NRC - Region IV
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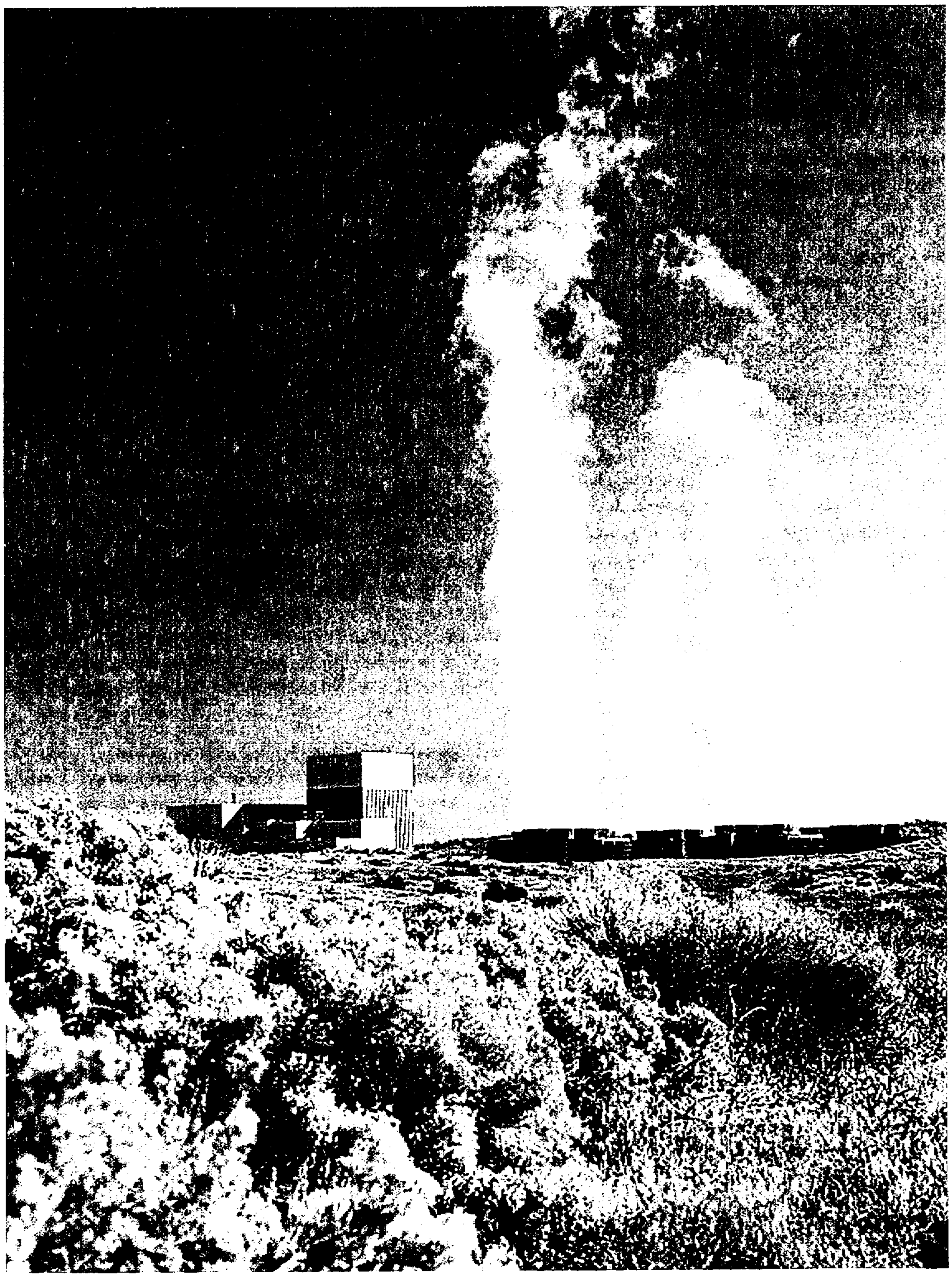
A001

Columbia Generating Station
Annual Operating Report
2000



COLUMBIA GENERATING STATION

ANNUAL OPERATING REPORT

2000

DOCKET NO. 50-397

FACILITY OPERATING LICENSE NO. NPF-21

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

Columbia Generating Station
2000 Annual Operating Report

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1.0 Introduction

The 2000 Annual Operating Report for Energy Northwest's Columbia Generating Station is submitted pursuant to Federal Regulations and Facility Operating License NPF-21. The plant is a 3486 MWt, boiling water reactor (BWR-5) that began commercial operation on December 13, 1984.

On June 26, 2000, following an operational run of 264 days, the plant automatically scrammed due to a trip of the main generator. The unit trip was caused by a grounded conductor in the current transformer circuit at main transformer E-TR-M3. The plant was returned to full power operation on July 6, 2000.

The plant remained on-line for 59 days until September 1, 2000, when the plant was taken off-line for a planned forced outage to replace the upper seal on reactor recirculation system pump RRC-P-1A. Following seal repair, the plant was restarted and full power operation was achieved on September 8, 2000.

On September 18, 2000, after 12 days on-line, the plant was manually scrammed in response to increasing main condenser backpressure. The cause of the increasing backpressure was attributed to a failed weld in a drain line inside the condenser. Repairs were made and the plant was restarted, returning to full power on September 23, 2000. The plant remained on-line for the remainder of the year.

Several radiation protection goals were achieved during calendar year 2000. Station dose achieved a record low of 48.04 rem, personnel contaminations were 55, and contaminated floor space was reduced to 1.62 percent of the accessible area in the station being contaminated.

Year 2000 was also the best generating calendar year in the history of Columbia Generating Station. The station produced 8,605,232 megawatt-hours of generation, plus had 621,256 megawatt-hours in economic dispatch. The next best year was 1997, where there were 6.1 million megawatt-hours of generation and 1.65 million megawatt-hours in economic dispatch.

2.0 Reports

The reports in this section are provided pursuant to: 1) the requirements of Technical Specification 5.6.1, "Occupational Radiation Exposure Report," 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 guidance contained in Regulatory Guide 1.16, "Reporting of Operating Information," Revision 4 - August 1975, and 5) the guidance contained in the NEI 99-04, "Guidelines for Managing NRC Commitment Changes," Revision 0, July 1999.

Technical Specification 5.6.1 requires that the following report be submitted in accordance with 10 CFR 50.4 by April 30 of each year:

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) for whom monitoring was performed, receiving an annual deep dose equivalent of greater than 100 mrems and the associated collective deep dose equivalent (reported in man-rem) according to work and job functions [e.g., reactor operations surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling]. This tabulation supplements the requirements of 10 CFR 20.2206. The dose assessments to various duty functions may be estimated based on electronic or pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totaling less than 20 percent of the individual total dose need not be accounted for. In aggregate, at least 80 percent of the whole body dose received from external sources should be assigned to specific major work functions.

Regulation 10 CFR 50.46 requires that, for each (non-significant) change to or error discovered in an acceptable Emergency Core Cooling System (ECCS) cooling performance evaluation model or in the application of such a model that affects the temperature calculation, the applicant or licensee shall report the nature of the change or error and its 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 requires that licensees submit, as specified in 10 CFR 50.4, a report containing a brief description of any changes, tests, or experiments, including a summary of the safety evaluation of each. The report may be submitted annually or at shorter intervals.

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 the 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 (or reference to reports) of any reportable occurrences pertaining to the outage or 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 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 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.

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

The *NEI Guidelines for Managing NRC Commitment Changes* is a commission-endorsed method for licensees to follow for managing or changing NRC commitments. As part of this process and for commitments that satisfy the NEI decision criteria, the guidance specifies periodic staff notification, either annually or along with the FSAR updates as required by 10 CFR 50.71(e).

2.1 Summary of Plant Operations

This section contains a narrative summary of operating experience and is included pursuant to Regulatory Guide 1.16, Sections C.1.b.(1) and C.1.b.(2).

January 2000

- The station began the month at 80% power as part of the station's plan for Y2K readiness. Full power operation resumed on the morning of January 1, 2000.
- On January 22, 2000, a down power to 77% was initiated for control rod pattern changes, front standard testing, bypass valve testing, and turbine valve surveillances. The station returned to full power the same day.

February 2000

- The station operated at full power during the month of February. Power reductions of less than 20% were conducted for scheduled testing and rod pattern changes.

March 2000

- The station operated at full power during the month of March, excluding down powers for the purposes of regional power management at the request of the Bonneville Power Administration (BPA).

April 2000

- The station began the month at 80% power. Power levels throughout the month ranged from 60% to 100% at the request of BPA for regional power management.

May 2000

- At the beginning of the month the station was operating at 60% power. On May 12, 2000, a planned down power from 60% to 20% was initiated to repair steam leaks. Power ascension was initiated on May 14, 2000.
- Power ascension was halted at 62% after discovery of the closure of main steam isolation valve MS-V-28A on May 15, 2000. A down power was commenced to about 42% to make repairs and reopen the valve. Power ascension to 70% was initiated on May 17, 2000.
- On May 31, 2000, power was reduced to 70% for a control rod pattern change followed by an increase to 90% power at the request of BPA for regional power management.

June 2000

- At the beginning of the month the station was operating at 90% power returning to full power on June 1, 2000. On June 9, 2000, a power reduction was initiated to 80% for scheduled testing. Power was returned to 100% on June 10, 2000.

- On June 16, 2000, power was reduced to 74% for investigation and repair of main steam governor valve MS-V-GV/1. During this evolution, power was raised and lowered several times between the levels of 72% and 90% and returned to 100% on June 18, 2000.

- On June 23, 2000, power reduction was initiated to 73% for scheduled testing. Power was returned to 100% on June 25, 2000.

- On June 26, 2000, Columbia Generating Station experienced a trip of the main generator, a turbine trip, and reactor scram. The shutdown was caused by a grounded conductor in the current transformer circuit at main transformer E-TR-M3, which resulted in an erroneous indication of phase current mismatch in the electrical output from the station. (Reference LER 2000-003-00.)

July 2000

- At the beginning of the month the plant was shutdown for repairs to the main generator phase current monitoring systems. Following repairs the plant restarted and reached full power on July 6, 2000.

- The station initiated a down power to 68% in response to drift of control rod CRD-ROD-2203 on July 18, 2000. Full power operation was resumed on July 19, 2000.

- A down power to 80% was initiated on July 28, 2000, for main turbine front standard trip testing. Full power operation was resumed on July 29, 2000.

August 2000

- At the beginning of the month the station was operating at full power. On August 12, 2000, an alarm was received indicating seal failure on reactor recirculation pump RRC-P-1A. On August 15, 2000, a power reduction was initiated to 84% for a control rod adjustment in preparation for the possibility of a pump trip. Power was returned to 100%. On August 16, 2000, a power reduction was initiated for the purpose of removing RRC-P-1A from service and entry into single loop operation.

September 2000

- On September 1, 2000, the plant was shutdown for the purpose of repairing the pump seal on RRC-P-1A.
- After repairs, the plant was restarted and reached full power on September 8, 2000.
- On September 18, 2000, power was reduced to 90% due to a slow loss of condenser vacuum. The plant was manually scrammed due to a loss of condenser vacuum. The problem was traced to a failed weld in a drain line inside the condenser. After repairs, the plant was restarted and reached full power on September 23, 2000. (Reference LER 2000-007-00.)

October 2000

- At the beginning of the month the station was operating at full power. Other than down powers of less than 20% for scheduled testing and rod pattern changes, the station remained at full power for the remainder of the month.

November 2000

- At the beginning of the month the station was operating at full power. Other than down powers of less than 20% for scheduled testing and rod pattern changes, the station remained at full power for the remainder of the month.

December 2000

- At the beginning of the month the station was operating at full power. The station commenced a planned down power to 78% on December 15, 2000, for rod sequence exchange and front standard testing. The reactor was restored to full power operation on December 16, 2000.
- A planned down power to 75% was initiated on December 22, 2000, for bypass valve and turbine valve testing and a steam tunnel entry for work on main steam valve MS-V-20. Full power operation resumed on December 23, 2000.
- A planned down power to 72% was initiated on December 29, 2000, for scram time testing. Full power operation was resumed on December 30, 2000, and was maintained at that level for the remainder of the month.

2.2 Significant Maintenance Performed on Safety-Related Equipment

This section contains brief descriptions of major, safety-related maintenance performed during outages or power reductions and is included pursuant to Regulatory Guide 1.16, Section C.1.b(2)(e).

Reactor Core Isolation Cooling System

Reactor core isolation cooling system inverter RCIC-IN-2R was replaced following failure to meet output voltage requirements during acceptance testing.

Reactor Recirculation System

The upper seal was replaced on reactor recirculation system pump RRC-P-1A following indications of seal failure.

Reactor Water Cleanup System

Reactor water cleanup system flow indicator RWCU-FI-602 was replaced following the discovery of a variance in blowdown flow indication during surveillance testing.

2.3 Radiation Exposure

The annual work and job function report is included as Appendix A and contains information pertaining to personnel radiation exposure. This information is included pursuant to Technical Specifications 5.6.1 and Regulatory Guide 1.16, Section C.1.b.(3).

The values are estimated doses for the listed activities and are based on direct reading dosimeter data. No correction factor was applied to the readings.

2.4 Fuel Performance

This section contains information relative to fuel integrity. There was evidence of a fuel defect during the calendar year 2000 portion of Cycle 15.

On November 28, 2000, offgas system sampling data indicated increased fission gas activities. The sum-of-six (S6) reading increased from 80 μCi/sec to 136 μCi/sec. Further analysis of fission gas isotopes (Xenon and Krypton) indicated a small fission gas leak had occurred between November 21, 2000 and November 28, 2000. Precautionary plant power maneuvering guidelines were placed in effect to suppress rapid nodal and global power changes, and offgas sampling frequency was increased.

2.5 10CFR50.46, Changes or Errors in ECCS LOCA Analysis Models

This section contains information relative to changes and errors in Emergency Core Cooling System (ECCS) cooling performance models.

Included in this section is a description of the impact of any non-significant changes and errors discovered in the ECCS cooling performance evaluation models or in the application of such a model where the change or error was determined to be non-significant. For the purposes of this report, non-significant errors are those that are less than or equal to 50 degrees Fahrenheit. (Significant errors are reported pursuant to 10 CFR 50.72 and 10 CFR 50.73 and are not included in this report.)

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

Both General Electric (GE) and Westinghouse (previously ABB) methodologies are applied to the Columbia Generating Station core. The GE methodology was used to license Siemens Power Corporation (SPC) fuel. This is the LOCA analysis of record for Columbia Generating Station.[1] The Westinghouse methodology was used to license Westinghouse SVEA-96 fuel.[2]

For 2000, there was one change in the ECCS LOCA evaluation model or application of the model for the SPC fuel and one error in the plant specific application of the model. Regarding the error, the ECCS piping inside the vessel (between the vessel wall and shroud) has various leakage paths through slip joints and vent holes. Not all of the ECCS water injected into the vessel reaches the region inside the shroud. Some of the water is lost through these leakage paths into the downcomer region. The core spray and LPCI flow rates must then be adjusted for use in the SAFER analysis to account for the leakage inside the vessel.

An evaluation was performed to determine the impact of the ECCS leakage on the peak cladding temperature (PCT). The PCT impact of the leakage is an increase of 5°F for Columbia Generating Station.

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[1] General Electric Report NEDC-32115P, Revision 2, "Washington Public Power Supply System Nuclear Project 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," GE Nuclear Energy, July 1993

[2] Westinghouse Report CE NPSD-801-P, Revision 1, "WNP-2 LOCA Analysis Report," Westinghouse Combustion Engineering Nuclear Operations, June 1996

Regarding the change in the application of the acceptable evaluation model, in response to a concern raised in a Boiling Water Reactor Owners Group assessment of the SAFER LOCA analysis process and methodology, an evaluation was performed to determine the impact of the time step size on LOCA calculations performed with SAFER. Based on the results of this study, smaller hydraulic and heat conduction time step sizes were recommended in the SAFER analyses for all plant types. The change resulted in a –5°F impact on the PCT for Columbia Generating Station.

For the Westinghouse fuel, there were no errors in the ECCS LOCA analysis model or application of the model for 2000.

2.6 10CFR50.59, "Changes, Tests and Experiments"

This section contains summaries of the Safety Evaluations (SE) completed for activities implemented during 2000 and is included pursuant to 10 CFR 50.59.

Regulation 10 CFR 50.59 and Columbia Generating Station Operating License NPF-21 allow changes to be made without prior NRC approval, unless the proposed change, test, or experiment involves a change in the Technical Specifications incorporated in the license or an unreviewed safety question.

Each change summarized in the following sections was evaluated and determined not to represent an unreviewed safety question or require a change to the Technical Specifications.

In certain instances, a single safety evaluation was used for several implementing activities which were within the scope of the proposed change. This is allowed by procedure only where an existing evaluation adequately covers the specific change being considered. A separate evaluation is also required if out-of-service equipment, equipment lineups, modifications, or temporary alterations are in place that invalidate the existing evaluation.

2.6.1 Plant Modifications

This section contains information pertaining to implemented Plant Modification Records (PMRs), Technical Evaluation Requests (TERs), and Temporary Modification Requests (TMRs) and is included pursuant to 10 CFR 50.59.

PMR 94-0084-0 (SE 98-011-01)

This modification provided for re-routing of piping and the installation of pumps to accommodate the addition of sodium bromide for the treatment of both the circulating water system and plant service water to improve water quality treatment and prevent further corrosion in piping. Revision 1 to this safety evaluation included a modification to FSAR Section 6.4.4.2.3 to specify the location of one 5000-gallon tank of sodium hypochlorite and one 5000-gallon tank of sodium bromide.

Safety Evaluation Summary

The safety evaluation concluded that a 5000-gallon spill of sodium bromide does not pose a hazard to control room habitability. Since control room habitability is not affected, all accidents analyzed that involve operator action or even delayed operator action remain unchanged.

PMR 96-0198-0A (SE 99-0057)

This modification permanently deactivates the standby service water (SSW) keepfull subsystem. The intended function of the SSW keepfull subsystem was originally non-safety related, which was to provide backpressure on the pump discharge valve during the start sequence and to minimize wear and cavitation that would occur upon dry startup of the pumps. This was determined to no longer be necessary as a result of an engineering analysis.

Safety Evaluation Summary

The safety evaluation concluded that the components associated with this design change have no connections to Class 1 electrical power and no functional interfaces with Quality Class 1 or 2+ components. Additionally, the piping and support system remains unchanged. The SSW keepfull subsystem does not have any safety-related function and the modification will not initiate any FSAR design basis accident or transient events. The change will also not affect the ability of the SSW to perform its design safety function.

PMR 98-0134-0-01 (SE 99-0063)

This design change (field change request) installed a supplemental forced air electric unit heater in the high pressure core spray system (HPCS) diesel generator room.

Safety Evaluation Summary

The safety evaluation concluded that installation of the heater ensures that the HPCS battery capacity will not be diminished during periods of cold weather, thereby, ensuring the HPCS system remains functional during these periods. This change does not change the function of the HPCS battery, nor does it alter any HPCS system or any other safety-related system functions. The heater will only be used in periods of cold weather as a "keep ready" function and it is not required in the event of a design basis accident with the HPCS system and diesel generator in operation.

PMR 97-0147-0 (SE 00-0011)

This modification installed a corrosion and biofouling monitoring coupon station as a subsystem to the plant service water (TSW) return line piping and is intended to monitor the water chemistry of the TSW system.

Safety Evaluation Summary

The safety evaluation concluded that the new coupon station is non-safety related and has no functional or physical interface with any quality Quality Class 1, 2+, Seismic I, or Seismic II/1 system, structure, or component. The change will not initiate any FSAR design basis accident or transient events and will not cause a malfunction of any safety-related equipment in the plant.

PMR 96-0153-0 (SE 00-0022)

This modification removed the seal-in contacts from main steam system valves MS-V-108A, 108B, 105A, 105B, BS-V-22A, V-22B, V-17A, and V-17B. None of these valves have an automatic open or close function and were throttled with operator work arounds by opening their respective disconnect switches at the local motor control center.

Safety Evaluation Summary

The safety evaluation concluded that removal of the seal-in contacts does not result in an unreviewed safety question. Turbine gland seal exhaust and condenser exhaust contributes essentially no airborne radioactivity releases due to the use of clean steam in the system and this change does not alter the source term available for release. No accident or malfunction more limiting than the complete system malfunction already assumed in the FSAR is introduced by this activity. The turbine gland seal steam system and feedwater system is non-safety related and does not support any safety function.

PMR 98-0081-0 (SE 00-0045)

This modification replaced the Division 1 and 2 diesel generator Woodward 24VDC power supplies with Lambda 24 VDC power supplies rated at an output current of 2.1 amps. This modification was required because the Woodward 24 VDC power supply, which feeds the speed sensing control circuit in the generator control panel, was undersized.

Safety Evaluation Summary

The safety evaluation concluded that the new Lambda 24 VDC power supplies are safety-related and were qualified and mounted to Seismic Category 1 requirements. Electrical connections to the new power supplies were made to Quality Class 1 requirements using approved plant procedures. The replaced power supplies have the same input/output voltage ratings but have an increased output current rating. This increased output current rating brings the power supply into compliance with the requirements of the design to provide reliable 24 VDC control circuit power when loaded to 0.75 amps, and also provides an adequate design margin for the power supply. The function of the speed sensing circuits, the output breakers, the diesel generators, and the safety-related buses remained unchanged. Therefore, implementation of this activity did not increase the probability of occurrence of an accident evaluated previously in the FSAR. The changes made by this modification do not constitute an unreviewed safety question.

TER 98-0092-0 (SE 98-108)

This modification provided for an upgrade to the entrance of the plant. Specifically, the southwest single door was blocked by a security gun post and the northwest single door was removed and replaced by a double glass door with a window on each side.

Safety Evaluation Summary

The safety evaluation concluded that elimination of one single exit door and the enlargement of the second single door does not impact any previously evaluated accident analysis. The doors are not addressed in any accident previously evaluated in the FSAR and they have no impact on equipment important to safety.

TER 99-0033-0 (SE 99-0018)

This modification added a pressure gauge and isolation valve on the condenser discharge line for the radwaste building chillers.

Safety Evaluation Summary

The safety evaluation concluded that the installation of a pressure gauge and isolation valve on the condenser discharge line does not pose a hazard to the safe operation of the plant and does not impair the operators' ability to shut down the reactor and maintain it in a safe condition. The addition of these instruments is for improved performance monitoring capabilities.

TER 99-0068-0 (SE 99-0026)

This modification replaced the existing (electronically activated) proximity switch circuit on the traversing incore probe system with a magnetically operated switch circuit.

Safety Evaluation Summary

The safety evaluation concluded that replacing the electronically activated proximity switch with a magnetically operated switch circuit has no adverse effect on the operation of the system or any accident associated with the system. The new switch circuit is functionally equivalent and a qualitatively more reliable design.

TER 00125401 (SE 00-0012)

This modification revised drawing M510-3 to show service air system valve SA-V-120 as "normally open." This manual isolation valve supplies air to the radioactive waste processing (EDR) demineralizer. Air is used during the EDR backwash cycle to remove depleted powdered resin from the EDR demineralizer septa.

Safety Evaluation Summary

The safety evaluation concluded that operation with SA-V-120 in the "normally open" position reduces the impact on operation and results in consistent operation of the EDR and floor drains radioactive waste processing systems. The change has no adverse impact on the previously evaluated function of the EDR waste processing systems. The FSAR does not consider any design basis accidents or anticipated operational occurrences involving the service air supply to the EDR waste processing system.

TER 99-0168-0 (SE 00-0032)

This modification added a new self-indicating flow indicator to the corrosion monitoring coupon station. This indicator will allow personnel to monitor the rate of turbine service water flow through the corrosion coupon station piping. Additionally, a new ¾-inch bypass line around the existing system pressure switch PS-RO-3 was added that includes a normally closed manual valve. This bypass will allow the coupon station piping to be periodically velocity flushed to preclude excessive silt accumulation in the piping.

Safety Evaluation Summary

The safety evaluation concluded that the new components installed do not alter the function of the coupon station, nor do they create any new interfaces with other plant equipment. This modification will not initiate any FSAR-evaluated design basis accidents or transients, nor will the modification impact any safety system, structure, or component.

TER 00134001 (SE 00-0033)

This modification provided for the replacement of reactor feedwater turbine seal steam relief valves with a 2 ½-inch discharge rather than a 2-inch discharge.

Safety Evaluation Summary

The safety evaluation concluded that the function of the seal steam relief valve to prevent excessive seal steam pressure is not affected by changing the discharge of the relief valves from 2 inches to 2 ½ inches. The pressure rating of the valves, piping, and flanges is unchanged. Because the piping pressure rating and performance function of the valves is not affected, there are no new accident scenarios introduced by the change. All plant systems and components required to mitigate the consequences of accidents previously evaluated are also not affected by this modification.

TMR 00-008 (SE 00-0036)

This modification installed a temporary dew point monitor (hygrometer) in place of CAS-ME-10 in order to locally monitor dew point in the control air (CAS) system while CAS-ME-10 and the associated indicator, CAS-MIS-10, are out of service for maintenance. Moisture element CAS-ME-10 and the associated instrument, CAS-MIS-10, monitor dew point in the CAS system. A small volume of air passes through the monitor then is discharged to atmosphere.

Safety Evaluation Summary

The safety evaluation concluded that installing a temporary dew point monitor in place of CAS-ME-10, to locally monitor dew point in the CAS system while CAS-ME-10 and CAS-MIS-10 are out of service for maintenance, will not affect the capability to perform a safety function of an important to safety system, structure, or component. Operation of the CAS system is not required for the initiation of any engineered safety feature systems or for safe shutdown of the reactor. No safety-related equipment will be affected by its physical presence. Installing the temporary monitor does not affect the function or operation of the CAS system but rather increases system reliability by allowing monitoring of the dew point while the installed unit is out of service for repairs.

TMR 00-0013 (SE 00-0050)

This modification will temporarily disable the position indication of reactor feedwater valve RFW-V-32A (outboard containment isolation check valve) to allow for the installation of a welded cover onto the stuffing box. Local and control room position indications will be disabled by removing the position indication switches mounting bracket assembly from the valve and installing the welded cover on the stuffing box. The reason for this proposed activity is to correct a recurring leak around the valve position indication shaft packing gland area.

Safety Evaluation Summary

The safety evaluation concluded that this modification will not increase the probability of occurrence, nor the consequences of those accidents evaluated in the FSAR. It will not increase the probability of occurrence or the consequences of malfunction of equipment important to safety. The installation of the welded cover on the stuffing box will enhance the pressure integrity function of the valve. Removal of the position indication switches assembly will affect control room reading and valve exercising requirements. However, there is no impact on the safety-related function of the valve. There is no control room action that relies upon the valve position indication during any accident scenario.

2.6.2 Licensing Document Changes

This section contains information pertaining to Licensing Document Change Notices (LDCNs) and is included pursuant to 10 CFR 50.59.

LDCN-ODCM-98-041 (SE 98-0024)

This LDCN revised the Offsite Dose Calculation Manual to include a definition section, delete an obsolete footnote, and add a statement to the introduction section that references the correct Applicability statement in the Technical Specifications.

<u>Safety Evaluation Summary</u>

The safety evaluation concluded that all the changes are either editorial, clarify requirements, or delete references to obsolete requirements. The change maintains the level radioactive effluent control required by 10 CFR 20, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and does not adversely impact the accuracy or reliability of current effluent, dose, or setpoint calculations.

LDCN-FSAR-99-048 (SE 99-0038)

This LDCN revised the FSAR to correct an error in the station blackout evaluation for the steam tunnel and revised pressure, temperature, and humidity limits for normal ranges in Table 3.11-1, "Normal Operating Conditions."

<u>Safety Evaluation Summary</u>

The safety evaluation concluded that no physical changes are made to plant structures, systems, or components. The changes revise, remove, or add area temperature, pressure, and humidity normal ranges or values. No safety-related equipment operability or safety function were affected by the new area/room temperature limits. There is no change to the assumptions used in the accident analyses thus the consequences of a Chapter 15 accident remain the same.

LDCN-LCS-99-047 (SE 99-0066)

This LDCN revised Licensee Controlled Specification (LCS) 1.7.1, "Area Temperature Monitoring" to:

- Provide new temperature limitations for components.
- Add a requirement to initiate a problem evaluation request when area and room temperatures are exceeded.
- Immediately take action to reduce area and room temperatures to be within LCS limits, and to declare certain equipment inoperable when temperature limits are exceeded.

Safety Evaluation Summary

The safety evaluation concluded that the changes to LCS 1.7.1 do not physically change any plant structures, systems, or components. The new operating temperature limits for areas and rooms covered by the LCS are based on detailed equipment evaluations. It was determined from the evaluation that the new temperature limits have no adverse impact on equipment operation, function, or reliability.

LDCN-FSAR-00-001 (SE 00-001)

This LDCN deleted FSAR Figure 13.5-1, "Operator at the Controls."

Safety Evaluation Summary

The safety evaluation concluded that deletion of FSAR Figure 13.5-1 is consistent with Regulatory Guide 1.70, Revision 2, which requires that an administrative procedure have control room details including the "Operator at the Controls" information. The Regulatory Guide does not require the figure to be located in the FSAR. Details from the FSAR figure were relocated to plant procedure PPM 1.3.1, "Operating Policies, Programs and Practices." Operator location will continue to be controlled in a manner such that operations personnel can effectively respond to component and system indications. The change is consistent with regulatory requirements and guidance pertaining to operators at the controls in the control room of a nuclear power unit.

LDCN-FSAR-00-006 (SE 00-0003)

This LDCN revised text to more clearly reflect the plant configuration pertaining to barriers (walls, doors, penetrations, etc.) between pump rooms. The terms "watertight" and "water resistant" used in the FSAR to describe the barriers between pump rooms located in the reactor building were replaced with discussions that more clearly state that some minimal leakage between rooms occurs during flooding events. In addition, FSAR sections were revised to more clearly describe the connection of the reactor core isolation cooling pump room and the control rod drive/condensate pump room to a common equipment drain sump located in the control rod drive/condensate pump room by means of an unisolable header pipe.

Safety Evaluation Summary

The safety evaluation concluded that eliminating the terms watertight and water resistant as applied to safety-related pump room walls, wall doors, and penetrations does not increase the frequency of accidents nor change the consequences of accidents or equipment malfunctions previously evaluated. In addition, analysis shows that even with a worst case single failure, in conjunction with equipment lost as a result of a flooding event, sufficient safe shutdown equipment remains available and safe plant shutdown can be accomplished.

LDCN-FSAR-00-021 (SE 00-0013)

This LDCN provided for the following changes to the FSAR:

- Revise FSAR Appendix F, Table F.3-1, "Comparison with BTP 9.5-1 Appendix A," as follows:
  - Delete commitment for flame spread rating of 25 for decontaminable coatings.
  - Clarify ATM E84 flame spread rates are not applicable to floor coverings.
  - Include floor coverings representing an unusual hazard in the Fire Hazards Analysis.
  - Define "non-combustible material" consistent with BTP CMEB 9.5-1.
  - Clarify that the combustibility of Thermo-Lag is based on Information Notice 92-82.
- Revise FSAR Appendix F, F.4.2.1, "Combustible Loading Assumptions," to include combustible loading represented by floor covering materials that represent an unusual hazard and list the applicable floor covering materials.

Safety Evaluation Summary

The safety evaluation concluded that these changes satisfy fire protection design requirements and do not result in a degradation of nuclear safety. These changes are in accordance with Columbia Generating Station License Condition 2.C(14), "Fire Protection Program (Generic Letter 86-10)" and will not adversely impact post fire safe shutdown. These changes have no impact on the probability or consequences of previously evaluated accidents or transients.

LDCN-FSAR-00-034 (SE 00-0018)

This LDCN deleted sections of the FSAR describing non-credited potential operator actions. In almost all cases the actions described are in addition to operator actions credited in the actual accident analyses, and are also separate from the FSAR section that describes the analysis. Credited operator action information was retained in the text portion of Chapter 15 and in the sequence of events tables.

Safety Evaluation Summary

The safety evaluation concluded that these changes are consistent with the guidance of Regulatory Guide 1.70, Revision 2, and NUREG-0800. Since the changes are consistent with the guidance documents, and the analyzed plant response to Chapter 15 events is unchanged, the proposed changes to the FSAR do not constitute an unreviewed safety question. This change has no impact on previously evaluated accidents and transients because these operator actions occur after an event and are mitigation actions not credited in the analysis. The plant will continue to be operated in the same manner using the same equipment as previously evaluated.

LDCN-LCS-00-037 (SE 00-0020)

This LDCN deleted LCS Surveillance Requirement 1.8.10.3 requirement to functionally test every 18 months a representative sample, on a rotating basis, of the 480VAC overcurrent circuit breakers. In addition, the list of corresponding 480VAC fused disconnects in Table 1.8.10-1 was deleted.

Safety Evaluation Summary

The safety evaluation concluded that deletion of the LCS SR 1.8.10.3 requirement to functionally test a representative sample of the 480VAC overcurrent circuit breakers every 18 months has no impact on any previously analyzed accident or transient. There are no low voltage circuit breakers currently installed at Columbia Generating Station for the protection of primary containment penetration conductors. The circuit breakers were previously replaced with fused disconnects.

In addition, deletion of the list of corresponding 480VAC fused disconnects in Table 1.8.10-1 has no impact on any previously analyzed accident or transient because fuses are considered to be reliable overcurrent protection devices. The basic design, simple construction, and passive operation make the fuse inherently reliable. Testing of these fuses does not improve the reliability of the overcurrent protective device.

LDCN-FSAR-00-041 (SE 00-0024)

This LDCN revised FSAR Section 9.5.4.4 to allow a 12-year interval between inspection and cleaning of the diesel fuel oil tanks.

Safety Evaluation Summary

The safety evaluation concluded that tank integrity is assured by performance of the ten-year pressure test and is maintained by the fuel oil filter polisher system. This change does not affect the ability of any system to mitigate an accident and does not create any mechanism to create a postulated or new accident condition.

LDCN-FSAR-00-044 (SE 00-0028)

This LDCN revised the description in FSAR Section 9.1.3 of the fuel pool cooling and cleanup (FPC) system. The change removed the words "occasional" and "on occasion" from paragraphs describing the suppression pool cleanup alignment of the FPC system. This change allowed extended operation in the suppression pool cleanup mode provided spent fuel pool water quality parameters are maintained.

Safety Evaluation Summary

The safety evaluation concluded that the change does not alter or introduce any new plant process or configuration, but removes the alignment duration constraints contained in the text. The change is needed to more accurately delineate that the FPC system may remain in the suppression pool cleanup configuration for extended periods, to filter and mix the suppression pool, provided spent fuel pool water quality requirements are maintained. This change does not alter or impact the design intent of how the spent fuel pool cooling and cleanup system is to be operated. Extended alignment of the FPC system in the suppression pool cleanup mode of operation does not impact the ability of any system to mitigate an accident and does not create any new mechanism or interface to create a postulated or new accident condition.

LDCN-FSAR-00-016 (SE 00-0038)

This LDCN removed statements from the FSAR which noted that leakage from closed systems (HPCS, RHR, LCPS, and RCIC) to connected water systems was equivalent to secondary containment bypass leakage. This leakage is not equivalent.

Safety Evaluation Summary

The safety evaluation concluded that the change does not affect the capability of any structures, systems, or components to perform its safety function. This change restores the FSAR use of secondary containment bypass leakage to be consistent with the SER and Standard Review Plan 6.2.3 and is consistent with the facility description as reviewed and accepted by the NRC.

LDCN-FSAR-00-049 (SE 00-0040 and SE 98-0087)

This LDCN changed the description of the Plant Support Facility radioactive contaminated laundry facility to show that the facility will only be used for storage and distribution of protective clothing and that an additional storage facility was added within the protected area but outside the power block.

Safety Evaluation Summary

The safety evaluations concluded that the changes do not involve or impact any plant systems or equipment, do not change radiological doses to the public or plant personnel, or adversely affect margin of safety. The additional storage location fire loading and the potential impacts of a fire on safety-related systems or the ability to safely shutdown will not be impacted. Therefore, this change has no impact on any previously identified accidents or transients.

LDCN-FSAR-00-058 (SE 00-0044)

This LDCN deleted the requirements in FSAR Section 7.7.1.3.2 to verify that the average rate of change for reactor recirculation pump speed is less than or equal to 10% of rated pump speed per second in either the increasing or decreasing direction.

Safety Evaluation Summary

The safety evaluation concluded that deletion of this testing requirement has no effect on equipment important to safety. These limits were tested during post-installation testing of adjustable speed drives and once during performance of the rate-of-change testing. Furthermore, the Columbia Generating Station reload licensing analysis methodology, which is the basis of the Core Operating Limits Report, does not assume any limit on the rate of change of recirculation pump speed.

LDCN-FSAR-00-039 (SE 00-0048)

This LDCN changed the following:

- Added clarification to FSAR Section 2.2.3.1 regarding the capability of the reactor building to withstand a specific exterior blast load.
- Corrected FSAR Sections 2.2.3.1 and 6.4.4.2.3 to reflect the potential maximum amount of hydrogen gas that could be stored at or near the hydrogen gas bottle storage building.
- Corrected FSAR Section 10.2.2 regarding the configuration of the primary and backup hydrogen gas supplies.
- Corrected FSAR Table 3.5-2, "Internally Generated Missiles Outside Containment," resolution code notes associated with LPCS-P-2, HPCS-P-3, CRD-P-1A(B), RHR-P-3, COND-P-3, RCIC-P-1, RCIC-P-2, RCIC-P-3, RCIC-P-4, and RCIC-DT-1.

Safety Evaluation Summary

The safety evaluation concluded that these changes have no adverse affect on nuclear safety. The change added clarification to the FSAR regarding the ability of the reactor building to resist an explosion of 20,000 lbs of dynamite on a railway car 510 feet from the reactor building. The consequences associated with this potential accident are not impacted and are bounded by the original design basis tornado loading condition. The change corrected statements regarding the amount of hydrogen gas stored in and around the hydrogen gas bottle storage building and the configuration of the backup bottles. The consequences associated with an explosion of the hydrogen gas stored at the hydrogen gas bottle storage area and the impact of hydrogen gas bottle leakage are not impacted. A change was made to the resolution codes associated with several internally generated missiles due to analyses that shows the missiles do not have sufficient energy to escape from their existing component casings. Implementation of these changes did not result in an unreviewed safety question.

LDCN-TSB-00-072 (SE 00-0064)

This LDCN revised Technical Specification Bases 3.6.1.3.8 to increase the test pressure range for the reactor instrument line excess flow check valve (EFCV) from 85 to 300 psig and allow reactor instrument line EFCV testing in Mode 4 up to 212°F.

Safety Evaluation Summary

The safety evaluation concluded that this change will not impact plant function and operability. Testing will be performed in Mode 4 within the existing limitation provided by design engineering memorandum TM-2080 and approved plant procedures. Critical instrumentation will remain available during testing and instrument isolation will be performed in accordance with proven methods and procedures. There is no increase in either the probability or consequences of a previously evaluated accident.

LDCN-FSAR-00-076 (SE 00-0065)

This LDCN revised the normal and off-normal temperatures in FSAR Table 3.11-1, "Normal Operating Conditions."

Safety Evaluation Summary

The safety evaluation concluded that the changes are minor and reflect the design database. No plant equipment is being changed. Engineering evaluations have determined there will be no impact to equipment operability or safety performance due to the changes in environmental temperatures. The safety-related equipment in the area/room has been evaluated for the range of temperatures proposed in the change and it was determined there would be no impact on equipment operability/function.

2.6.3 Miscellaneous Changes

This section contains information pertaining to other plant activities and is included pursuant to 10 CFR 50.59.

Problem Evaluation Request 298-2002-02 (SE 00-0004)

This problem evaluation request reported a discrepancy in the design of the bypass and inoperable system indication (BISI) system. Specifically, the low pressure core spray system level out-of-service BISI annunciator does not actuate when the battery, diesel generator, or service water system is out of service. This change revises the description of compliance to Regulatory Guide 1.47 in FSAR Section 7.1.2.4 and in the associated system logic diagrams.

Safety Evaluation Summary

The safety evaluation concluded that the BISI system has no system control functions and provides operator system availability status indication only. No safety system design or function is affected by the design change. Therefore, this change does not result in an unreviewed safety question, does not affect the consequences of previously evaluated accidents or transients, does not create new or different accidents or transients, does not create or affect important to safety equipment malfunctions, and does not affect the margin of safety as defined in the basis for any Technical Specification.

Plant Procedure PPM 10.3.24 (SE 00-0041)

This is a new procedure that provides a processing and handling methodology for irradiated non-fuel material in the spent fuel pool. It also provides for safe handling of heavy loads over safety-related equipment and to provide equipment safe load paths.

Safety Evaluation Summary

The safety evaluation concluded that the ability of the spent fuel pool cooling system to remove heat will not be adversely affected by this change. The integrity of the spent fuel pool and liner is also not compromised. In addition, rigging and casks meet the applicable requirements of NUREG-0612.

Plant Procedures ABN-ASH and ABN-ASH-BASES (SE 99-0058)

This change documented several changes to abnormal operating procedures that were made pertaining to the plant response to an ash fall event.

Safety Evaluation Summary

The safety evaluation concluded that the level of protection remains acceptable and that the current plant response, when compared to the previous commitments in the area of ash fall protection, does not involve an unreviewed safety question or a change to the Technical Specifications.

PID 200-0470-3 (SE 00-0027-00)
PID 200-0470-4 (SE 00-0027-01)

This Problem Evaluation Request Interim Disposition provided for application of furmanite to the valve position indication shaft stuffing box packing gland area on reactor feedwater containment isolation check valve RFW-V-32A for a temporary repair of a steam leak.

Safety Evaluation Summary

The safety evaluation concluded that the proposed activity provided a method for temporarily stopping the leak using a common industry approved process. The furmanite process will not prevent the valve from performing its normal operation and safety function. The furmanite process is not expected to adversely affect reactor operation. The amount of material to be injected was specified and controlled. The change has no impact on any previously identified accident or transient.

Configuration Document Change Request CDCR 98-01-006 (SE 00-0063)

The potable water flow diagram was updated to reflect current conditions. Several buildings and trailers, including their associated piping and valves have been removed or added to the system.

Safety Evaluation Summary

The safety evaluation concluded that the modifications to the potable water system posed no increased risk to the safe operation of Columbia Generating Station. All plant systems and components required to mitigate the consequences of accidents previously evaluated are unaffected by these modifications.

2.6.4 Tests and Experiments

This section contains information pertaining to tests and experiments and is included pursuant to 10 CFR 50.59.

Plant Procedure PPM 8.3.416 (SE 00-0021)

This procedure provided direction to install, functionally test, and remove a Westinghouse 50DHP-VR350 vacuum circuit breaker in the breaker cubicle for E-CB-1/500S and to demonstrate the new model will operate as the original in fit, form, and function.

<u>Safety Evaluation Summary</u>

The safety evaluation concluded that the installation and testing of the Westinghouse vacuum breaker does not increase the probability of occurrence or consequences of an accident. The replacement breaker offers circuit protection and reliability equal to or better than the existing Westinghouse air-magnetic circuit breakers.

2.7 Regulatory Commitment Changes (NEI Process)

This section contains information pertaining to Regulatory Commitment Changes (RCC) and is included pursuant to the NEI Guidelines for Managing NRC Commitment Changes. Included are those commitment changes that satisfied the NEI criteria for reporting.

This section does not include those commitment changes where the NRC Staff was notified of the change under separate correspondence.

RCC-62133-00 (Stability Monitoring)

The original commitment reads, "Revise procedures associated with power oscillation monitoring to require that the ANNA system be operable and in service from greater than 25% reactor power and less than 50% core flow." The intent of the commitment was to have the system operable and a qualified user available for continuous monitoring when the plant was in the Area of Increased Awareness (AIA). The primary reason for continuous monitoring was that the advance neutron noise analysis (ANNA) system does not provide what would be considered an audible alarm. The monitor instead provides an audible computer-type "beep" that is not enhanced by speakers.

The commitment was made in response to a Confirmatory Action Letter and in response to NRC Inspection Report 92-37. [Reference Letters GO2-92-205, dated August 29, 1992, AL Oxsen to NRC, Response to Confirmatory Action Letter, and GO2-93-008, dated January 11, 1993, JV Parrish to NRC, NRC Inspection Report 92-37, Response To Notice Of Violations And Notice Of Deviation-Power Oscillation Event].

The commitment was revised. The commitment revision eliminates the ANNA watch while in the AIA steady state conditions. The oscillation power range monitor (OPRM) alarm function must be available to provide audible indication of potentially unstable conditions or the ANNA watch must be established. Continuous ANNA watch is still required when entering, exiting, or maneuvering in the AIA regardless of the OPRM alarm status.

Since the time of the original commitment, an OPRM has been installed which provides an audible alarm. Although the OPRM trip capabilities are currently deactivated, the OPRM alarms and indications are operational to enhance operator ability to recognize and respond to an instability event. The OPRM functions are currently being monitored to ensure that the OPRM algorithms perform according to design specifications and the system is measured under a variety of operating conditions before the system is armed (pending NRC approval of a Technical Specification amendment request).

During the period of time prior to arming of the system, we are continuing controls which are currently in place to avoid power oscillations and to detect and suppress them if they occur. The methods we use are extensive and consistent with the Boiling Water Reactor Owners' Group interim operating recommendations and our commitments made in response to Generic Letter 94-02, "Long-Term Solutions and Upgrade of Interim Operating

Recommendations for Thermal-Hydraulic Instability Events in Boiling Water Reactors." These interim monitoring measures, in addition to the OPRM operational alarms and indications, are effective and provide adequate protection against reactor core instability events until the long-term solution (arming of the OPRM) is implemented.

RCC-110598-00 (Fire Protection)

The original commitment reads, "the licensee revised the pre-fire plans during the inspection to include this action for fire brigade leaders on all fire occurrences in safety-related areas of the plant. Based on the licensee's corrective actions, resolution of this item is acceptable."

The commitment was relied upon by the NRC to close an item of concern with the Fire Protection Program. [Reference Letter GI2-89-048, dated May 22, 1989, NRC to GC Sorensen, Safety Evaluation by the Office of Nuclear Reactor Regulation Evaluating Implementing Details of the Approved Fire Protection Program].

The commitment was deleted. Procedures have been changed to require that the control room contact the fire brigade leader and evaluate shutting down HVAC system(s) in the affected building/area. Since this action is directed from the control room by approved plant procedures, it is not necessary to be included as pre-fire plan direction for the fire brigade leader.

RCC-170082-00 (Fire Doors)

The original commitment reads, for doors R-6 and R-6A, located between the residual heat removal (RHR) pump rooms, "...these doors will remain closed and locked and will only be opened during shutdown modes of the plant and in the unlikely event of a fire in the RHR room in the northeast corner of the plant (Fire Zone R-V)."

The commitment was made in response to an NRC Fire Protection site audit concern [Reference Letter GO2-83-597, dated July 1, 1983, GD Bouchey to NRC, Response to Fire Protection Site Audit Concerns].

The commitment has been deleted. These doors, like all other doors in the plant, will be maintained closed except during personnel passage. These are also "watertight" doors and special training was provided to plant personnel on the need to maintain these doors closed. Since the fire protection provided is related to the doors being closed, maintaining the doors closed but not locked will provide an equal level of fire protection to that provided by the doors being closed and locked. The NRC was concerned that the doors between the RHR pump rooms would be opened to fight a fire in one of the rooms, thus exposing both divisions to the fire. Columbia Generating Station agreed to lock the doors closed and put in the Pre-Fire Plans to only open the doors between the pump rooms for a fire in the RHR-P-2A pump room when there is a hose stationed to protect the south (RHR-P-2B) pump room equipment. This same method of fire fighting will still be maintained, thus, providing the

necessary protection for RHR, Division B. The only difference will be that the doors will not be locked and will, thus, make fire-fighting actions easier to perform since the key to this door is in the main control room. Administrative means will be used to keep doors R-6 and R-6A normally closed and to control fire-fighting actions.

Appendix A

Annual Personnel Radiation Exposure Work and Job Function Report

Work & Job Function	Number of Personnel Receiving >100 mrem			Total Man-Rem		
	Station Employees	Utility Employees	Contract Workers and Others	Station Employees	Utility Employees	Contract Workers and Others
Reactor Operations & Surveillance						
Maintenance Personnel	34	2	0	5.042	0.120	0.042
Operating Personnel	23	0	1	3.966	0.000	0.128
Health Physics Personnel	16	0	1	2.168	0.056	0.080
Supervisory Personnel	1	0	0	0.053	0.000	0.000
Engineering Personnel	1	0	0	0.132	0.000	0.001
Routine Maintenance						
Maintenance Personnel	26	1	13	7.491	0.133	5.545
Operating Personnel	1	0	0	0.375	0.000	0.021
Health Physics Personnel	1	0	0	1.420	0.006	0.012
Supervisory Personnel	0	0	0	0.008	0.000	0.000
Engineering Personnel	1	0	1	0.351	0.000	0.275
Inservice Inspection						
Maintenance Personnel	1	0	0	0.332	0.000	0.000
Operating Personnel	0	0	0	0.047	0.000	0.000
Health Physics Personnel	0	0	0	0.000	0.000	0.000
Supervisory Personnel	0	0	0	0.000	0.000	0.000
Engineering Personnel	0	0	0	0.000	0.000	0.000
Special Maintenance*						
Maintenance Personnel	5	1	2	3.224	0.413	0.627
Operating Personnel	0	0	0	0.261	0.000	0.000
Health Physics Personnel	1	0	0	0.546	0.004	0.001
Supervisory Personnel	0	0	0	0.046	0.000	0.000
Engineering Personnel	0	0	0	0.103	0.000	0.000
Waste Processing						
Maintenance Personnel	4	1	6	1.322	0.343	3.503
Operating Personnel	0	0	1	0.001	0.000	0.220
Health Physics Personnel	1	0	0	0.954	0.164	0.015
Supervisory Personnel	0	0	0	0.000	0.000	0.000
Engineering Personnel	1	0	0	0.107	0.000	0.000
Refueling						
Maintenance Personnel	0	0	0	0.008	0.000	0.000
Operating Personnel	0	0	0	0.015	0.000	0.000
Health Physics Personnel	0	0	0	0.006	0.000	0.000
Supervisory Personnel	0	0	0	0.000	0.000	0.000
Engineering Personnel	0	0	0	0.000	0.000	0.000
TOTAL						
Maintenance Personnel	70	5	21	17.419	1.009	9.717
Operating Personnel	24	0	2	4.665	0.000	0.369
Health Physics Personnel	20	0	1	5.094	0.230	0.108
Supervisory Personnel	1	0	0	0.107	0.000	0.000
Engineering Personnel	3	0	1	0.693	0.000	0.276
Grand Total	**118**	**5**	**25**	**27.978**	**1.239**	**10.470**

Total number of personnel receiving >100 mrem = 148. Total man-rem for personnel receiving > 100mrem = 39.687

Report produced from electronic dosimeter data

*Special Maintenance:

- RFW-V-32A steam leak investigations, repairs, & associated support work (forced outage)
- RRC-P-1A seal replacement and associated support work (forced outage)
- BS-MANWAY-3/1 & 3/2 steam leak repairs
- MS-HX-1A Manway steam leak repairs
- MS-V-GV/1 servo valve oil leak repairs
- IRM-DET-2E replacement (forced outage)
- MS-V-20 adjust/repack (Steam Tunnel entry at power)
- TER 99-0068-0 TIP-POS-5A through 5E proximity switch replacement