March 27, 1995

Mr. J. V. Parrish (Mail Drop 1023)
Vice President, Nuclear Operations
Washington Public Power Supply System
P. O. Box 968
Richland, Washington 99352-0968

SUBJECT: ISSUANCE OF AMENDMENT FOR THE WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2 (TAC NO. M90837)

Dear Mr. Parrish:

The Commission has issued the enclosed Amendment No. 135 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 October 31, 1994.

The amendment relocates requirements regarding safety/relief valve position indication instrumentation from TS 3/4.3.7.5, "Accident Monitoring Instrumentation," and TS 3/4.4.2, "Safety/Relief Valves," to other licensee-controlled documents.

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

James W. Clifford, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosures: 1. Amendment No. 135 to NPF-21
2. Safety Evaluation

cc w/encls: See next page

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Docket File
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PDIV-2/RF
RIV, WCFO (4)

DOCUMENT NAME: WNP90837.AMD

OFC	LA/DRPW	PE/PD4-1	HICB	OTSB	OGC	PM/PDIV-2
NAME	DFoster-Curseen	RSchaaf:pk	JWermiel	CGrimes	RBachmann	JClifford
DATE	2/23/95	2/23/95	2/27/95	3/2/95	3/3/95	3/21/95

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Washington Public Power Supply System
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OFFICIAL RECORD COPY #95-051

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 27, 1995

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Vice President, Nuclear Operations
Washington Public Power Supply System
P. O. Box 968
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Division of Reactor Projects III/IV
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Docket No. 50-397

Enclosures: 1. Amendment No. 135 to NPF-21
2. Safety Evaluation

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Mr. J. V. Parrish
Washington Public Power Supply System

WPPSS Nuclear Project No. 2
(WNP-2)

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

DOCKET NO. 50-397

NUCLEAR PROJECT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 135
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 October 31, 1994, 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. 135 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 immediately and will be implemented prior to restart from the spring 1995 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

James W. Clifford

James W. Clifford, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 27, 1995

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 135 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
3/4 3-71	3/4 3-71
3/4 3-73	3/4 3-73
3/4 3-74	3/4 3-74
3/4 4-7a	3/4 4-7a
B 3/4 4-1a	B 3/4 4-1a

TABLE 3.3.7.5-1
ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. Reactor Vessel Pressure	2	1	1, 2	80
2. Reactor Vessel Water Level	2	1	1, 2	80
3. Suppression Chamber Water Level	2	1	1, 2	80
4. Suppression Chamber Water Temperature	2/sector	1/sector	1, 2	80
5. Suppression Chamber Air Temperature	2	1	1, 2	80
6. Drywell Pressure	2	1	1, 2	80
7. Drywell Air Temperature	2	1	1, 2	80
8. Drywell Oxygen Concentration	2	1	1, 2	80
9. Drywell Hydrogen Concentration	2	1	1, 2	80
10.				
11. Suppression Chamber Pressure	2	1	1, 2	80
12. Condensate Storage Tank Level	2	1	1, 2	80
13. Main Steam Line Isolation Valve Leakage Control System Pressure	2	1	1, 2	80

Table 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION STATEMENTS

ACTION 80 -

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 81 - With the number of OPERABLE accident monitoring instrumentation channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

a. Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

b. In lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	APPLICABLE OPERATIONAL CONDITIONS
1. Reactor Vessel Pressure	M	R	1,2
2. Reactor Vessel Water Level	M	R	1,2
3. Suppression Chamber Water Level	M	R	1,2
4. Suppression Chamber Water Temperature	M	R	1,2
5. Suppression Chamber Air Temperature	M	R	1,2
6. Primary Containment Pressure	M	R	1,2
7. Drywell Air Temperature	M	R	1,2
8. Drywell Oxygen Concentration	M	R	1,2
9. Drywell Hydrogen Concentration	M	Q	1,2
10.			
11. Suppression Chamber Pressure	M	R	1,2
12. Condensate Storage Tank Level	M	R	1,2
13. Main Steam Line Isolation Valve Leakage Control System Pressure	M	R	1,2
14. Neutron Flux:			
APRM	M	R	1,2
IRM	M	R	1,2
SRM	M	R	1,2
15. RCIC Flow	M	R	1,2
16. HPCS Flow	M	R	1,2
17. LPCS Flow	M	R	1,2

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 a) The safety valve function of at least 12 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:*

2 safety/relief valves @ 1150 psig +1%/-3%
4 safety/relief valves @ 1175 psig +1%/-3%
4 safety/relief valves @ 1185 psig +1%/-3%
4 safety/relief valves @ 1195 psig +1%/-3%
4 safety/relief valves @ 1205 psig +1%/-3%

APPLICABILITY: OPERATIONAL CONDITIONS 1, and 2, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

b) The safety valve function of at least 4 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:*

2 safety/relief valves @ 1150 psig +1%/-3%
4 safety/relief valves @ 1175 psig +1%/-3%
4 safety/relief valves @ 1185 psig +1%/-3%
4 safety/relief valves @ 1195 psig +1%/-3%
4 safety/relief valves @ 1205 psig +1%/-3%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3, when THERMAL POWER is less than 25% of RATED THERMAL POWER.

ACTION:

a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 90°F, close the stuck open safety/relief valve(s); if unable to close the open

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

valve(s) within 2 minutes or if suppression pool average water temperature is 110°F or greater, place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor recirculation loop inoperable has been evaluated and been found to be acceptable provided the unit is operated in accordance with the single recirculation loop operation Technical Specifications herein.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design-basis-accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation.

Recirculation loop flow mismatch limits are in compliance with the ECCS LOCA analysis design criteria. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA. Where the recirculation loop flow mismatch limits cannot be maintained during two recirculation loop operation, continued operation is permitted in the single recirculation loop operation mode.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 145°F.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve capacity is designed to limit the primary system pressure, including transients, in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, 1971, Nuclear Power Plant components (up to and including Summer 1971 Addenda). The Code allows a peak pressure of 110% of design pressure (1250 (design) X 1.10 = 1375 psig maximum) under upset conditions. In addition, the Code specifications require that the lowest valve setpoint be at or below design pressure and the highest valve setpoint be set so that total accumulated pressure does not exceed 110% of the design pressure.

The safety valve sizing evaluation assumes credit for operation of the scram protective system which may be tripped by one of two sources; i.e., a direct position switch or neutron flux signal. The direct scram signal is derived from position switches mounted on the main steamline isolation valves (MSIV's) or the turbine stop valve, or from pressure switches mounted on the dump valve of the turbine control valve hydraulic actuation system. The position switches are actuated when the respective valves are closing, and following 10% travel of full stroke. The pressure switches are actuated when a fast closure of the control valves is initiated. Further, no credit is taken for power operation of the pressure relieving devices. Credit is only taken for

3/4.4.2 SAFETY/RELIEF VALVES (Continued)

the dual purpose safety/relief valves in their ASME Code qualified mode (spring lift) of safety operation.

The overpressure protection system must accommodate the most severe pressurization transient. There are two major transients that represent the most severe abnormal operational transient resulting in a nuclear system pressure rise. The evaluation of these events with the final plant configuration has shown that the MSIV closure is slightly more severe when credit is taken only for indirect derived scrams; i.e., a flux scram. Utilizing this worse case transient as the design basis event, a minimum of 12 safety/relief valves are required to assure peak reactor pressure remains within the Code limit of 110% of design pressure.

Testing of safety/relief valves is normally performed at lower power with adequate steam pressure and flow. It is desirable to allow an increased number of valves to be out of service during testing. Therefore, an evaluation of the MSIV closure without direct scram was performed at 25% of RATED THERMAL POWER assuming only 4 safety/relief valves were operable. The results of this evaluation demonstrate that any 4 safety/relief valves have sufficient flow capacity to assure that the peak reactor pressure remains well below the code limit of 110% of design pressure.

Demonstration of the safety/relief valve lift settings will be performed in accordance with the provisions of Specification 4.0.5.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

The primary containment sump flow monitoring system monitors the UNIDENTIFIED LEAKAGE collected in the floor drain sump with a sensitivity such that 1 gpm change within 1 hour can be measured. Alternatively, other methods for measuring flow to the sump which are capable of detecting a change in UNIDENTIFIED LEAKAGE of 1 gpm within 1 hour with an accuracy of ± 2% may be used, for up to 30 days, when the installed system is INOPERABLE.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 135 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 October 31, 1994, Washington Public Power Supply System submitted a request for changes to the Technical Specifications (TS) for Nuclear Project No. 2. The proposed changes would relocate requirements regarding safety/relief valve (SRV) position indication instrumentation from the TS to other licensee-controlled documents. The requirements for this instrumentation are currently contained in TS 3/4.3.7.5, "Accident Monitoring Instrumentation," and TS 3/4.4.2, "Safety/Relief Valves." The proposed changes are consistent with guidance contained in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4," and NUREG-1434, "Standard Technical Specifications, General Electric Plants, BWR/6."

Section 182a of the Atomic Energy Act (the Act) requires applicants for nuclear power plant operating licenses to state TS to be included as part of the license. The Commission's regulatory requirements related to the content of TS are set forth in 10 CFR 50.36. That regulation requires that the TS include items in five specific categories, including (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. However, the regulation does not specify the particular requirements to be included in a plant's TS.

The Commission has provided guidance for the contents of TS in its "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" ("Final Policy Statement"), 58 FR 39132 (July 22, 1993), in which the Commission indicated that compliance with the Final Policy Statement satisfies Section 182a of the Act. In particular, the Commission indicated that certain items could be relocated from the TS to licensee-controlled documents, consistent with the standard enunciated in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). In that case, the Atomic Safety and Licensing Appeal Board stated that "technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety."

Consistent with this approach, the Final Policy Statement identified four criteria to be used in determining whether a particular matter is required to be included in the TS, as follows: (1) installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary; (2) a process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (3) a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (4) a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.[1] As a result, existing TS requirements which fall within or satisfy any of the criteria in the Final Policy Statement must be retained in the TS, whereas TS requirements which do not fall within or satisfy these criteria may be relocated to other licensee-controlled documents.

2.0 EVALUATION

The licensee proposed to relocate limiting conditions of operation, action statements, surveillance requirements, and notes regarding SRV position indication instrumentation from TS 3/4.3.7.5, "Accident Monitoring Instrumentation," to other licensee-controlled documents. The primary purpose of the accident monitoring instrumentation is to display plant variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design-basis events. The instruments that monitor these variables are identified by the licensee in accordance with guidance contained in Regulatory Guide 1.97. Regulatory Guide 1.97 defines five types of variables (Types A, B, C, D, and E) to be monitored by the control room operator during the course of an accident and during the long term stable shutdown phase following an accident. The Regulatory Guide also provided design and qualification criteria for this instrumentation, separated into three categories which provide a graded approach to requirements depending on the importance to safety of the measurement of a specific variable.

In general, accident monitoring instrumentation is required to provide sufficient information to the operator in the control room to assess plant response in the event of an accident, i.e., to indicate that automatic safety

[1]The Commission recently promulgated a proposed change to § 50.36, pursuant to which the rule would be amended to codify and incorporate these criteria. This proposed rule clarified the contents of the Bases in the improved standard technical specifications and specified that only limiting conditions for Reactor Core Isolation Cooling, Isolation Condenser, Residual Heat Removal, Standby Liquid Control, and Recirculation Pump Trip meet the guidance for inclusion in the TS under Criterion 4. In the proposed change to § 50.36, the Commission specifically requested public comments regarding application of Criterion 4. Until additional guidance has been developed, Criterion 4 will not be applied to add TS restrictions other than those indicated above. See Proposed Rule, "Technical Specifications," 59 FR 48180 (September 20, 1994).

systems are performing properly and deviations from the expected accident course are minimal. The staff has stated in NUREG-1433 and NUREG-1434 that accident monitoring instrumentation which satisfies the definition of Type A in Regulatory Guide 1.97 meets Criterion 3 of the Policy Statement. The staff has also determined that Category I, non-Type A monitoring instruments satisfy the Final Policy Statement for inclusion in the TS based on their important contribution to the reduction to risk following an accident. Thus, the staff clarified in NUREG-1433 and NUREG-1434 that accident monitoring instrument functions necessary to avert an immediate threat to the public health and safety are limited to Regulatory Guide 1.97 Type A and Category 1, non-Type A instruments, and these instruments should remain in TS. The remaining Regulatory Guide 1.97 instrument functions need not be retained in TS.

The WNP-2 Regulatory Guide 1.97 analysis identified SRV position indication as a Category 2, Type D, variable. This classification was reviewed and approved by the staff, as documented in a safety evaluation report dated March 23, 1988. Based on this classification and the discussion above, the staff concludes that the SRV position indication instrumentation is not necessary to avert an immediate threat to public health and safety. Therefore, the requirements associated with this instrumentation may be relocated from TS 3/4.3.7.5 to other licensee-controlled documents.

The licensee also proposed to relocate action statements and surveillance requirements regarding SRV position indication instrumentation from TS 3/4.4.2. The licensee stated that the position indication instrumentation is a non-intrusive design that does not affect the operability of the SRVs. Failure of the instrumentation would not increase the severity of a stuck open SRV event, nor would it reduce the capability of the SRV to perform its safety function. The instrumentation provides valve position indication and alarms only, and does not perform any control or accident mitigating functions. Therefore, the staff finds that operability of the SRV position indication instrumentation is not necessary to avert an immediate threat to public health and safety, and the requirements associated with this instrumentation may be relocated from TS 3/4.4.2 to other licensee-controlled documents.

The licensee also applied the Final Policy Statement criteria to the associated technical specifications to determine the acceptability of relocating the SRV position indication instrumentation requirements. The analysis determined that the requirements do not meet any of the Policy Statement criteria for inclusion in the TS. The licensee's analysis is summarized as follows:

(1) Although the SRV position indication instrumentation can indicate a breech of the reactor coolant pressure boundary (RCPB) via a stuck-open SRV, this event does not involve a significant degradation of the RCPB. The event causes only a slight decrease in thermal margins and does not result in fuel damage. Furthermore, operators can rely on other instruments (such as suppression pool temperature and reactor pressure) to indicate the existence of a breech of the RCPB.

(2) Operability of the SRV position indication instrumentation is not an initial condition for the stuck-open SRV transient analysis or any other analyzed accident or transient. Operator response to a stuck-open SRV is based on a suppression pool high-temperature alarm.

(3) No credit is taken for operation of the SRV position indication instrumentation in the stuck-open SRV transient analysis. Operator response for this event is assumed to be initiated based on a suppression pool high-temperature alarm.

(4) The stuck-open SRV event does not lead to an uncontrolled activity release to the environment; therefore, the SRV position indication instrumentation is not significant to public health and safety.

Based on the licensee's analysis, the staff finds that the SRV position indication instrumentation does not meet any of the Policy Statement criteria for inclusion in TS, and may be relocated to other licensee-controlled documents. The SRV position indication instrumentation will continue to be identified in the FSAR, and the relocated requirements will be maintained in licensee-controlled procedures. Any changes to this instrumentation or the relocated requirements will be controlled in accordance with 10 CFR 50.59.

The staff concludes that these requirements are not required to be in the TS under 10 CFR 50.36 or Section 182a of the Atomic Energy Act, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Further, they do not fall within any of the four criteria set forth in the Commission's Final Policy Statement, discussed above. In addition, the Staff finds that sufficient regulatory controls exist under 10 CFR 50.59 to adequately control future modifications to these requirements. Accordingly, the staff has concluded that these requirements may be relocated from the TS to their respective licensee-controlled documents.

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 (59 FR 65831). 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.

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