March 13, 1992

Docket No. 50-397

Mr. G. C. Sorensen, Manager
Regulatory Programs
Washington Public Power Supply System
3000 George Washington Way
P.O. Box 968
Richland, Washington 99352

Dear Mr. Sorensen:

SUBJECT: ISSUANCE OF AMENDMENT FOR THE WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2 (TAC NO. M82668)

The Commission has issued the enclosed Amendment No. 100 to the Facility Operating License No. NPF-21 for WPPSS Nuclear Project No. 2. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated January 21, 1992, and supplemented by letter dated February 14, 1992.

The amendment revises the technical specifications to more accurately define the acceptance criteria for the capacity of the blowers in the main steam isolation valve leakage control system.

A copy of the related Safety Evaluation is also enclosed. A notice of issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,
Original signed by H. Rood for
Patricia L. Eng, Project Manager
Project Directorate V
Division of Reactor Projects III/IV/V Office of
Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 100 to NPF-21
2. Safety Evaluation

cc w/enclosures: *See previous concurrences
See next page

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OFFICIAL DOCUMENT NAME: WNP82668.AMD

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Ted Quay for
Patricia L. Eng, Project Manager
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OFC	PDV/LA	PDV/PM	OGC	BC/SPLB	PDV/D
NAME	DFoster	PEng	J Hull	CMcCracken	TQuay
DATE	3/9/92	3/9/92	3/11/92	3/10/92	3/9/92

OFFICIAL DOCUMENT NAME: WNP82668.AMD

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 13, 1992

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H. Rood for

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Office of Nuclear Reactor Regulation

Enclosures:
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See next page

Mr. G. C. Sorensen
Washington Public Power Supply System

WPPSS Nuclear Project No. 2
(WNP-2)

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

DOCKET NO. 50-397

NUCLEAR PROJECT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 100
License No. NPF-21

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Washington Public Power Supply System (licensee) dated January 21, 1992, and supplemented by letter dated February 14, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-21 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 100 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore R. Quay, Director
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 13, 1992

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 100 TO FACILITY OPERATING LICENSE NO. NPF-21

DOCKET NO. 50-397

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
xiv	xiv
3/4 6-7	3/4 6-7
B 3/4 6-2	B 3/4 6-2
B 3/4 6-3	B 3/4 6-3
B 3/4 6-4	B 3/4 6-4
B 3/4 6-5	B 3/4 6-5

BASES

SECTION | PAGE

INSTRUMENTATION (Continued)

BASES

SECTION PAGE

CONTAINMENT SYSTEMS

MSIV LEAKAGE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.4 Two independent MSIV leakage control system (LCS) subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With one MSIV leakage control system subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 Each MSIV leakage control system subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1. Starting the blower(s) from the control room and operating the blower(s) for at least 15 minutes.

   2. Energizing the heaters and verifying the current to be ± 10% of rated current for each heater.

b. During each COLD SHUTDOWN, if not performed within the previous 92 days, by cycling each depressurizing valve and steam isolation valve through at least one complete cycle of full travel.

c. At least once per 18 months by:

   1. Performance of a functional test which includes simulated actuation of the subsystem throughout its operating sequence, and verifying that each automatic valve actuates to its correct position and the blower starts.

   2. Verifying that the blower develops at least the -17" H_2O at the blower suction, with 30 cfm of dilution flow.

d. By verifying the flow, pressure and temperature instrumentation to be OPERABLE by performance of a:

   1. CHANNEL FUNCTION TEST at least once per 31 days, and
   2. CHANNEL CALIBRATION at least once per 18 months.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.5.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the structural integrity of the primary containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 The structural integrity of the exposed accessible interior and exterior surfaces of the primary containment shall be determined during the shutdown for each Type A containment leakage rate test by a visual inspection of those surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.5.2 Reports Any abnormal degradation of the primary containment structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.1. This report shall include a description of the condition of the primary steel containment, the inspection procedure and the corrective actions taken.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure of 34.7 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50 with the exception of exemptions granted for main steam isolation valve leak testing and testing the airlocks after each opening.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment.

3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main steamline isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR Part 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIVs such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the MSIVs when isolation of the primary system and containment is required.

MSIV LEAKAGE CONTROL SYSTEM (Continued)

Design specifications require the system to accommodate a leak rate of five times the Technical Specification leakage allowed for the MSIVs while maintaining a negative pressure downstream of the MSIVs. The allowed leakage value per each valve is 11.5 scfm, or a total of 230 scfh (3.8 scfm).[a] When corrected for worst case pressure, temperature and humidity expected to be seen during surveillance testing conditions, the flow would never exceed an indicated value (uncorrected reading from local flow instrumentation) of 5 cfm. The 30 cfm acceptance criterion provides significant margin to this design basis requirement and provides a benchmark for evaluating long term blower performance. The Technical Specification limit for pressure of -17" H_2O W.C. was also established based on a benchmark of the installed system performance capability. This -17" H_2O W.C. provides assurance that the negative pressure criterion can be met.

3/4.6.1.5 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 34.7 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression chamber internal pressure ensure that the containment peak pressure of 34.7 psig does not exceed the design pressure of 45 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 2 psid. The limit of 1.75 psig for initial positive containment pressure will limit the total pressure to 34.7 psig which is less than the design pressure and is consistent with the safety analysis.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during LOCA conditions and is consistent with the safety analysis.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

The 24-inch and 30-inch drywell and suppression chamber purge supply and exhaust isolation valves are required to be sealed closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves sealed closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the purge system. To provide assurance that the 24-inch and 30-inch valves cannot be inadvertently opened, they are sealed closed in accordance with Standard Review Plan 6.2.4, which includes mechanical devices to seal or lock the valve closed or prevent power from being supplied to the valve operator.

(a) Letter, G02-75-238, dated August 18, 1975, NO Strand (SS) to OD Parr (NRC), "Response to Request for Information Main Steam Isolation Valve Leakage Control System"

DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM (Continued)

The use of the drywell and suppression chamber purge lines is restricted to the 2-inch purge supply and exhaust isolation valves since, unlike the 24-inch and 30-inch valves, the 2-inch valves will close during a LOCA or steam line break accident and therefore the SITE BOUNDARY dose guidelines of 10 CFR Part 100 would not be exceeded in the event of an accident during PURGING operations. The design of the 2-inch purge supply and exhaust isolation valves meets the requirements of Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations."

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of those valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2. DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure of 45 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1020 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss-of-coolant accident, the pressure of the liquid must not exceed 45 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant and to be considered is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in this specification, containment pressure during the design basis accident is approximately 34.7 psig which is below the design pressure of 45 psig. Maximum water volume of 128,827 ft^3 results in a downcomer submergence of 12 ft and the minimum volume of 127,197 ft^3 results in a submergence approximately 4 inches less. The majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate. The maximum temperature at the end of the blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

DEPRESSURIZATION SYSTEMS (Continued)

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

Under full power operating conditions, blowdown from an initial suppression chamber water temperature of 90°F results in a water temperature of approximately 135°F immediately following blowdown which is below the 200°F used for complete condensation via quencher devices. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus, there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling, there is no dependency on containment overpressure for post-LOCA operations.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak bulk temperature of the suppression pool is maintained below 200°F during any period of relief valve operation with sonic conditions at the discharge exit for quencher devices. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety/relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety/relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety/relief valve to assure mixing and uniformity of energy insertion to the pool.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures for those isolation valves designed to close automatically that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 VACUUM RELIEF

Vacuum relief breakers are provided to equalize the pressure between the suppression chamber and drywell and between the reactor building and suppression chamber. This system will maintain the structural integrity of the primary containment under conditions of large differential pressures.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. There are nine pairs of valves to provide redundancy and capacity so that operation may continue indefinitely with no more than two pairs of vacuum breakers inoperable in the closed position.

3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The reactor building and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a vacuum in the reactor building with the standby gas treatment system once per 18 months, along with the surveillance of the doors, hatches, dampers, and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

The OPERABILITY of the standby gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting SITE BOUNDARY radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Continuous operation of the system with the heaters OPERABLE for 10 hours during each 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

The OPERABILITY of the systems required for the detection and control of hydrogen gas ensures that these systems will be available to maintain the hydrogen concentration within the primary containment below its flammable limit during post-LOCA conditions. Either drywell and suppression chamber hydrogen recombiner system is capable of controlling the expected hydrogen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. The hydrogen control system is consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," September 1976.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 100 TO FACILITY OPERATING LICENSE NO. NPF-21

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

NUCLEAR PROJECT NO. 2

DOCKET NO. 50-397

1.0 INTRODUCTION

By letter dated January 21, 1992, and supplemented by letter dated February 14, 1992, Washington Public Power Supply System submitted a request for changes to the Technical Specifications (TS) for Nuclear Project No. 2. The proposed changes would revise the technical specifications to more accurately define the acceptance criteria for the capacity of the blowers in the main steam isolation valve leakage control system.

2.0 EVALUATION

On January 16, 1992, the Washington Public Power Supply System (WPPSS) requested relief from WPPSS Nuclear Project No. 2 (WNP-2) Technical Specification (TS) Section 4.6.1.4.c, "Containment Systems - MSIV Leakage Control System Surveillance Requirements." The TS required that the main steam isolation valve (MSIV) leakage control system blowers be demonstrated operable at least once every 18 months by verifying that the leakage control system blower develops at least 17 inches of water vacuum at a flow rate of 30 standard cubic feet per minute (scfm). WPPSS stated that the surveillance could not be successfully conducted with the existing acceptance criteria due to building configuration and requested temporary relief from the surveillance requirement acceptance criteria.

On January 17, 1992, the NRC granted relief from the surveillance test acceptance criteria with the understanding that the licensee would submit a request to amend the technical specifications in a timely manner.

On January 21, 1992, the licensee submitted a request for amendment to the technical specifications revising the surveillance test acceptance criteria to require that the MSIV leakage control system blower develop at least 17 inches of water vacuum at a flow rate of 30 cubic feet per minute (cfm). Essentially, the licensee requested lifting of the requirement to correct the measured flow rate to standard temperature and pressure conditions.

The WNP-2 Final Safety Analysis Report (FSAR) states that the design function of the MSIV leakage control system is to minimize the release of fission products during an accident from the main steam lines. If not diverted, this leakage could bypass the standby gas treatment system (SGTS) after a loss of coolant accident. The MSIV leakage control system directs leakage from the closed main steam isolation valves to the SGTS using blowers that maintain the pressure in the main steam lines at a slight negative pressure.

The design requirement for the MSIV leakage control system is to accommodate a leakage rate of five times the MSIV technical specification leakage of 11.5 standard cubic feet per hour (scfh) per MSIV. Total MSIV leakage, including the safety factor of five, is 230 scfh or 3.8 scfm. This value, when corrected for the worst case pressure and temperature conditions expected during surveillance testing, would not exceed 5 cfm. Therefore, the revised acceptance criterion of 30 cfm provides a significant margin of safety and the inability to meet this value would identify MSIV blower degradation well before it was unable to perform its function.

In addition, the blowers in the MSIV leakage control system are able to draw suction downstream of either the inboard or outboard MSIVs. Under normal conditions, suction is taken between the inboard and outboard MSIVs, downstream of the inboard valve. In the event that an inboard MSIV fails to isolate the main steam line, a pressure increase between the inboard and outboard MSIVs will occur. The MSIV leakage control system is designed such that a pressure increase between the two MSIVs results in isolation of the suction path between the two valves. Suction is then taken from a line downstream of the outboard valve and diverted to the SGTS.

The licensee also noted that a review of the Burns and Roe engineering documents associated with the MSIV leakage control system stated that the original units for the surveillance test acceptance criteria were cfm as opposed to scfm.

By letter dated February 14, 1992, the licensee supplemented its January 21, 1992, submittal requesting revision of the Bases section to more clearly describe the MSIV leakage control system and provided additional information regarding the performance of its MSIVs. The MSIVs were modified in 1989 and 1990 and resulted in significantly reduced valve leak test results. No MSIV overhaul activities have been necessary following completion of the modifications.

The staff has reviewed the licensee's request and finds that revision of the MSIV leakage control system blower surveillance test acceptance criteria from a flow rate of 30 standard cubic feet per minute to 30 cubic feet per minute acceptable. This determination is based on the fact that the revised acceptance criteria provides for timely identification of blower degradation well before it is unable to perform its function, the system's ability to take suction from either upstream or downstream of the outboard MSIV, and the leakage test history of the MSIVs.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Washington State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (57 FR 5028). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Patricia L. Eng

Date: March 13, 1992