ENERGY NORTHWEST

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

February 27, 2003
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U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397 ANNUAL OPERATING REPORT 2002**

Dear Sir or Madam:

The annual operating report for calendar year 2002 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,

RL Webring
Vice President, Nuclear Generation
Mail Drop PE04

Attachment

cc: EW Merschoff - NRC RIV
BJ Benney - NRC NRR
REIRS Project Manager - NRC NRR
NRC Sr. Resident Inspector – 988C
RN Sherman – BPA/1399
TC Poindexter - Winston & Strawn

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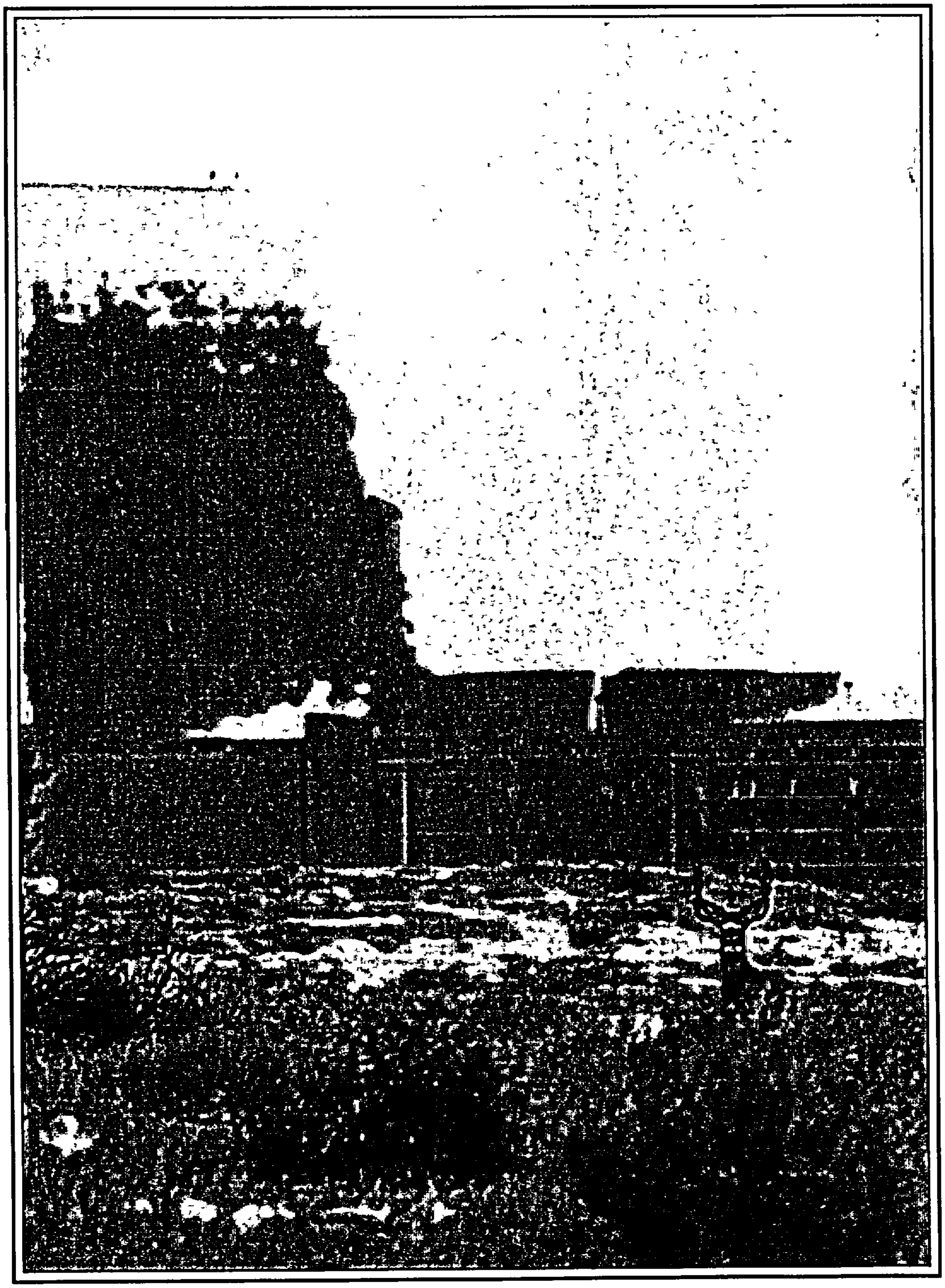
Columbia Generating Station Annual Operating Report 2002



COLUMBIA GENERATING STATION

ANNUAL OPERATING REPORT

2002

DOCKET NO. 50-397

FACILITY OPERATING LICENSE NO. NPF-21

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

Columbia Generating Station
Annual Operating Report
2002

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1.0 Reporting Requirements

The reports in this document 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-Appendix A Technical Specifications," Revision 4, August 1975; and 5) the guidance contained in NEI 99-04, "Guidelines for Managing NRC Commitment Changes," Revision 0, July 1999.

Technical Specification 5.6.1 requires that an occupational radiation exposure report be submitted in accordance with 10 CFR 50.4 by April 30 of each year. The report is required to include 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 and 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 assignments 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 the aggregate, at least 80 percent of the total 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).

2.0 Summary of Plant Operations

This section contains a narrative summary of the operating experience at Columbia Generating Station during calendar year 2002. This information is provided in accordance with Regulatory Guide 1.16, Section C.1.b.(1).

There was no planned outage scheduled during 2002 as the plant is operating on a 24-month fuel cycle. The next maintenance and refueling outage is scheduled to begin in May 2003.

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

On February 14, 2002, the plant entered a forced outage to repair safety-related circuit breakers (see section 3.0). Corrective actions consisted of performing maintenance on active safety-related 4160 and 6900-volt cubicle switchgear. The plant returned to the grid on February 24, 2002 after approximately 231.5 hours off-line.

Planned power reductions as low as 45% occurred at the request of the Bonneville Power Administration (BPA) to support regional power system management within the Federal Columbia River Power System. These requests occurred during the months of April through July.

On September 25, 2002, a ventilation fan failure resulted in power being lost to an electrical load distribution center that in turn caused the station to reduce power due to a partial loss of feedwater heating (see section 3.0). Corrective actions included the resolution of fuse coordination issues and restoring power to the distribution center. Station power was further reduced to recover the feedwater heaters and full power operation was resumed on September 26. About 17.0 hours of operation at reduced power occurred as a result of this event.

On November 21, 2002, the station exceeded the previous generation run record of 270 days. The new record run, which began on February 24, 2002, continued through December 31, 2002. The station ended the year with an availability factor of approximately 97%.

3.0 Outages and Forced Reductions in Power

This section contains information 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 due to an outage (scheduled or forced) or a component failure or other condition that requires that the load on the unit be reduced for corrective action immediately or up to and including the very next weekend (Regulatory Guide 1.16, Section C.1.b.(2)).

February 14 – 24, 2002 (approximately 231.5 generator-off-line hours)

Due to problems with the operation of the DG-2 emergency diesel generator output breaker, the station was manually shutdown on February 14, 2002. Investigation and evaluation revealed that the circuit breaker had closed but the mechanism-operated cell switch assembly had failed to change state as expected. Corrective actions consisted of performing maintenance on active safety-related 4160 and 6900-volt cubicle switchgear and enhancing an existing circuit breaker program to provide guidance on breaker and cell maintenance (reference LER 2002-001-00). On February 24, 2002, the station was returned to service.

September 25 – 26, 2002 (approximately 17.0 hours at reduced power)

On September 25, 2002, the plant was operating at 100% power when the motor for turbine building ventilation exhaust system fan TEA-FN-1C seized which caused the feeder breaker for non-safety-related 480-volt motor control center E-MC-3A to open. Subsequent investigations found that the inboard bearing on the TEA-FN-1C motor had failed causing an electrical fault in the motor windings. Due to a miscoordination between the fan motor fuses and the overcurrent protective device supplying power to E-MC-3A, the fuses did not clear and the circuit breaker supplying E-MC-3A tripped. The loss of E-MC-3A caused a partial loss of reactor feedwater heating that resulted in a reduction of station power. Corrective actions included the resolution of the fuse coordination issues, strengthening of vibration analysis and monitoring of non-class 1E rotating equipment, and performing a risk assessment of a loss-of-power to each identified MCC with miscoordinated overcurrent protective devices. Power was reduced further to recover the feedwater heaters and full power operation was resumed on September 26, 2002.

October 7 – 8, 2002 (approximately 26.2 hours at reduced power)

On October 7, 2002, power was reduced to about 90% in response to a failure of the 400 HZ power supply for reactor recirculation system adjustable speed drive channel 1B/2. Immediate corrective action consisted of replacing the failed power supply board. The following day power was reduced to about 65% to recover the channel and the station returned to full power.

4.0 Radiation Exposure

The following annual work and job function report contains information pertaining to occupational radiation exposure. This information is included pursuant to Technical Specification 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.

Columbia Generating Station

Reg. Guide 1.16 Work and Job Function Report for CY 2002

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	3 79	0 00	0 05	1 555	0 000	0 007
Operating Personnel	0.37	0 00	0 00	0 408	0 000	0 000
Health Physics Personnel	0 14	0 00	0 01	0 086	0 025	0 006
Supervisory Personnel	0 00	0 00	0 00	0 000	0 000	0 000
Engineering Personnel	0 06	0 05	0 00	0 028	0 001	0 000
Routine Maintenance						
Maintenance Personnel	43 85	0 20	12 34	6 154	0 036	3 613
Operating Personnel	21 23	0 00	0 00	2 824	0 000	0 000
Health Physics Personnel	23.10	1 94	0 95	3 356	0 382	0 111
Supervisory Personnel	0 21	0 00	0 00	0 036	0 000	0 000
Engineering Personnel	1.59	0 97	1 00	0 191	0 160	0 112
Inservice Inspection						
Maintenance Personnel	0 00	0 00	0 48	0 000	0 000	0 194
Operating Personnel	0 00	0 00	0 00	0 000	0 000	0 000
Health Physics Personnel	0 00	0 00	0 01	0 008	0 000	0 011
Supervisory Personnel	0 00	0 00	0 00	0 000	0 000	0 000
Engineering Personnel	0 04	0 23	0 00	0 003	0 022	0 000
Special Maintenance*						
Maintenance Personnel	1 17	0 00	9 15	1 042	0 000	1 836
Operating Personnel	0 14	0 00	0 00	0 094	0 000	0 000
Health Physics Personnel	0.50	0 03	0 03	0 248	0 045	0 012
Supervisory Personnel	0 00	0 00	0 00	0 000	0 000	0 000
Engineering Personnel	0 31	0 42	0 00	0 039	0 048	0 000
Waste Processing						
Maintenance Personnel	1.90	0 80	1.00	0 388	0 243	0 151
Operating Personnel	0 15	0 00	0 00	0 087	0 000	0 000
Health Physics Personnel	2 22	0 03	0 01	0 630	0 047	0 007
Supervisory Personnel	0 00	0 00	0 00	0 000	0 000	0 000
Engineering Personnel	0 00	0 34	0 00	0 000	0 035	0.000
Refueling						
Maintenance Personnel	0.10	0 00	0 00	0 016	0 000	0 000
Operating Personnel	0 09	0 00	0 00	0 058	0 000	0 000
Health Physics Personnel	0 04	0 00	0 00	0 018	0 000	0 000
Supervisory Personnel	0 00	0 00	0 00	0 000	0 000	0 000
Engineering Personnel	0 00	0 00	0 00	0 000	0 000	0 000
TOTAL						
Maintenance Personnel	50 81	1 00	23 02	9 155	0 279	5 801
Operating Personnel	21 98	0 00	0 00	3 471	0 000	0 000
Health Physics Personnel	26 00	2 00	1 00	4 346	0 499	0.147
Supervisory Personnel	0 21	0 00	0 00	0 036	0 000	0 000
Engineering Personnel	2 00	2 01	1 00	0 261	0 266	0 112
Grand Total	**101.00**	**5 01**	**25 02**	**17.269**	**1.044**	**6.060**

Total number of personnel receiving >100 mrem = 131

Total man-rem for personnel receiving > 100 mrem = 24 373

Report produced from electronic dosimeter data

*Special Maintenance:
- Install and paint platform to access EDR-V-19 & 20
- Install conduit and cable on Reactor 441' Railway Bay
- Furmanite repair on SS-V-49E
- Hard pipe vent line on RHR-V-632 to equipment drain
- Sheet metal kick plate installation and repair
- Load spent fuel into casks #1, 2, 3, 4 & 5 (ISFSI)

5.0 Fuel Performance

This section contains information relative to fuel integrity pursuant to FSAR Section 4.2.4.3, "Post-Irradiation Surveillance," and Regulatory Guide 1.16, Section C.1.b.(4).

5.1 Fuel Integrity

No fuel failure was identified during the calendar year 2002 portion of Cycle 16. This conclusion was based on readings of offgas radioactivity from the pre-treatment process radiation monitoring system.

The sum-of-six readings have stayed considerably below the INPO threshold for fuel failures of 300 microCi/sec (mostly in the range below 200 microCi/sec). The values for the Xe-133/Xe-135 and Xe-138/Xe-133 activity ratios have been within the range for intact fuel.

5.2 Fuel Corrosion Update

On October 2, 2002, during a meeting at NRC headquarters, Energy Northwest provided the NRC staff with an update on the status of increased fuel corrosion indications we initially discovered during fuel inspections in 2001 (Refueling Outage R-15 - Spring 2001). Follow-up inspections performed to date include visual examinations, measurements of fuel oxide thickness, and obtaining and analyzing fuel crud samples. Those results led to the following status of the extent of fuel corrosion observed:

- High levels of nodular corrosion
- Some spallation observed on 4th burned bundles
- Accelerated copper deposition occurring
- Thicker oxide (corrosion) and crud layers than found during previous Columbia inspections

An operability determination was performed that conservatively predicts the extent of fuel cladding oxide growth to the end of Cycle 16. The operability determination demonstrates that adequate margin to design limits continues to exist.

The ongoing root cause investigation has found that, during a specific period late in Cycle 15, there were strong trends and/or anomalies in several water chemistry parameters. A reactor water conductivity anomaly coincident with strong trends in some short-lived activation products like Tc-99m and Cr-51, along with trends in other reactor water chemistry indicators, suggests a possible disruption of the fuel crud layers during this period. Some substance or substances were allowed into the reactor water and either caused a crud layer disruption, caused other chemical reactions near the fuel surfaces, or deposited directly on the fuel, initiating nodule formation and accelerated corrosion. While the exact mechanisms and substance or substances responsible are not known, there is evidence that the main condenser leak, in conjunction with condensate filter demineralizer performance problems during the specific period, allowed the substance or substances through to the reactor water. The water chemistry trends and anomalies have not reappeared since the specific period in Cycle 15. Therefore, it is not expected that accelerated corrosion continued in Cycle 16. Several corrective actions are in place to preclude a recurrence of the Cycle 15 event.

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

This section contains information relative to non-significant changes and errors in Emergency Core Cooling System (ECCS) cooling performance models pursuant to 10 CFR 50.46.

Westinghouse methodology is applied to the Columbia Generating Station core and was used to license SVEA-96 fuel. No errors were discovered in the Westinghouse ECCS LOCA analysis model or in the application of the model during 2002.

7.0 10 CFR 50.59 Changes, Tests, and Experiments

This section contains summaries of safety evaluations (SE) for activities implemented during 2002 that were assessed pursuant to 10 CFR 50.59 requirements.

Columbia Generating Station implemented the revised 10 CFR 50.59 rule in August 2001. In 2002, there were no evaluations performed under the new rule. There were evaluations performed under the old rule that were implemented in 2002. 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.

7.1 This section contains information pertaining to implemented plant modifications [Basic Design Changes (BDCs)] and is included pursuant to 10 CFR 50.59.

BDC 93-0037-15 (SE 00-0009)

This modification provided for the installation of a permanent platform in the reactor building to provide access to equipment drain system valves EDR-V-19 and EDR-V-20. The platform replaced a long-standing scaffold that had been used by plant personnel to perform testing and surveillances on the valves.

<u>Safety Evaluation Summary</u>

The safety evaluation concluded that the platform performs a passive function of providing access to EDR-V-19 and EDR-V-20. Therefore, the proposed platform will not have any adverse impacts or interactions with important to safety systems or components.

Temporary support 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.

BDC 93-0037-16 (SE 00-0017)

This modification provided for the installation of a permanent platform in the reactor building to provide access to residual heat removal system valve RHR-V-27A. The platform replaced a long-standing scaffold that had been used by plant personnel to perform testing and surveillances on the valve.

Safety Evaluation Summary

The safety evaluation concluded that the platform performs a passive function of providing access to RHR-V-27A. Therefore, the proposed platform will not have any adverse impacts or interactions with important to safety systems or components.

Temporary support 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.

BDC 93-0058 (SE 00-0006)

This modification provided for deactivation and removal of the water treatment system located in the General Services Building (Yakima Building). The system was obsolete and potable and demineralized water is supplied by other systems.

Safety Evaluation Summary

The safety evaluation concluded that the water treatment system does not provide any safety-related function. The system is Quality and Seismic Class 2. There is no interaction with systems, structures, or components that are important to safe operation and safe shutdown of the plant.

There are also no postulated accidents or transients impacted by the modification. The modification has no interface with any system, structure, or components described in the accident analyses. Therefore, removal of the system will not increase the probability of occurrence or consequence of an accident previously evaluated.

BDC 96-0132 (SE 96-0043)

This modification provided for the removal of reactor building electronics room air conditioning unit RRA-AC-16. The air conditioning unit was originally intended to provide additional comfort cooling to the electronics room.

Safety Evaluation Summary

The safety evaluation concluded that removal of the non-safety-related air conditioning unit does not affect any safety-related process equipment or systems. The normal reactor building HVAC system provides adequate cooling and air handling to the room. Removal of the unit will reduce the overall heat load internal to the reactor building. The unit is not relied upon in any accident analysis. Therefore, the modification will not increase the probability of occurrence or consequence of an accident previously evaluated.

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

LDCN-FSAR-98-078 (SE 00-0030)

This LDCN provided for several changes to the FSAR that were the result of a re-evaluation of the station flooding analysis. Proposed changes pertaining to: 1) an unisolable drain line between the reactor core isolation cooling and control rod drive/condensate system pump rooms; and 2) minimal acceptable safe shutdown equipment necessary for flood mitigation were determined to represent an unreviewed safety question and required prior NRC approval to implement. Approval was granted by Amendment 176 to Facility Operating License NPF-21 (letter dated June 19, 2002, J Hickman (NRC) to JV Parrish (Energy Northwest), "Columbia Generating Station - Issuance of Amendment Regarding TAC No. MB1777").

The remaining changes proposed by LDCN-FSAR-98-078 were determined not to represent an unreviewed safety question and are summarized as follows: 1) changing the maximum emergency core cooling system (ECCS) pump room flooding elevation above the foundation mat in FSAR Sections 3.4, 3.8 and 10.4; 2) clarifying the results of passive ECCS failures in FSAR Section 6.3; and 3) revising the list of moderate energy fluid systems outside primary containment in FSAR Table 3.6-2.

<u>Safety Evaluation Summary</u>

The safety evaluation concluded the following:

1) Changing the maximum flooding elevation in the ECCS pump rooms from 46.4 feet (formerly the worst case flood level from a condensate line break) to 44 feet in FSAR Sections 3.4, 3.8, and 10.4 was acceptable because pipe stress calculations for the condensate line routed in the low pressure core spray system determined that a crack in this section of piping need not be postulated. The design flood level of 44 feet is also consistent with design basis reactor building pump room wall parameters and the maximum flood height that occurs from a condensate pipe break in the residual heat removal (RHR) system "C" pump room.

2) Revision to FSAR Section 6.3 clarifies the results of a long term (after 24 hours) passive single failure which is postulated to occur during a LOCA. Analysis indicates that, during a LOCA, the worst case passive single failure is a 23 gpm RHR system pump seal failure. This failure will not affect ECCS operation since the resulting flood heights in RHR-A and RHR-B pump rooms are slow developing and are less than the emergency operating procedure safe flood levels for each room. After flood detection, operator action will isolate the source of the leak prior to the water level affecting ECCS operation.

3) FSAR Table 3.6-2 was revised to update the list of moderate energy fluid systems outside primary containment to accurately reflect the fluid systems outside primary containment where potential moderate energy pipe cracks can occur.

Accordingly, it was determined that these changes will neither increase the frequency nor consequences of accidents, transients or equipment malfunctions previously evaluated in the safety analysis report.

8.0 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.

RCC-110953-00 (Backup Scram Valve Testing)

The original commitment was that the backup scram valves would be tested at 18-month intervals as part of the reactor protection system (RPS) logic system functional test and that the tests would verify the operability of each backup scram valve (reference letter GO2-83-1076, dated November 18, 1983, GC Sorenson to NRC, Response to Generic Letter 83-28).

The commitment was made in response to Generic Letter 83-28, Item 4.5.2. The NRC relied upon the original commitment as the basis for a safety decision in an SER. The commitment that the backup scram valves would be tested at 18-month intervals as part of the RPS logic system functional test led NRC staff to conclude that the backup scram valve test interval met the staff position.

The commitment was revised to support the 24-month fuel cycle. The backup scram valve testing frequency was revised from 18 months to 24 months to coincide with the current RPS logic system functional test (reference Technical Specification SR 3.3.1.1.14). Based upon redundancy of actuation devices, on-line testing of scram valves and diversity of the scram system, a high reliability of the scram system is ensured without requiring on-line testing of the backup scram valves.

RCC-140312-00 (Effectiveness Assessment of Industry Events Training)

The original commitment was to revise our internal processes to enhance the effectiveness of industry events training by July 1, 1993 (reference letter GO2-93-067, dated March 22, 1993, JV Parrish to NRC, NRC Inspection of Facility Training, Action Status).

The commitment has been deleted. Attached to the referenced letter was an internal Energy Northwest memorandum that defined a new assessment process which included specific guidance for evaluating the effectiveness of industry events training, the specifics of which are no longer applicable. The industry events program at Energy Northwest has matured and has been broadened and integrated into station departments with strong management support. Periodic assessments of the program are performed, resulting in continual program improvements.

RCC-185190-00 (Augmented RFW Nozzle Inspections)

The original commitment was to perform augmented reactor feedwater (RFW) system nozzle inspections in accordance with the Energy Northwest response to NUREG-0619 (reference letter GO2-82-036, dated January 13, 1982, GD Bouchey to NRC, WNP-2 Response to NUREG-0619).

The commitment was revised to implement the latest NRC approved industry recommendations. The revised commitment is to perform augmented RFW nozzle inspections in accordance with GE-NE-523-A71-0594-A, Revision 1, "Alternate BWR Feedwater Nozzle Inspection Requirements." This alternative for performing augmented RFW nozzle inspections was approved by NRC in an SER dated March 10, 2000, "Final Safety Evaluation of BWR Owner's Group Alternate Boiling Water Reactor (BWR) Feedwater Nozzle Inspection (TAC No. MA6787)." The alternative method includes improved inspection techniques by accepting ultrasonic testing (UT) as the basis to eliminate supplemental liquid penetrant testing of the inside radius of the reactor pressure vessel feedwater nozzles. This method also allows for the lengthening of the time interval between routine ultrasonic testing and reduces the inspection area of the inner radius of the RPV feedwater nozzles.