April 20, 1990

Docket No. 50-397

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

Dear Mr. Sorensen:

SUBJECT: ISSUANCE OF AMENDMENT NO. 81 TO FACILITY OPERATING LICENSE NO. NPF-21 - WPPSS NUCLEAR PROJECT NO. 2 (TAC NO. 75384)

The U.S. Nuclear Regulatory Commission has issued the enclosed amendment to Facility Operating License NPF-21 to the Washington Public Power Supply System for WPPSS Nuclear Project No. 2, located in Benton County near Richland, Washington. This amendment is in response to your letter dated November 29, 1989 (G02-89-214) as supplemented by your letter dated March 21, 1990 (G02-90-058).

This amendment revises Technical Specification 3/4.3.6, "Control Rod Block Instrumentation," by modifying the requirement for performance of the channel functional test for two trip functions during certain conditions. When WNP-2 is in mode 5, the channel functional test for the source range and intermediate range monitor not full in trip functions may be satisfied by administratively controlling the positions of the detectors and by visually verifying those positions. The bases section for the technical specification is amended to include the circumstances for utilizing the alternative for the channel functional test as set forth in your supplemental letter.

A copy of the related safety evaluation supporting the amendment is enclosed. A Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

/s/

Robert B. Samworth, Senior Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 81 to Facility Operating License No. NPF-21
2. Safety Evaluation

cc w/enclosures:
See next page

DRSP/PD5	DRSP/PD5	OGC	(A)DRSP:PD5
PShea	RSamworth:sg	RBachmann	CTrammell
4/3/90	3/30/90	4/5/90	4/20/90

w/change to SE as noted

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

April 20, 1990

Docket No. 50-397

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Regulatory Programs
Washington Public Power Supply System
P.O. Box 968
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2. Safety Evaluation

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See next page

Mr. G. C. Sorensen

WPPSS Nuclear Project No. 2
(WNP-2)

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(10)

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. 81
License No. NPF-21

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

   A. The application for amendment filed by the Washington Public Power Supply System (the licensee), dated November 29, 1989 and supplemented by letter dated March 21, 1990 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 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 set forth in 10 CFR Chapter I;

   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. 81, 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

Terence L. Chan

Terence L. Chan, Acting Director
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 20, 1990

ENCLOSURE TO LICENSE AMENDMENT NO. 81

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. Also to be replaced are the following overleaf pages.

AMENDMENT PAGE	OVERLEAF PAGE
3/4 3-56	3/4 3-55
3/4 3-57	3/4 3-58
B 3/4 3-4	B 3/4 3-3
*B 3/4 3-5	
*B 3/4 3-6	

*The text on these pages is shifted but no change is made to the content of the text.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971, and NEDO-24222, dated December 1979.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the reactor protection system and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine throttle valves and fast closure of the turbine governor valves.

A fast closure sensor from each of two turbine governor valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine governor valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine throttle valves provides input to one EOC-RPT system; a position switch from each of the other two throttle valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine governor valves and a 2-out-of-2 logic for the turbine throttle valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 190ms, less the time allotted for sensor response, i.e., 10ms, and less the time allotted for breaker arc suppression determined by test, as correlated to manufacturer's test results, i.e., 83ms, and plant preoperational test results.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of Specifications 3/4.1.4, Control Rod Program Controls, 3/4.2, Power Distribution Limits and 3/4.3.1 Reactor Protection System Instrumentation. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The test exception to the weekly Channel Functional Test of the SRM/IRM Detector Not Full In instrumentation noted in Table 4.3.6-1, Control Rod Block Instrumentation Requirements, is intended to avoid cable damage and radiation exposure during operational condition 5 periods when outage work is being done in the under core region. Upon completion of all the work in this area, when access for maintenance or construction efforts is no longer required, the test will be completed per the prescribed frequency within seven days.

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels; (2) the alarm is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with 10 CFR Part 50, Appendix A, General Design Criteria 19, 41, 60, 61, 63, and 64.

The criticality monitor alarm setpoints were calculated using the criteria from 10 CFR 70.24.a.1 that requires detecting a dose rate of 20 Rads per minute of combined neutron and gamma radiation at 2 meters. The alarm setpoint was determined by calculational methods using the gamma to gamma plus neutron ratios from ANSI/ANS 8.3-1979, Criticality Accident Alarm System, Appendix B and assuming a critical mass was formed from a seismic event, with a volume of 6' x 6' x 6' at a distance of 27.7 feet from the two detectors. The calculated dose rate using the methodology is 5.05 R/hr. The allowable value for the alarm setpoint was, therefore, established at 5R/hr.

MONITORING INSTRUMENTATION (Continued)

3.4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit. This instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological monitoring instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February, 1972.

3/4.3.7.4 REMOTE SHUTDOWN MONITORING INSTRUMENTATION

The OPERABILITY of the remote shutdown monitoring instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of Appendix A to 10 CFR Part 50.

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.7.6 SOURCE RANGE MONITORS

The source range monitors provide the operator with information of the status of the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions shall not be made without this flux level information available to the operator. When the intermediate range monitors are on scale, adequate information is available without the SRMs and they can be retracted.

3/4.3.7.7 TRAVERSING IN-CORE PROBE SYSTEM

The OPERABILITY of the traversing in-core probe system with the specified minimum complement of equipment ensures that the measurements obtained from use of this equipment accurately represent the spatial neutron flux distribution of the reactor core.

MONITORING INSTRUMENTATION (Continued)

3/4.3.7.9 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the loose-part detection system ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.7.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50. The purpose of tank level indicating devices is to assure the detection and control of leaks that if not controlled could potentially result in the transport of radioactive materials to UNRESTRICTED AREAS.

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale	≤ 0.66 W + 40%	≤ 0.66 W + 43%
b. Inoperative	N.A.	N.A.
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux Upscale	≤ 0.66 W + 42%*	≤ 0.66 W + 45%*
b. Inoperative	N.A.	N.A.
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale, Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	$< 1 \times 10^5$ cps	$< 1.6 \times 10^5$ cps
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 0.7 cps	≥ 0.5 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	< 108/125 divisions of full scale	< 110/125 divisions of full scale
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	< 527 ft 2 in. elevation	< 527 ft 4 in. elevation
b. Scram Trip Bypass	N.A.	N.A.
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ 108/125 divisions of full scale	≤ 111/125 divisions of full scale
b. Inoperative	N.A.	N.A.
c. Comparator	≤ 10% flow deviation	≤ 11% flow deviation

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	N.A.	S/U(b)(c), M(c)	Q	1*
b. Inoperative	N.A.	S/U(b)(c), M(c)	N.A.	1*
c. Downscale	N.A.	S/U(b)(c), M(c)	Q	1*
2. APRM				
a. Flow Biased Neutron Flux Upscale	N.A.	S/U(b), M	Q	1
b. Inoperative	N.A.	S/U(b), M	N.A.	1, 2, 5
c. Downscale	N.A.	S/U(b), M	Q	1
d. Neutron Flux - Upscale, Startup	N.A.	S/U(b), M	Q	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	N.A.	S/U(b), W(#)	N.A.	2, 5
b. Upscale	N.A.	S/U(b), W	Q	2, 5
c. Inoperative	N.A.	S/U(b), W	N.A.	2, 5
d. Downscale	N.A.	S/U(b), W	Q	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	N.A.	S/U(b), W(#)	N.A.	2, 5
b. Upscale	N.A.	S/U(b), W	Q	2, 5
c. Inoperative	N.A.	S/U(b), W	N.A.	2, 5
d. Downscale	N.A.	S/U(b), W	Q	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	N.A.	Q	R	1, 2, 5**
b. Scram Trip Bypass	N.A.	M	N.A.	5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	N.A.	S/U(b), M	Q	1
b. Inoperative	N.A.	S/U(b), M	N.A.	1
c. Comparator	N.A.	S/U(b), M	Q	1

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) Within 24 hours prior to startup, if not performed within the previous 7 days.

(c) Includes reactor manual control multiplexing system input.

* With THERMAL POWER $\geq$ 30% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

This CHANNEL FUNCTIONAL TEST may be satisfied while in MODE 5 provided the detector is administratively controlled in the full in position and is visually verified to be in once per 24 hours, unless the CHANNEL FUNCTIONAL TEST has not been performed within the past 92 days.

3/4.3.7 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.1 The radiation monitoring instrumentation channels shown in Table 3.3.7.1-1 shall be OPERABLE with their alarm setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3.7.1-1.

ACTION:

a. With a radiation monitoring instrumentation channel alarm setpoint exceeding the value shown in Table 3.3.7.1-1, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION required by Table 3.3.7.1-1.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.1 Each of the above required radiation monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the conditions and at the frequencies shown in Table 4.3.7.1-1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 81 TO FACILITY OPERATING LICENSE NO. NPF-21

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

NUCLEAR PROJECT NO. 2

DOCKET NO. 50-397

1.0 INTRODUCTION

In correspondence dated November 29, 1989 and March 21, 1990, Washington Public Power Supply System (the licensee) requested an amendment to the Technical Specifications (TS) for Washington Nuclear Plant No. 2 (WNP-2). The licensee's March 21, 1990 letter only provided information regarding the purpose of the amendment to be inserted in the basis section and did not alter the staff's initial no significant hazards determination. The proposed amendment provides an alternative to the weekly channel functional test of the Source Range Monitor/Intermediate Range Monitor (SRM/IRM) NOT-FULL-IN Control Rod (CR) Block when the following conditions exist:

1) The plant is in MODE 5 (REFUELING) operations,
2) The SRM/IRM cables are rolled up and tied out of the way in preparation for under core work,
3) A channel functional test of the NOT-FULL-IN CR Block has been performed in the previous 92 days.

The licensee's proposed alternative to the weekly channel functional test is 1) to verify visually at least once each 24 hours that all SRM/IRM detectors are fully inserted into the reactor core, and 2) to administratively control the detectors in the full in position. Administrative control is achieved by removing the fuses from the circuits connecting the SRM/IRM drive mechanisms to their power supplies to prevent SRM/IRM detector movement during this period.

2.0 EVALUATION

The SRMs and IRMs indicate neutron flux and reactor period for reactor power operations extending from shutdown conditions to the lower portion of the reactor power range. When fully inserted, the SRMs and IRMs are positioned at the axial centerline of the core, and thereby monitor the high-flux region of the reactor core. If the SRM/IRM detectors are not located at the axial centerline of the core during MODE 5 operations, control rod withdrawal would not be safe and is therefore prohibited by a SRM/IRM Detector NOT-FULL-IN signal.

To insure a CR cannot be withdrawn when the SRMs and IRMs are not fully inserted into the reactor core, the WNP-2 TS requires the licensee to test the SRM/IRM Detector NOT-FULL-IN CR Block feature by withdrawing the SRMs and IRMs while monitoring actuation of the CR Block signal. This test must be performed every seven days while the reactor is in MODE 5 (REFUELING) operations.

During under core maintenance operations, the licensee rolls up the SRM and IRM detector cables and ties the rolled cables out of the way to prevent cable damage. The licensee states that the SRM/IRM instrument cables must be unrolled and laid out to allow unrestricted SRM/IRM detector assembly movement while conducting the SRM/IRM Detector NOT-FULL-IN CR Block channel functional test. Upon completing the tests, the cables are again rolled up and tied out of the way. This process of rolling and unrolling the cables requires approximately 8 man hours/week. Consequently, during the period when the SRM/IRM detector cables are rolled up and tied out of the way, performing the SRM/IRM Detector NOT-FULL-IN CR Block channel functional test will expose maintenance personnel to extreme work conditions and increase the exposure of maintenance personnel to radiation by 8 man hours per week.

Additionally, the licensee asserts that rolling and unrolling the SRM/IRM cables flexes the cables, causing wear points and, in turn, breaks in the cable/connector protector tube. Repairs of damaged cables in the under core region (SRM, IRM, CR, etc) during the licensee's previous outage required approximately 300 man hours labor and incurred approximately 3 man rem exposure. Exemption from the SRM/IRM Detector NOT-FULL-IN CR Block channel functional test would reduce the number of man hours spent in extreme work conditions to repair damaged cables in the under core region. The staff concurs with this conclusion.

The licensee states that visually verifying the SRM/IRM assemblies are fully inserted, and administratively prohibiting SRM/IRM movement (by removing the fuses from the SRM/IRM drive mechanism connections to the power supplies) ensure that the SRMs and IRMs will not be withdrawn from the axial midplane of the core. Consequently, the channel functional test for SRM/IRM Detector NOT-FULL-IN CR Block feature is not required because the purpose of the channel functional test is to insure CRs will not be moved if a SRM or IRM is withdrawn from the axial midplane of the core. The staff finds acceptable the licensee's conclusion that verifying SRM/IRM position and preventing movement of the SRMs and IRMs ensure that the SRM/IRM Detector NOT-FULL-IN CR Block feature is not required during this period of maintenance operations.

In summary the staff finds the proposed amendment to the technical specifications acceptable based on the licensee's proposed administrative controls and surveillances as enumerated below:

1) The licensee will perform a visual verification of the position of the SRM/IRM detector assemblies at least once each 24 hours while the SRM/IRM cables are rolled up to insure that all SRM/IRM assemblies are fully inserted into the reactor core.

2) While the SRM/IRM cables are rolled up and tied out of the way for under core maintenance operations, the licensee will administratively control the position of the detectors by removing the fuses connecting the power supplies to the SRM/IRM drive mechanisms such that the SRM/IRM detector assemblies cannot be withdrawn from their fully inserted position in the reactor core.

3) The licensee will perform a channel functional test of the SRM/IRM Detector NOT-FULL-IN CR Block function at the normal test frequency (seven days for MODE 5 operations) when the SRM/IRM cables are not rolled up for under core maintenance operations.

4) The licensee will perform a channel functional test of the SRM/IRM NOT-FULL-IN CR Block if the test has not been performed in the previous 92 days.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the surveillance of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that this 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this 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 this amendment.

4.0 CONTACT WITH STATE OFFICIAL

The Commission made a proposed determination that the amendment involves no significant hazards consideration (55 FR 4287, February 7, 1990) and consulted with the State of Washington. No public comments were received, and by letter dated March 20, 1990 the State of Washington advised that they have no comment.

5.0 CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Dated: April 20, 1990