ENERGY
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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 2001**

The annual operating report for calendar year 2001 is attached. If you have any questions or desire additional information pertaining to this report, please contact Ms. CL Perino at (509) 377-2075.

Respectfully,

GO Smith
Vice President, Generation
Mail Drop PE04

Attachment

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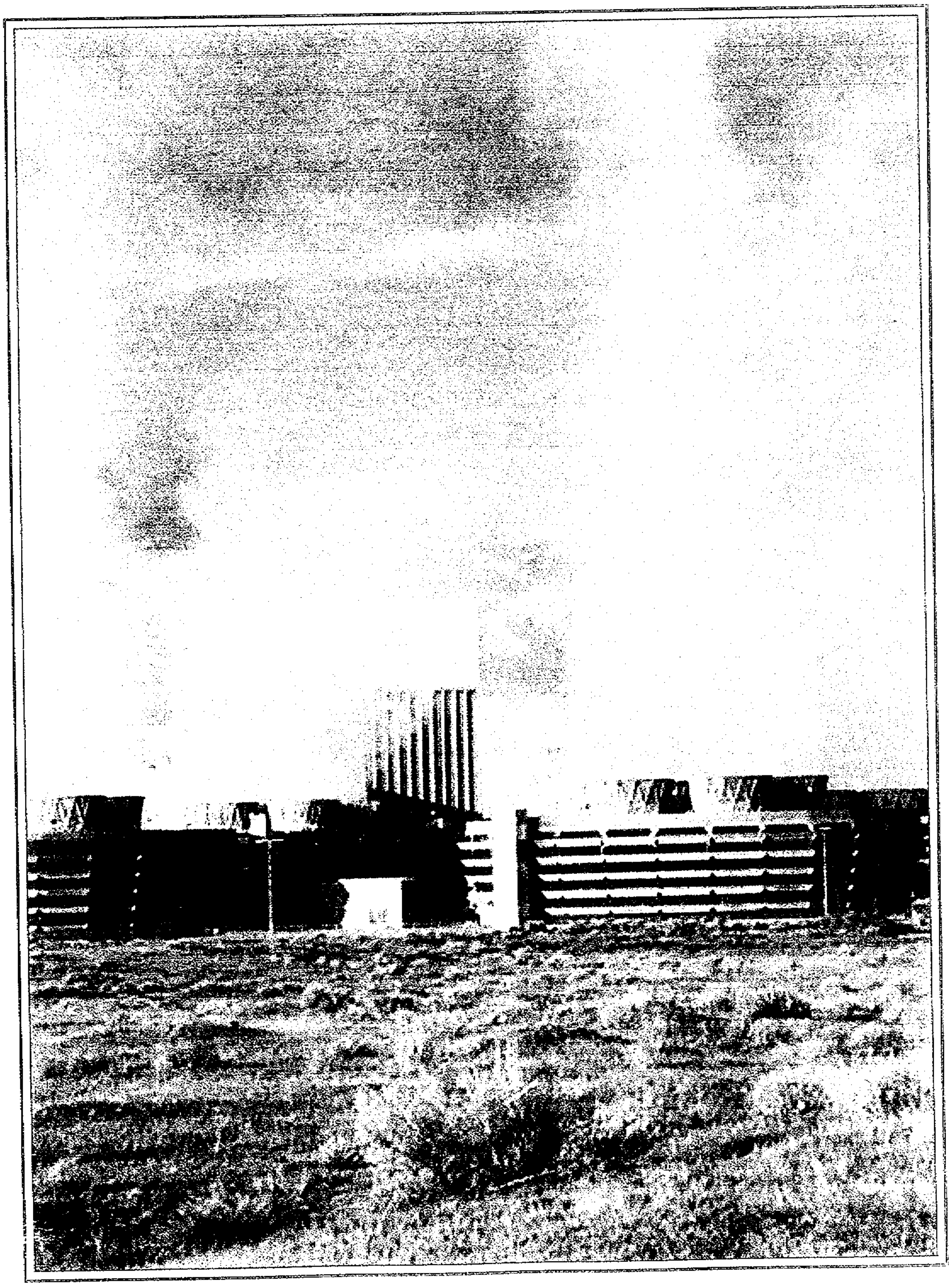
Rec'd 03/19/02

Columbia Generating Station Annual Operating Report 2001



COLUMBIA GENERATING STATION

ANNUAL OPERATING REPORT

2001

DOCKET NO. 50-397

FACILITY OPERATING LICENSE NO. NPF-21

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

Columbia Generating Station
2001 Annual Operating Report

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1.0 Introduction

The 2001 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 May 18, 2001, following an operational run of 238 days, the plant was taken off line for a maintenance and refueling outage. The station's outage ended on July 2, 2001, with final synchronization to the electrical grid. Full power operation was reached on July 6, 2001.

The plant remained on-line for 24 days until July 26, 2001, when the plant was taken off line for a planned forced outage. The scope of this outage was to replace a seal in reactor recirculation system pump RRC-P-1B, repair a tube leak in the main condenser water box B, and identify and repair a leak in the steam tunnel. Following repair efforts, the plant was synchronized to the electrical grid on August 2, 2001. The plant was returned to 100 percent full power operation on August 3, 2001 and remained on line for the rest of the year.

During 2001 the station produced 8,250,429 megawatt-hours of generation, which was its second highest year of production. An extended refueling outage and the forced outage in July prevented it from being a record year.

The generation of 2,497,758 megawatt-hours during the fourth quarter of 2001 was the highest quarter in station history. December 2001 was the fourth consecutive month that the station generated more than 99 percent of its capability. Previously, this had only been accomplished approximately one month out of every nine.

During 2001, the Radiation Protection organization achieved its goal of having less than one percent of recoverable floor space that is contaminated.

Construction of an Independent Spent Fuel Storage Installation began in June of 2001 and is scheduled for completion by April 2002. This dry cask storage system will allow for safe and efficient storage of the station's spent fuel until such time as it can be transported to a national repository.

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 evaluation of each. The 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 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 2001

- At the beginning of the month the station was operating at full power. On January 12, 2001, a downpower to about 80% was initiated for turbine front standard testing. The station returned to full power the following day. Other than downpowers of less than 20% for scheduled testing and rod pattern changes, the station remained at full power for the remainder of the month.

February 2001

- At the beginning of the month the station was operating at full power. On February 2, 2001, a downpower to approximately 70% was initiated for a rod sequence exchange and scheduled testing. The station returned to full power the following day. Other than downpowers of less than 20% for scheduled testing and rod pattern changes, the station remained at full power for the remainder of the month.

March 2001

- At the beginning of the month the station was operating at full power. On March 9, 2001, a downpower to approximately 72% was initiated for rod pattern adjustment and scheduled turbine valve testing. The station returned to full power the following day. Other than downpowers of less than 20% for scheduled testing and rod pattern changes, the station remained at full power for the remainder of the month.

April 2001

- At the beginning of the month the station was operating at full power. On April 8, 2001, reactor power began decreasing marking the beginning of coast down. Other than downpowers of less than 20% for scheduled testing and rod pattern changes, the station remained at as high a power as the fuel could sustain for the remainder of the month.

May 2001

- At the beginning of the month the station was operating at 88% power. Coast down to approximately 82% power was achieved prior to shutdown for refueling. Power reduction for the scheduled refueling outage was initiated on the May 18, 2001, and on May 19, 2001, a manual scram was inserted as part of the normal procedure for shutdown.

June 2001

- The station was in a maintenance and refueling outage for the entire month.

July 2001

- At the beginning of the month the station was preparing for restart following the refueling outage. On July 1, 2001, criticality was achieved and the station reached 88% power on July 5, 2001.

- On July 5, 2001 a downpower to 29% was initiated to recover a tripped adjustable speed drive for recirculation system pump RRC-P-1B. Full power operation was reached on July 6, 2001.

- Power reduction for a forced outage to repair a recirculation system pump RRC-P-1B pump seal was initiated on July 26, 2001, and a manual scram was inserted as part of the normal procedure for shutdown.

August 2001

- At the beginning of the month the station was preparing for restart following the forced outage. On August 1, 2001, criticality was achieved and the station resumed full power on August 3, 2001. Other than downpowers of less than 20% for scheduled testing and rod pattern changes, the station remained at full power for the remainder of the month.

September 2001

- At the beginning of the month the station was operating at full power. On September 28, 2001, a downpower to 73% was initiated for a rod pattern adjustment and scheduled testing. Full power operation resumed the following day.

- On September 30, 2001, a downpower to 76% was initiated to test main steam valve MS-V-164B. Full power operation was resumed later the same day.

October 2001

- At the beginning of the month the station was operating at full power. On October 12, 2001, a downpower to 70% was initiated for a rod sequence exchange. The station returned to full power the following day. Other than downpowers of less than 20% for scheduled testing and rod pattern changes, the station remained at full power for the remainder of the month.

November 2001

- At the beginning of the month the station was operating at full power. On November 24, 2001, a downpower to about 80% was initiated for scheduled testing. Full power operation was resumed later the same day.

- On November 29, 2001, a downpower to 77% was initiated for a rod sequence exchange. Full power operation was resumed the following day.

December 2001

- At the beginning of the month the station was operating at full power. On December 20, 2001, a downpower was initiated for bypass valve and turbine front standard tests. The station commenced raising power on December 21, 2001. However, due to problems with reactor feedwater system heater level controls, the power ascension was halted at 90% and later lowered back to 80%. After recovery of the reactor feedwater heaters, the reactor was returned to full power on December 22, 2001. The station remained at full power for the remainder of the month.

2.2 Significant Maintenance Performed on Safety-Related Equipment

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

Emergency Diesel Generator [DG-1 and DG-2]

Weld repairs of the channel head divider plates on diesel cooling water heat exchanger DCW-HX-1A2 and DCW-HX-1B1 were performed as part of preventive maintenance activities.

A tube bundle was rotated on diesel cooling water heat exchangers DCW-HX-1A1 and DCW-HX-1B2. Eddy current testing had shown that the two bundles were mis-oriented.

Emergency Diesel Generator [DG-3]

A six-year maintenance overhaul was performed. This included disassembly and reassembly of the top deck for replacing the head gaskets on all 20 cylinders, replacement of miscellaneous flex hoses, cleaning and rebuilding of the pneumatic governor booster and confirmation of proper operation of the engine turbocharger.

Reactor Core Isolation Cooling System

Reactor core isolation cooling system valve RCIC-V-63 was completely rebuilt including valve refurbishment, stem replacement, torque switch replacement and repair to correct a packing leak problem.

Reactor Recirculation System

The seal on recirculation system pump RRC-P-1A was replaced due to wear. The seal on pump RRC-P-1B was also replaced as a scheduled change during the refueling outage. However, upon startup, seal performance degraded to the point where it was necessary to replace it for a second time during a forced outage.

Leaks were repaired and the reed switch assembly was replaced on recirculation system valve RRC-V-19 following indications of leakage.

Residual Heat Removal System

Inspection and tube plugging were performed on residual heat removal heat exchanger RHR-HX-1A as part of preventive maintenance activities.

Standby Service Water System

Weld repairs were performed on downstream piping associated with standby service water valves SW-V-2A and SW-V-2B as part of preventative maintenance activities.

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. A fuel defect was identified during the calendar year 2001 portion of Cycle 15. This defect was initially identified as a suspected defect and reported during calendar year 2000.

Analysis of fission gas isotopes indicated a small fission gas leak occurred between November 21, 2000 and November 28, 2000. A full-core sipping was performed during the maintenance and refueling outage in May 2001. Sipping efforts located the leak as coming from Bundle WQE106, which was a once-burned bundle.

The bundle was removed from the core and a detailed examination in November 2001 identified the leaking fuel rod (Rod B7). The failure was determined to be debris fretting under the second spacer location. There were indications of secondary hydriding, but no secondary failure occurred.

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.)

Both General Electric (GE) and Westinghouse (previously Asea Brown Boveri - ABB) methodologies were 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 the SPC fuel, there was an error in the GE ECCS LOCA analysis model which resulted in the accumulated licensing basis upper-bound peak cladding temperature being increased by 5° Fahrenheit (1478°F to 1483°F). During the 2001 refueling outage, all remaining SPC fuel was discharged from the core.

For the Westinghouse fuel, there were no errors in the ECCS LOCA analysis model or application of the model for 2001.

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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 5, "Columbia LOCA Analysis Report," Westinghouse Electric Company, February 2001

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

Columbia Generating Station implemented the revised 10 CFR 50.59 rule in August 2001. In 2001, there were no evaluations performed under the new rule. There were several evaluations performed under the old rule.

This section contains summaries of safety evaluations (SE) for activities implemented during 2001 that were assessed pursuant to the previous 10 CFR 50.59 reporting requirements. Accordingly, the term *unreviewed safety question* still applies in the evaluation of those changes.

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 Design Changes (PDCs), Plant Modification Records (PMRs), Technical Evaluation Requests (TERs), and Temporary Modification Requests (TMRs) and is included pursuant to 10 CFR 50.59.

PDC 0000000847 (SE 01-0018)

This modification provided the justification for the use of a new mode of reactor core isolation cooling system operation to be used for increased control of reactor vessel water level when the system is in use. Increased reactor pressure vessel water level control is accomplished by allowing the reactor core isolation cooling system to enter a batch flow process that will direct all RCIC flow either to the vessel by means of injection valve RCIC-V-13 (RPV injection mode) or to the condensate storage tanks through series valves RCIC-V-22 and RCIC-V-59 (the normal path for the RCIC full flow test, CST-to-CST). Flow control to the CSTs will be provided by throttling RCIC-V-22. This method of operation will reduce the number of system starts and stops due to vessel level tracking.

Safety Evaluation Summary

The safety evaluation concluded that the change does not affect reactor core isolation cooling system/component design or safety function, including system equipment required for high energy line break event mitigation or containment isolation functions during a design basis accident, nor does it affect the automatic response of the system to accidents, transients or special events.

This change adds a new mode of system operation to be used to provide increased control of reactor pressure vessel water level when the system is in operation. This method of operation will reduce the number of reactor core isolation cooling system starts and stops as vessel water level varies between reactor vessel Level 2 and Level 8. This change requires no physical plant modifications.

The change does not increase the probability or consequences of previously-analyzed accidents, transients or special events, or malfunctions of equipment important to safety nor create the possibility of a new or different type of accident, transient or malfunction of equipment important to safety. Safety margins defined in Technical Specifications are maintained. The proposed change does not alter the ability of the operator to monitor system conditions, detect component failures, and to act upon them. This change does not affect the operation of any other important-to-safety system.

PDC 00126701 (SE 00-0068)

This modification provided for the installation of a check valve, related instrumentation and changes in the configuration of the keep fill suction piping in the reactor core isolation cooling system to reduce the probability of water hammer. The proposed changes are necessary to prevent water hammer due to drain down of pump discharge piping during the period when the reactor core isolation cooling system is tripped between reactor vessel Level 8 and Level 2. Water hammer may result in loss of RCIC piping integrity, loss of primary containment integrity, flooding, and other potential effects.

Safety Evaluation Summary

The safety evaluation concluded that, by preventing water hammer and its consequent potential for piping failure, the modification ensures the system and other important-to-safety systems potentially affected by water hammer in the reactor core isolation cooling system will continue to respond to and mitigate the effects of associated design basis events.

These design changes enhance the system's capability to serve as backup to the high pressure core spray system for the control rod drop accident and anticipated transient without scram, and as the preferred source of reactor vessel inventory for a station blackout event. They also ensure that the reactor core isolation cooling system will not fail in such a manner as to affect safety related systems during other design basis events.

The proposed modifications do not affect postulated pipe breaks/cracks generated within the reactor core isolation cooling system. No existing or new types of accidents, transients or equipment malfunctions are affected or will result, from these modifications. The modification will improve system reliability during its response to reactor Level 2 and Level 8 signals by ensuring that water hammer conditions do not occur in the event a single failure of either valve RCIC-V-46 and/or RCIC-V-19 to close.

PMR 85-0528 (SE 94-0142)

This modification provided for the replacement and recalibration of overcurrent relays to allow for increased precision of inverse time-current settings and coordination with upstream and downstream overcurrent devices. These overcurrent relays provide overcurrent protection for the 4.16KV and 6.9KV buses and feeders and station service auxiliary transformers TR-S, TR-N1, TR-N2 and TR-B. This design change also provided for revised overcurrent relay and low voltage circuit breaker setpoints.

Safety Evaluation Summary

The safety evaluation concluded that this change would not impact the ability of the electrical distribution system to function and monitor overcurrent transients and accidents. The modification also does not change the functional design, logic or control scheme of any component or system. The proposed activity consisted of resetting of existing relays and replacement of additional relays which are of the same electromechanical type in order to provide proper overcurrent coordination of the medium voltage switchgears.

No new system failure modes have been added and no existing failure modes have been changed since the new relays are of the same electromechanical induction disk design. All other changes consist of recalibrations of existing equipment.

PMR 89-0234 (SE 00-0025)

This modification provided for the replacement of the two vital, Division 1 and Division 2, safety related, 120 VAC inverters with four new inverters (one active and one installed spare in each division). It also replaces the two vital, safety related, Division 1 and Division 2, 125 VDC battery chargers with four new safety related chargers (one active and one installed spare in each division). The vendor no longer manufactures replacement parts for our existing inverters and chargers and there was no backup inverter or charger available to carry the load if one of these inverters or chargers fails.

Safety Evaluation Summary

The safety evaluation concluded that the replacement equipment performs the same functions as the obsolete equipment it is replacing. The reliability of the system is being improved by installing redundant units in place of single units, thus, providing built-in spares. In each case, one inverter or charger will be in operation and the other will act as an installed spare.

The two inverters in each division will provide a built in spare inverter. If the on-line inverter fails, power will be transferred to the alternate power source. The off-line inverter will be placed in service and the load will be transferred back to this inverter.

This change will provide redundant chargers for each of the 125 VDC batteries. These chargers will both be full capacity chargers so a single charger will be capable of supplying all the DC loads for its division. One charger will be floating on the bus and the other charger will be shut down.

PMR 93-0037 (SE 99-0024)

This design change provided for the installation of a permanent platform in the reactor building to provide access to high pressure core spray system valve HPCS-V-4. The platform replaced a long standing scaffold which has been required by plant personnel to perform testing and surveillances on HPCS-V-4.

Safety Evaluation Summary

The platform performs a passive function of providing access to HPCS-V-4. Thus, the proposed platform will not have any adverse impacts or interactions with important to safety systems or components.

Temporary members or restraints were provided during the installation phase as required by plant procedures to ensure that interim configurations of the platform were in acceptable seismic configurations. This ensured that there were no adverse impacts between the permanent platform and important to safety systems or components during the installation phase. The activity did not directly or indirectly affect the capability of any important to safety structures, systems or components to perform their intended safety function as described in the safety analysis report.

PMR 94-0274-03 (SE 00-0067)

This modification provided for the installation and testing of a mitigation monitoring data acquisition system for the reactor water cleanup system in support of NobleChem application. The new system will be used to help determine when a NobleChem re-application is required. The system also provides the capability to determine and monitor reactor coolant electrochemical corrosion potential which provides a measure of susceptibility of the wetted reactor internals and reactor recirculation system piping to intergranular stress corrosion cracking. The electrochemical corrosion potential data will be used to help determine the required hydrogen injection rate for the hydrogen generation system.

Safety Evaluation Summary

The safety evaluation concluded that the addition of the mitigation monitoring data acquisition system to the reactor water cleanup system will comply with all applicable standards and codes. There will be no increased probability of occurrence or increased consequences for any accident or malfunction of equipment previously evaluated in the safety analysis report. Further, there are no new accident scenarios or equipment malfunctions introduced by the proposed change. All plant systems and components required to mitigate the consequences of previously evaluated accidents are not impacted by the proposed change.

PMR 94-0274-04 (SE 00-0074)

This modification provided for the addition of an electro-chemical corrosion potential measurement system for in-core measurement. The main component in this system is a GE-designed assembly that contains four local power range monitor detector strings, a calibration tube and three electro-chemical corrosion potential electrode strings. The in-core system provides primary measurements which are used to adjust hydrogen injection rate.

Safety Evaluation Summary

The safety evaluation concluded that the in-core electro-chemical corrosion potential measurement system is non-safety and has no direct or indirect effect on any other plant system. There are no important to safety effects resulting from this modification that change the operation or function of any system or change any safety margin in the Technical Specifications. This activity does not increase the probability of occurrence of any accident/transient or consequences of any accident/transient evaluated previously in the safety analysis report. It also does not increase the probability of occurrence of or change the consequences of important to safety equipment malfunctions.

PMR 94-0274-05 (SE 00-0039)

This modification provided for the removal of the existing dissolved oxygen sensor and the installation of a new hydrogen/oxygen sample panel in support of implementation of the hydrogen generation system in order to monitor water chemistry for long-term protection of reactor internals.

Safety Evaluation Summary

The safety evaluation concluded that removal of the existing oxygen sensors and replacement with a new, more reliable, oxygen and hydrogen monitoring system will not increase the possibility or consequences of any accidents or transients previously evaluated. The new sample rack will perform the same basic function as the existing system, but will enhance the system function and provide, in addition to the dissolved oxygen saturation level already monitored, the dissolved hydrogen saturation level as well. With this new system, the time required to calibrate both hydrogen and oxygen monitoring systems is greatly reduced and the span between calibrations is increased which, in turn, reduces out-of-service time.

PMR 94-0274-06 (SE 01-0012; SE 01-0014; SE 01-0015R0; SE 01-0015R1; SE 01-0017R0; SE 01-0017R1)

This modification provided for the injection of noble metal compounds into the reactor vessel to create a catalytic layering of noble metals (platinum/rhodium) to reduce the

hydrogen addition injection rate thereby achieving intergranular stress corrosion cracking protection while minimizing the effects of high dose rates attributed to regular hydrogen injection rates. This process is commonly called the NobleChem process.

Safety Evaluation Summary

The safety evaluations concluded that the NobleChem process does not create any adverse equipment interaction. No new malfunction of any equipment important to safety is created during NobleChem application. After the noble metal application is completed, the reactor vessel, the reactor internals and some of the associated primary pressure boundary piping and equipment becomes better protected from intergranular stress corrosion cracking and, therefore, have reduced probabilities of failures and/or malfunctions.

NobleChem experience and testing to-date indicates that NobleChem will lead to increased cladding corrosion. However, this increased corrosion is not expected to lead to adverse effect on the thermal or mechanical properties of any fuel bundle component if the noble metal loading on fuel is at a low to moderate range. This is expected to be true for all normal operating conditions over the design life of the fuel. The specified acceptable fuel design limits as specified in 10CFR50, GDC-10 and NRC Standard Review Plan will be met under NobleChem application conditions.

The noble metal layer will eventually wear and eventually become ineffective. Reapplication will be required at periodic intervals to maintain the ability to reduce the electro-chemical corrosion potential to a beneficial level with moderate hydrogen addition rates.

PMR 96-0010 (SE 00-0031)

This modification provided for an increase in the slope and line size of the seal drain lines for reactor feedwater system pumps RFW-P-1A and RFW-P-1B to minimize oil and water contamination.

Safety Evaluation Summary

The safety evaluation concluded that the purpose and function of all equipment associated with the piping is not impacted by the proposed modification. All impacted components will continue to meet the applicable ANSI/ASME Code and Columbia Generating Station requirements. There will be no increased probability of occurrence or increased consequences of any accident or malfunction of equipment previously evaluated. Further, there will be no new accident scenarios or equipment malfunctions introduced by the change.

PMR 96-0157 (SE 00-0049)

This modification provided for the installation of the plant process computer replacement system (PPCRS) upgrade to replicate and integrate with the recently-upgraded transient data acquisition system (TDAS) computer (previously upgraded under the Y2K project). This will provide redundant process computer systems. In its final configuration, either system will be able to provide complete and independent operation of both TDAS and PPCRS functions.

Safety Evaluation Summary

The safety evaluation concluded that the change does not impact any of the previously-evaluated accident analyses. Components and circuits installed by this modification do not create significant new sources of electronic interference, nor is the new installation affected by any electronic interference impacts from existing equipment. New control power circuits are installed in grounded flex conduits within the main control room. cables routed between control panels are fiber optic, which can not induce electromagnetic fields on adjacent cables or components. New computer equipment operates at low signal levels. The main control room has instrument and equipment grounding systems to minimize electronic fields, with most equipment located within grounded metallic enclosures (panels).

There will be no increased probability of occurrence or increased consequences for any accident or malfunction of equipment previously evaluated in the safety analysis report as a result of this change. All plant systems and components required to mitigate the consequences of previously evaluated accidents will not be unacceptably impacted by the proposed change.

PMR 97-0040-03 (SE 01-0022)

This modification provided for implementation of the Cycle 16 core design. In addition, the ultimate heat sink (UHS) analysis was revised to incorporate a 24-month fuel cycle, a full spent fuel pool, and a greater siphon differential (22 inches instead of 18 inches).

Safety Evaluation Summary

It was concluded from the safety evaluation that assumptions and bounding conditions were identified and accounted for relevant to the Cycle 16 safety analysis. The COLR limits are established to maintain the probabilities and consequences in the safety analysis and maintain margin of safety. The enrichment change in the reload fuel is within the approved scope of NRC approved methodology. In addition, the design change does not affect the safety function of the UHS or standby service water (SSW) systems. The safety function of these systems is to provide cooling water for up to 30

days without makeup water to help mitigate the consequence of a transient. The UHS and SSW systems continue to meet their basic design requirements.

PMR 97-0137 (SE 00-0015)

This modification provided for the elimination of two moisture separator reheater steam trap stations to eliminate energy that was being lost to the main condenser. The trap stations had also been cycling at frequent intervals, which required frequent servicing or replacement of the associated valves.

Safety Evaluation Summary

The safety evaluation concluded that the basic operation of the moisture separator reheater will not change. The change in the system will result in simpler operation because there will be less equipment to test, monitor and maintain. There are also no equipment or components that are important to plant safety that will be adversely impacted by this change.

PMR 98-0082-0 (SE 99-0017)

This modification provided for an improved seating configuration for the main steam safety relief valves (MSRVs) utilizing a redesigned disc insert. Crosby Valve and Gage Company, the original designer and manufacturer, developed the improved seating configuration. This new seating configuration is necessary because the MSRVs had excessive seat leakage during plant operation.

This design change provided for the modification of four MSRVs to incorporate the vendor's improved seat design to reduce seat leakage. Modified and improved MSRVs were installed in the plant in accordance with approved procedures.

Safety Evaluation Summary

The safety evaluation concluded that, based on the prototype testing and the evaluation of the changes by Crosby Valve and Gage Company and Columbia Generating Station Engineering, the modification does not impact MSRV operation, performance, or function, except for seat leakage reduction. The MSRVs continue to maintain their rated capacity, setpoint and blowdown requirements. All MSRV safety setpoints as listed in the Technical Specifications remain unchanged; therefore, the proposed modification does not reduce the margin of safety as defined in the basis for any Technical Specification. The proposed modification does not create a new accident or any new malfunction of important to safety equipment.

PMR 98-0117 (SE 00-0010)

This modification allowed for the replacement of low pressure turbine seal steam pressure transmitters/transducers SS-PT-5B/PTD-5B and SS-PT-7A/PTD-7A (obsolete-no longer available), and replaces them with Rosemount 1153 single unit transmitters with improved functional capabilities.

Safety Evaluation Summary

The safety evaluation concluded that the implementing activity is the replacement of obsolete non-safety related turbine seal steam pressure transmitters. The replacement transmitters are radiation resistant, have the same or improved performance capability and will serve the same functional requirements as the original devices. The previously evaluated accident analyses are not affected by this change.

PMR 99-0051 (SE 00-0007)

This modification provided for removal of the mechanical components (position indicator rod assembly and sensor arms) used for disk position indication from within the inside of the residual heat removal system check valves RHR-V-41B, RHR-V-41C, RHR-V-50B, and RHR-V-89. The disk indication feature of these testable check valves has proven to be prone to malfunctions, and in some instances may actually interfere with the required surveillance testing requirements (INPO SOER 86-03). Therefore, the removal of these internal components eliminated the possibility of invalidating test results.

Safety Evaluation Summary

The safety evaluation concluded that this change does not affect the original safety-related design functions of the RHR system or the check valves. These valves are normally closed, and while in the closed position, function as containment isolation valves and high-low pressure interface valves between the reactor coolant and portions of the emergency core cooling system (ECCS). These valves must open to facilitate operation of part of the ECCS. They are also designed to fail closed and isolate to provide containment isolation, and the fail-safe mode will not be impacted by this change.

PMR 99-0103 (SE 99-0062)

This modification provided for a second set of undervoltage relay contacts in the automatic pump start logic for low pressure core spray system pump LPCS-P-1 and residual heat removal system pump RHR-P-2C to prevent breaker tripping caused by overlap of the undervoltage trip and the automatic start signals. This modification also inserted a set of contacts from the bus undervoltage trip relay into the logic of the automatic pump start circuit for pumps LPCS-P-1 and RHR-P-2C.

Safety Evaluation Summary

The safety evaluation concluded that there is no safety significance associated with this modification. The change had no impact on system function, operation, alarms, or operating procedures. The only change is that the automatic start circuit on pumps LPCS-P-1 and RHR-P-2C will not initiate until the trip circuit undervoltage relay has reset when the undervoltage condition clears.

PMR 99-0107 (SE 00-0055)

This modification provided for a change to the sealing mechanism for RCIC-V-31 to provide for venting of pressure between the wedge disc to the high pressure side (containment side) of the valve. A bypass line was installed from the valve body cavity to the containment side of the valve weld end (containment side).

Safety Evaluation Summary

It was concluded from the safety evaluation that the bypass line would not interfere with the performance of the valve or system. This change enhances the reliability of the opening function of the valves through the elimination of pressure locking as a potential failure mode. The change will provide a passive form of pressure equalization. The reactor isolation function is not affected. The non-containment-side disc continues to perform the containment isolation function.

PMR 99-0108 (SE 00-0059)

This modification provided for a change to the sealing mechanism for HPCS-V-4 to provide for venting of pressure between the wedge disc to the high pressure side (containment side) of the valve. A hole was drilled in the containment side disc.

Safety Evaluation Summary

It was concluded from the safety evaluation that the bypass line would not interfere with the performance of the valve or system. This change enhances the reliability of the opening function of the valves through the elimination of pressure locking as a potential failure mode. The change will provide a passive form of pressure equalization. The reactor isolation function is not affected. The non-containment-side disc continues to perform the containment isolation function.

PMR 99-0109 (SE 00-0052)

This modification provided for a change to the sealing mechanism for HPCS-V-12 to provide for venting of pressure between the wedge disc to the high pressure side

(containment side) of the valve. A bypass line was installed from the valve body cavity to the containment side of the valve weld end (containment side).

Safety Evaluation Summary

It was concluded from the safety evaluation that the bypass line would not interfere with the performance of the valve or system. This change enhances the reliability of the opening function of the valves through the elimination of pressure locking as a potential failure mode. The change will provide a passive form of pressure equalization. The reactor isolation function is not affected. The non-containment-side disc continues to perform the containment isolation function.

PMR 99-0140 (SE 00-0046)

This modification provided for the replacement of all safety related Division 1 and Division 2, 4160-volt circuit breakers with new vacuum circuit breakers to enhance auxiliary power system reliability.

Safety Evaluation Summary

The safety evaluation concluded that the replacement circuit breakers are direct roll-in replacement breakers for use in the existing metal-clad switchgear assemblies and are supplied by the original equipment manufacturer of the original switchgear. The replacement circuit breakers are designed, rated, and tested to comply with the same ANSI/IEEE standards as the original circuit breakers. The replacement circuit breakers are also qualified by test to the same seismic and environmental requirements as the original circuit breakers. The replacement breakers generate less heat load and their DC control power is well within the capabilities of the plant DC control power capacity. No change in the DC battery emergency load profile is required. Therefore, there will be no adverse effect on the 125 V DC Class 1E battery loading.

PMR 00-0120 (SE 99-0068)

This change provided for the reduction in the HPCS battery peak loading, during the first thirteen seconds of HPCS operation to restore the battery's lost aging margin. Battery peak load was reduced by changing the timing of the changing motor operation for the HPCS pump motor and HPCS diesel generator output breakers. In addition, the duty cycle values for field flashing for diesel lube oil system pump DLO-P-10 were revised to match the values used in the battery sizing calculation.

Safety Evaluation Summary

The safety evaluation concluded that the change did not affect the HPCS diesel generator or HPCS pump motor safety-related functions. The HPCS battery is adequately sized and

the margin in the battery sizing calculation was reduced from 125% to 116% to avoid exceeding the battery duty cycle peak currently referenced in the Technical Specifications. The change reduced the HPCS battery duty cycle peak current, which also lowers the surveillance test acceptance values, and restored the battery's lost aging margin.

The operation of the charging motor breaker was changed to prevent the breaker from being re-closed for one-to-two seconds after the trip; the time it takes to recharge the closing spring. This design is acceptable since none of the HPCS system accident or transient response scenarios require the re-closure of the HPCS emergency diesel generator (DG3) output breaker or the HPCS pump breaker (HPCS-CB-P1) after trip. When a LOCA signal is received with the diesel generator in the test mode and paralleled with the offsite power source, the DG will auto-return from the test mode and the output breaker will trip and the loads will be applied to the off site source.

In addition, as an unrelated change to this design modification, the duty cycle values for field flashing and DLO-P-10 were revised to match the values used in battery sizing calculation. This calculation was revised to use device resistance data and nominal system voltage to conservatively determine the duty cycle value for field flashing.

TER 96-0049 (SE 98-0110)

This change provided for the replacement of threaded pipe caps with quick-disconnects on control rod drive scram discharge volume vents. The quick-disconnects will facilitate venting air when filling the scram discharge volume with water to calibrate scram discharge volume level transmitters and level switches. Installation of the quick-disconnects will reduce labor time, which will result in less radiation exposure and also minimize the potential for contamination.

Safety Evaluation Summary

The safety evaluation concluded that the new quick-disconnects function is equivalent to or better than the original pipe caps. The pipe caps and the replacement quick-disconnects are neither safety-related nor are they pressure-retaining components. This activity does not result in a change to the plant that will adversely impact any safety-related systems, structures or components.

TER 99-0013 (SE 99-0036)

This change provided for the replacement of existing containment instrument air system valves CIA-RV-5A and CIA-RV-5B Lonergan Model LCT-30 relief valves with Crosby 800 Omni-Trim valves. The Lonergan models are no longer available. The replacement Crosby valves meet the CIA design criteria for Code Group C (ASME III, Section 3). In addition, a test port was provided in the discharge path of each relief valve at a location prior to the junction of the discharge paths into a common pipe.

Safety Evaluation Summary

The safety evaluation concluded that the replacement of original equipment valves with Crosby items, and the addition of test ports in the discharge piping, did not increase the probability or consequence of any accident or equipment malfunction.

Changing CIA-RV-5A and 5B to Crosby components does not affect the capability of any system, structure or component to perform its intended safety function. The Crosby replacement valves are designed and manufactured to ASME III Class 3 specifications. Since Crosby is now the owner of the Lonergan line of valves, this portion of the change is considered a component replacement with an acceptable substitute. Installation of the test ports will allow for relief valve testing without removing any portion of the CIA system from service. Installation and materials for the test connections meet Quality Class 1 (QC1) and ASME Code Group D piping requirements.

TER 99-0039 (SE 00-0047)

This change provided for the replacement of a hose with a hard pipe vent line from the reactor core isolation cooling system head spray area. This will allow plant operators to vent the piping without having to go into a high radiation zone.

Safety Evaluation Summary

The safety evaluation concluded that the new piping was designed and analyzed in accordance with the ASME III, Class 2 piping requirements. The new pipe supports were designed in accordance with Seismic Category 1 criteria. The new material is ASME III material which meets the code specifications for the piping. The new design will meet the code requirements for this application and will not affect the safety function of the vent line (which is to maintain pressure integrity). The new vent line piping has been designed and will be constructed to maintain the passive safety function of pressure integrity.

TER 99-0175 (SE 99-0070)

This change provided for the removal of the pressure reducing valves associated with main turbine hood spray controllers COND-PC-30A, COND-PC-30B and COND-PC-30C to allow more stable pressure control of the condensate water sprayed into the turbine exhaust steam entering the top of the main condenser when hood spray is in operation.

Safety Evaluation Summary

The safety evaluation concluded that the modification will not change or effect any operating or design characteristics. The change allows for the hood sprays to function properly and as originally intended and will not adversely impact any system, structure, component or safety analysis.

TER 99-0182 (SE 00-0053)

This change provided for the replacement of turbine service water supply and return piping to correct a pinhole leak and enhance the existing piping configuration by removing dead legs which accumulate debris and rust.

Safety Evaluation Summary

The safety evaluation concluded that the new piping and pipe fitting material meets the piping design specifications for the turbine service water system. The new configuration could not cause a different accident than those described in the safety analysis report because the piping and pipe supports are designed and constructed to the same specifications as the existing configuration.

TER 00-136701 (SE 00-0035)

This change provided for the replacement of high pressure core spray system valve HPCS-V-102 with a welded pipe plug to minimize or eliminate the occurrence of vibration fatigue failure in the weld connection.

Safety Evaluation Summary

The safety evaluation concluded that the modification will actually reduce the weld and pipe stresses at the critical location and will not impact operation of the instrument line or of the various in-line equipment. The modification will not result in a degradation of nuclear safety or in an unsafe configuration. The system will also continue to meet the design requirements of the ASME Code.

TER 00-136801 (SE 00-0034)

This change provided for the replacement of low pressure core spray system valve LPCS-V-83 with a welded pipe plug to minimize or eliminate the occurrence of vibration fatigue failure in the weld connection.

Safety Evaluation Summary

The safety evaluation concluded that the modification will actually reduce the weld and pipe stresses at the critical location and will not impact operation of the instrument line or of the various in-line equipment. The modification will not result in a degradation of nuclear safety or in an unsafe configuration. It system will also continue to meet the design requirements of ASME Code.

TMR 01-008 (SE 01-0020)

This change provided for the deactivation of the loose parts detection system (LPDS) by electrically isolating the system from the input power sources, and removing interconnections with the rod drive control system.

Safety Evaluation Summary

The safety evaluation concluded that deactivation of the LPDS does not yield an unreviewed safety question. The LPDS is neither an initiator nor mitigator of any previously analyzed accident or equipment malfunction. Furthermore, deactivation of LPDS is an industry initiative documented in BWROG Owners Group Topical Report NEDC 32975P, "Regulatory Relaxation for BWR Loose parts Monitoring Systems." The topical report was approved by the NRC on January 25, 2001.

TMR 01-025 (SE 01-0031)

This change provided for deactivation of electro-hydraulic operators (EHOs) for reactor closed cooling system valves RCC-TCV-71A, RCC-TCV-71B, RCC-TCV-71C, and RCC-TCV-72B to provide continuous full flow to the containment air system cooling units.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change has no impact on any previously analyzed accident or transient. Operation with RCC-TCV-71A, RCC-TCV-71B, RCC-TCV-71C, and RCC-TCV-72B in the full-open position is within the design capacity of the reactor closed cooling water system. All the temperature control valves are normally full-open during power operation. The flow of cooling water through the cooling coils of CRA-FC-2A and CRA-FC-2C will be maximized when the temperature control valves are completely open. This configuration will not affect the containment isolation function of the reactor closed cooling water system or its ability to remove heat from nonessential systems. Drywell temperatures will continue to be maintained within Technical Specification limits.

TMR 01-026 (SE 01-0035)

This change provided for the removal of thermocouple TG-TE-SMTIC2 and its associated mounting bracket from the Low Pressure (LP) No. 2 turbine inner cylinder of the main turbine generator. The mounting bracket was found broken loose and lodged in the LP No. 2 stationary blading when the turbine was disassembled during the 2001 maintenance and refueling outage. The thermocouple was not replaced when the LP No. 2 turbine was reassembled.

Safety Evaluation Summary

The safety evaluation concluded that the thermocouple is used only for monitoring and provides post-failure information for the main turbine generator. The turbine generator is not required for safe shutdown nor does it perform any safety function. Furthermore, the thermocouple provided no protective function for the turbine generator.

TMR 01-034 (SE 01-0042)

This change provided the temporarily disabling of the position indication of reactor feedwater system valves RFW-V-32A and RFW-V-32B to allow the installation of a welded cover onto the stuffing box and correct a recurring leak around the valve position indication shaft packing gland area.

Safety Evaluation Summary

The safety evaluation concluded that this will have no impact on plant system safety or operability. The installation of the welded cover on the stuffing box will enhance 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 are no control room operator actions that rely upon valve position indication during accident scenarios.

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-FSAR-99-098 (SE 99-0069)

This LDCN changed the position of Shift Technical Advisor (STA) in Section 13 of the FSAR to a "function" that can be staffed by either a dedicated individual (licensed or non-licensed) or by an individual filling a dual role as the Control Room Supervisor (CRS), Shift Manager (SM) or Shift Support Supervisor. When the STA qualified individual is filling a dual role as the CRS or SM, another Senior Reactor Operator (SRO) is required to be on-shift to provide independent oversight and emergency response support to the SM.

Safety Evaluation Summary

The safety evaluation concluded that the NRC has determined this is an acceptable way of meeting STA requirements. NRC Generic Letter 86-04 provides two methods for licensees to satisfy the requirements of NUREG 0737. Option 1 provides for elimination of the separate STA position by allowing licensees to combine one of the required (SRO) positions with the STA position into a dual-role (SRO/STA) position. Option 2 states that a licensee may continue to use an NRC-approved STA program while meeting licensed operator staffing requirements. Columbia Generating Station has previously used Option 2. These changes would allow the use of either Option 1 or 2.

LDCN-FSAR-00-043 (SE 01-0016)

This LDCN changed Section 9.1.1.3.2 of the FSAR by removing the statement that the bail handle yields at less than 1000 lb force. This statement implied that the yield strength was a design feature to protect the integrity of the new fuel storage racks. With the yield strength of the bail handles somewhat higher for the current fuel designs this feature no longer can be credited. In order to limit the force on the rack, the change was made to state that the jib crane is used to handle new fuel in the new fuel vault. In conjunction with this statement, any reference to the auxiliary hoist handling fuel in the vault was removed.

Safety Evaluation Summary

The safety evaluation concluded that these changes preserved the design feature of the new fuel vault by limiting the forces that could be applied to the rack and, therefore, there is no unreviewed safety question.

LDCN-FSAR-00-099 (SE 01-0002)

This LDCN corrected or added information to Sections 3.9.2.2 and 3.10 of the FSAR to improve the integrity of this data, which demonstrates seismic/dynamic qualification of safety related equipment.

Safety Evaluation Summary

The safety evaluation concluded that these changes did not result in an unreviewed safety question. While some reported allowable or resultant stress values changed, the components addressed remained seismically/dynamically qualified based on required stresses meeting allowable values.

LDCN-FSAR-00-101 (SE 00-0070)

This LDCN revised the wording describing the jockey fire pumps in Section F.2.4.1 of the FSAR and added additional NFPA code deviations to Table F.2-1.

Safety Evaluation Summary

The safety evaluation concluded that the changes do not constitute an unreviewed safety question. They do not adversely affect the ability to achieve and maintain post-fire safe shutdown or adversely impact associated plant systems or design/license bases.

LDCN-FSAR-01-002 (SE 01-0003)

This LDCN changed Section 12.5.2.d.2 of the FSAR to clarify that electronic dosimeters will be allowed for use when their alarm may not be heard except when they are used to fulfill the alarming dosimeter function described in Technical Specification 5.7. When used in accordance with the Technical Specifications, alternative methods of warning are required when the audible alarm may not be heard.

Safety Evaluation Summary

The safety evaluation concluded that this change did not constitute an unreviewed safety question, as it is only clarifying the use of alarm and alternative alarm devices.

LDCN-FSAR-01-016 (SE 01-0011)
LDCN-LCS-01-017

These LDCNs changed the FSAR and the Licensee Controlled Specifications (LCS) to reflect the use of the Westinghouse Multi Lift Tool (MTL) on the refuel bridge monorail hoist.

Safety Evaluation Summary

The safety evaluation concluded that the implementation of increased monorail hoist load set points for MTL operation, plus the installation and use of the MTL and temporary control rod blade exchange racks does not create an unreviewed safety question. The installation and use of these tools does not create conditions adverse to safety, nor do they create new accident scenarios, nor do they increase the probability or consequences of previously evaluated accidents in the FSAR.

LDCN-FSAR-01-042 (SE 01-0024)

This LDCN deleted the detailed description of the testing methods and the testing requirements for the containment atmosphere and suppression pool level excess flow check valves (EFCVs). It established the component's importance to safety designation and that the plant maintenance program is the proper program for controlling testing.

Safety Evaluation Summary

The safety evaluation concluded that the deletion of specific requirements to test and perform testing at a specified interval does not increase the probability of a malfunction of a component important to safety because these EFCVs do not have an active safety function to close. The periodic exercising of the EFCV does not improve, degrade, or affect its passive integrity ability in any way. Likewise, the periodic cycling at least once every 30 months of the EFCV does not improve, degrade, or effect its ability to remain open to provide accident mitigating sensors the ability to sense changing containment atmosphere or suppression pool parameters. Therefore, there is no increase in the probability of failure or consequence of a component important to safety as a result of this change.

LDCN-FSAR-01-049 (SE 01-0032)

This LDCN amended and revised detail from the FSAR description of the dryer and separator slings and the reactor pressure vessel head detensioning/tensioning process. These changes will allow future changes to the reactor pressure vessel head detensioning/tensioning processes and replacement of the wire rope slings with equivalent or higher rated Kevlar slings.

Safety Evaluation Summary

The safety evaluation concluded that the changes have no impact upon any Columbia Generating Station or FSAR requirement. There is no impact upon the probability of occurrence or the consequences associated with any accident or malfunction evaluated previously. The possibility of an accident or malfunction of a different type than any

evaluated previously is not created. There is no impact upon the margin of safety as defined in the basis of any Technical Specification.

The change does not alter the design, safe working load or safe load path requirements for any lifting device. All applicable NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," requirements will continue to be met for the dryer and separator sling lifting device assemblies and reactor building crane MT-CRA-2.

LDCN-LCS-01-030 (SE 01-0019)

This LDCN relaxed the maximum isolation time value specified for RCIC-V-63 located in LCS Table 1.6.1.3-1. RCIC-V-63 functions as an inboard primary containment isolation valve for the RCIC steam supply line. This will facilitate backseating of RCIC-V-63 if required due to packing degradation while maintaining RCIC-V-63 operable in support of RCIC system operability.

<u>Safety Evaluation Summary</u>

The safety evaluation concluded that there was no unreviewed safety question identified for the proposed increase in RCIC-V-63 maximum isolation time requirement due to back seating the valve to full open position. Valve RCIC-V-63 is being electrically backseated, thus positioning its wedge further open, and as such, increasing its closing time.

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

This LDCN revised the Offsite Dose Calculation Manual (ODCM) to: 1) replace references to Washington Public Power Supply System or Supply System with Energy Northwest; 2) replace a quotation from RFO 6.2.1.1 with a reference to RFO 6.2.1.1; 3) add a statement to the introduction which references the applicability statements of the technical specifications as applicable to the ODCM requirements for operability; 4) add a definition section which replaces the frequency notation and some of the definitions for consistency with the improved technical specifications; and 5) remove a footnote statement because driving regulation 10 CFR 50.36 was revised in a manner that eliminates the need for the statement.

<u>Safety Evaluation Summary</u>

The safety evaluation concluded that the changes were either editorial, clarified requirements or deleted references to obsolete requirements. The definitions and frequency notation table were taken verbatim from previously-evaluated and NRC-approved technical specification content. The applicability statements referenced are in the NRC-approved Standard Technical Specification NUREGs. The change maintains the level of radioactive effluent control required by 10 CFR 20, 40 CFR 190, 10 CFR

50.36(a), and 10 CFR 50, Appendix I, and does not adversely impact the accuracy or reliability of current effluent, dose, or setpoint calculations.

LDCN-TSB-00-104 (SE 00-0073)

This LDCN revised Technical Specification Bases Surveillance Requirement 3.6.1.3.8 to allow an increase in the pressure range for reactor instrument line excess flow check valve testing.

Safety Evaluation Summary

The safety evaluation concluded that the change did not result in an unreviewed safety question. The test pressure range will be increased from 85-300 psig to 85-1050 psig. The range was deemed conservative in comparison to the maximum pressure allowed by Technical Memorandum TM-2080, which was prepared to address reactor instrument line EFCV testing during reactor pressure vessel hydrostatic testing. The technical memorandum contains the requirements for the performance of the test.

LDCN-TSB-01-038 (SE 01-0001)

This LDCN added notes to the Technical Specifications Bases regarding movement of irradiated fuel in secondary containment. These notes state, "that handling a cask/canister loaded with spent fuel, after the canister is seal welded and leak tested, is not considered to be movement of irradiated fuel."

Safety Evaluation Summary

The safety evaluation concluded that Technical Specification Bases pertaining to secondary containment operability do not apply to handling of a loaded spent fuel canister within the reactor building once the canister is sealed and passes the leak test requirements. This is due to the fact that once a canister is properly loaded and sealed, handling of irradiated fuel ceases and handling of an NRC certified spent fuel canister begins. This position was communicated to the NRC by separate letter.

2.6.3 Miscellaneous Changes

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

Action Request 1556 (SE 01-0007)

This request provided for replacement valves for glycol system relief valves GY-RV-18A and GY-RV-18B (the existing models were no longer available). The new valves have a one-inch discharge port instead of a three-quarter-inch discharge port. The height of the discharge port is also different between the old and new valves and discharge piping required modification in order to use the new style relief valves.

Valves GY-RV-18A and 18B protect the tube side of the offgas dryer chiller heat exchangers and discharge to the Glycol Storage Tank GY-TK-2.

<u>Safety Evaluation Summary</u>

The safety evaluation concluded that the associated piping pressure rating and performance, as well as the function of relief valves GY-RV-18A and GY-RV-18B, will be unaffected by this change. Because the change will have no impact, there are no new accident scenarios introduced by the change. All plant systems and components required to mitigate the consequences of accidents previously evaluated will not be impacted by this modification.

Action Request 1566 (SE 01-0005)

This request provided for the cutting and capping of a cross connection pipe between the tower makeup system and the potable water system. This pipe had been isolated by closed valve TMU-V-102 because production of potable water was moved from the tower makeup system to the potable water distribution system. It has been determined that the single closed valve (TMU-V-102) has not been a sufficient cross connection control device in accordance with State of Washington drinking water regulations.

<u>Safety Evaluation Summary</u>

The safety evaluation concluded that none of the systems, equipment or components associated with the change are safety related, important to any safety function, or in direct support of any safety related function. There is no accident analysis that uses the systems, equipment or components. The change will not create the possibility of any new accident types or different malfunctions of equipment important to safety. The change also does not affect system operation or how it functions.

Action Request 1573 (SE 00-0071)

This request provided for revising the normal operating positions of turbine service water valves TSW-V-123A and TSW-V-123B from "open" to "closed" to isolate radwaste building automatic vent valves TSW-AV-2A and TSW-AV-2B.

Safety Evaluation Summary

The safety evaluation concluded that the turbine service water system is designed to provide cooling water to remove the maximum expected heat rejected from non-essential, non-safety related auxiliary equipment during normal plant operations. Valves TSW-AV-2A and TSW-AV-2B are mechanical devices and have no electrical interfaces with plant equipment. The operation of these vent valves through manual actions, as dictated by operating procedure during system startups and shutdowns, does not adversely impact the physical integrity or functioning of the turbine service water system nor does it create any new interfaces with other plant equipment. Accordingly, the activity will not initiate any previously evaluated design basis accidents or transients, nor will the activity impact any safety system, structure, or component.

Action Request 2089 (SE 01-0047)

This request provided for a permanent method of mitigating a steam leak from the packing gland on main steam system globe valve MS-V-706A by changing the valve to a gate valve.

Safety Evaluation Summary

The safety evaluation concluded that there will be no adverse impact to the integrity of the main steam system as a result of changing from a globe valve to a gate valve. All ASME Code requirements will be maintained. The change from a globe valve to a gate valve will not impact the ability of the main steam system piping to maintain its pressure integrity for accident mitigation. The manner in which the valve is operated and controlled will not change.

Clearance Order D-PMA-AC-51-001 (SE 01-0044)

This activity evaluated the acceptability of circulating water system air conditioning unit PMA-AC-51 being out of service for an extended period of time. The new replacement unit has been installed and connected except for the electrical connections. A technical evaluation request was initiated to address the difference in the current draw between the new unit and the old one.

Safety Evaluation Summary

The safety evaluation concluded that HVAC equipment in the circulating water system pump house is not considered in any accident scenario in the safety analysis report. This equipment has no safety function. Malfunction or failure of this equipment will not impair normal or emergency plant operation. Since this equipment is inoperative, it can be considered malfunctioned or failed and inoperable and will not adversely affect plant operation.

Caution Tag C-ANN-P602/A5 001 (SE 01-0043)

This activity provided for the disabling of the offgas sample flow high/low annunciator until problems with moisture in sample lines can be resolved.

Safety Evaluation Summary

The safety evaluation concluded that the offgas pretreatment radiation monitor is used to provide indication and alarm functions and initiates no protective functions. The monitor has no function which is credited in any accident analysis. The annunciator for offgas pretreatment sample flow high/low has no channel trip associated with it. Disabling the annunciator will not prevent the monitor from continuing to perform the monitoring function. During offgas system operations, pretreatment flow is established by differential pressure between steam jet air ejector outlet and main condenser pressures. This differential pressure does not vary significantly and average sample flow is relatively constant. Furthermore, Chemistry personnel perform a daily check of sample flow to ensure adequate flow.

Danger Tag 01-0784 (SE 01-00041)

This activity provided for deactivation of containment monitoring system wetwell sample valve PI-VX-269, a containment sample valve for sample rack CMS-SR-14, and closing its associated manual block valve (PI-V-X84B). Due to mounting constraints, the sample valve does not consistently close when required.

Safety Evaluation Summary

The safety evaluation concluded that deactivation of sample valve PI-VX-269 does not represent an unreviewed safety question. The function for monitoring will be maintained using the remaining drywell sample points on CMS-SR-14 and the drywell and wetwell sample points on CMS-SR-13. Since the manual valve will be closed, there is no adverse impact to the containment isolation function. The purpose of the sampling is to monitor the drywell and wetwell for hydrogen and oxygen concentrations after a LOCA. This monitoring information is used to determine when to remove the containment atmosphere control system from operation. The containment atmosphere control system is started

manually to assure a mixing of the gasses in containment. Deactivating this valve will not affect this function as there remains an adequate number of sample points to obtain a representative sample. In addition, the grab sample function of the post accident sampling system is available to supplement this sampling function.

Configuration Document Change Request 0000000857 (SE 01-0009)

This request revised a plant flow diagram to indicate that, for reactor core isolation cooling system valves RCIC-V-1 and RCIC-V-2, the Code Group classification is the same as that of RCIC turbine RCIC-DT-1 because the valves are part of the turbine skid. This change made the flow diagram consistent with FSAR Table 3.2-1.

Safety Evaluation Summary

The safety evaluation concluded that change to the flow diagram has no impact on the plant or the function and operation of the reactor core isolation cooling system. It simply makes the flow diagram consistent with the FSAR Table 3.2-1, which is in accordance with the original design.

Plant Procedure 10.3.21 (SE 01-0013)
Plant Procedure 10.3.22

The change revised the load path for the portable fuel transfer shield (cattle chute) to allow it to remain submerged in water while traversing from its storage location in the dryer/separator pool to its in-service location at the cavity/fuel pool channel. The old load path required the cattle chute to be out of the water from the time it was lifted from its storage location until it reached its in-service location. This change will reduce exposure to plant personnel working on the refuel floor during transfer of the cattle chute. The procedures include the option to travel along either the old or the new load path.

Safety Evaluation Summary

The safety evaluation concluded that the handling system (rigging, lifting lugs and MT-CRA-2) used for lifting the cattle chute complies with the requirements of NUREG-0612 for a single-failure-proof system. The movement of the cattle chute underwater is bounded by the movement of the steam separator which ensures that there will not be any adverse impacts with the head studs or pool curb. To the extent practical, both cattle chute load paths follow existing structural members.

Problem Evaluation Request 201-1461 (SE 01-0046)

This provided for the temporary repair, by means of the Furmanite process, of a steam leak from the packing gland on main steam system valve MS-V-706A. An attempt to backseat the valve failed to stop the leak. The repair was performed by drilling holes in

the valve bonnet and injecting Furmanite material into the packing area to mitigate the leak.

Safety Evaluation Summary

The safety evaluation concluded that the process would be controlled by Furmanite and Columbia Generating Station procedures, and an ASME, Section XI, repair plan. There will be no adverse impact to the pressure integrity of the valve. All ASME code requirements will be maintained and the process will not adversely affect reactor operations.

Work Order 01016880-12 (SE 01-0025)

This work order provided for an alternate power source for augmented quality motor control center E-MC-7CB which is taken from non-safety related low voltage switchgear E-SL-31/4C (spare) during plant shutdown to ensure continuous power was available for refueling activities during a divisional outage.

Safety Evaluation Summary

The safety evaluation concluded that the temporary modification of the power source to E-MC-7CB has no impact on the ability of the electrical system or any other system to perform its credited safety function. Overcurrent and overload protection is employed to ensure the effects of a motor control center branch circuit will not have any adverse effects on the non-safety related switchgear performance, as defined in design basis calculations. The motor control center and switchgear will continue to operate safely without affecting any other equipment.

Work Order 01016880-14 (SE 01-0026)

This work order provided for an alternate power source for fuel pool cooling pump FPC-P-1B from the safety related motor control center E-MCC-7BB Cubicle 8C (spare) during plant shutdown to assist in maintaining spent fuel pool temperature below 125 degrees Fahrenheit.

Safety Evaluation Summary

The safety evaluation concluded that temporary modification of the power source to FPC-P-1B has no effect on the ability of the fuel pool cooling system or any other system to perform its intended safety function. Overcurrent and overload protection is employed to ensure the effects of a motor control center branch circuit will not have any adverse effects on the non-safety related switchgear performance, as defined in design basis calculations. The motor control center and switchgear will continue to operate safely without affecting any other equipment.

Work Order 01016880 (SE 01-0028)

This work order provided for a power source for a temporary load center, which is taken from the non-safety related switchgear E-SL-11/3A, to support divisional outage activities.

Safety Evaluation Summary

The safety evaluation concluded that the temporary modification of a power source to a temporary load center has no adverse impact on the ability of the electrical system or any other system to perform its credited safety function. Overcurrent and overload protection is employed to ensure the effects of the addition of the load center will not have any adverse effect on the non-safety related switchgear performance, as defined in design basis calculations. The switchgear will continue to operate safely without affecting any other equipment.

Work Order 01016880-12 (SE 01-0034)

This work order provided for temporary power from a weld receptacle to non-safety related battery charger C1-7 during the time the normal power supply is out of service to support divisional outage activities.

Safety Evaluation Summary

The safety evaluation concluded that charger C1-7 is a Class 2 component that is not credited in any accident scenario or the Technical Specifications. The power to this charger will allow battery B1-7 to remain charged while the normal power supply is removed from service for preventive maintenance activities.

Work Order 01016880-16 (SE 01-0037)

This work order provided for temporary power to equipment drain system pump EDR-P-14B from MC-5D while normal power supply MC-6C is de-energized for preventive maintenance activities.

Safety Evaluation Summary

The safety evaluation concluded that EDR-P-14B plays no part in the mitigation of accidents or transients. Temporary power to EDR-P-14B will allow the pump to be used during the divisional outage activities.

Work Order 01016880-16 (SE 01-0038)

This work order provided for temporary power to reactor water cleanup system demineralizer desiccant pump RWCU-P-27 from MC-5D while normal power supply, MC-6C, is de-energized for preventive maintenance activities.

Safety Evaluation Summary

The safety evaluation concluded that RWCU-P-27 plays no part in the mitigation of accidents or transients. Temporary electrical power to RWCU-P-27 will allow liquid waste processing to continue while the normal power supply is out of service for preventive maintenance.

Work Order 01016880-42 (SE 01-0036)

This work order provided for temporary power to the radwaste building elevator from MC-3D while normal power supply MC-6B is de-energized for preventive maintenance activities.

Safety Evaluation Summary

The safety evaluation concluded that the radwaste building elevator plays no part in the mitigation of accidents or transients. Temporary power to the elevator will allow the elevator to continue to be used for outage activities.

2.7 Regulatory Commitment Changes (NEI Process)

This section is used for information pertaining to Regulatory Commitment Changes (RCC) pursuant to the NEI Guidelines for Managing NRC Commitment Changes. During 2001, there were no commitment changes that satisfied the NEI criteria for reporting. One commitment change was processed where the NRC staff was notified of the change under separate correspondence.

Appendix A

Annual Personnel Radiation Exposure Work and Job Function Report

Reg. Guide 1.16 Work and Job Function Report for CY 2001

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	8	0	2	2.966	0.125	0.937
Operating Personnel	4	0	0	4.078	0.000	0.000
Health Physics Personnel	18	1	14	3.152	0.125	1.510
Supervisory Personnel	0	0	0	0.206	0.000	0.000
Engineering Personnel	1	0	2	0.129	0.084	0.494
Routine Maintenance						
Maintenance Personnel	110	4	185	26.935	0.896	53.717
Operating Personnel	29	0	0	5.003	0.000	0.048
Health Physics Personnel	12	0	12	5.058	0.023	6.697
Supervisory Personnel	5	0	0	0.683	0.001	0.000
Engineering Personnel	5	3	26	0.577	0.336	4.539
Inservice Inspection						
Maintenance Personnel	1	0	61	0.812	0.002	30.112
Operating Personnel	0	0	0	0.490	0.000	0.000
Health Physics Personnel	0	0	1	0.181	0.000	0.226
Supervisory Personnel	0	0	0	0.000	0.000	0.000
Engineering Personnel	3	1	26	0.639	0.393	13.352
Special Maintenance*						
Maintenance Personnel	6	0	29	5.186	0.156	7.370
Operating Personnel	0	0	0	0.604	0.000	0.000
Health Physics Personnel	1	0	0	0.673	0.038	0.156
Supervisory Personnel	0	0	0	0.223	0.000	0.000
Engineering Personnel	0	1	3	0.068	0.232	0.294
Waste Processing						
Maintenance Personnel	1	1	2	1.068	0.462	0.920
Operating Personnel	0	0	1	0.000	0.000	0.102
Health Physics Personnel	0	0	0	0.134	0.010	0.013
Supervisory Personnel	0	0	0	0.000	0.000	0.000
Engineering Personnel	0	0	0	0.000	0.000	0.000
Refueling						
Maintenance Personnel	20	1	8	11.215	0.102	2.346
Operating Personnel	0	0	0	0.346	0.000	0.000
Health Physics Personnel	1	0	3	0.488	0.000	0.749
Supervisory Personnel	4	1	0	0.688	0.211	0.000
Engineering Personnel	0	7	2	0.000	0.891	0.285
TOTAL						
Maintenance Personnel	146	6	287	48.182	1.743	95.402
Operating Personnel	34	0	1	10.521	0.000	0.150
Health Physics Personnel	32	1	29	9.686	0.196	9.351
Supervisory Personnel	9	1	0	1.800	0.212	0.000
Engineering Personnel	8	12	60	1.413	1.936	18.964
Grand Total	**229**	**20**	**377**	**71.602**	**4.087**	**123.867**

Total number of personnel receiving >100 mrem = 626   Total man-rem for personnel receiving > 100mrem = 199.556

Report produced from electronic dosimeter data

*Special Maintenance:

- RRC-P-1B seal replacement and associated support work (forced outage)
- MMS sample rack installation for NMC/HWC (BDC 94-0274-3D)
- LPCS-V-83 removal & modification (TER 136801)
- HPCS-V-102 removal & modification (TER 136701)
- SFP cask area modifications (ISFSI)
- ECP/LPRM replacement (BDC 94-0274-4F)
- Replace TSW piping to/from PSR-SR-8, 9, & 11
- SRM/IRM conduit 7SRM13-1 & cable replacement
- MS-V-GV/1 DEH leak repair