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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 69 TO FACILITY OPERATING LICENSE NO. NPF-21

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

NUCLEAR PROJECT NO. 2

DOCKET NO. 50-397

1.0 INTRODUCTION

By letters dated March 3, 1989, April 20, 1989 and June 1, 1989 (Ref. 1), Washington Power Supply System (WPPSS), the licensee, proposed to amend Operating License NPG-21 to support Cycle 5 operation of their Nuclear Plant No. 2 (WNP-2) with Advanced Nuclear Fuels Corporation (ANF) reload fuel. In support of the Cycle 5 reload, the licensee submitted reports consisting of a reload summary (Ref. 2), the reload analysis (Ref. 3), the plant transient analysis (Ref. 4), and the proposed Technical Specification (TS) changes.

The April 20, 1989 submittal provided additional analyses to support the proposed changes and clarifications to contain TS changes. The June 1, 1989 submittal provided graphic quality copies of the four figures affected by the proposed amendment. Additional detail was provided for one of the four figures (Figure 3.2.3-1). These submittals did not alter the action noticed in the Federal Register on April 5, 1989 and did not affect the initial no significant hazards determination.

2.0 EVALUATION

2.1 Reload Description

The WNP-2 Cycle 5 reload will incorporate a total of 136 unirradiated ANF 8x8C fuel assemblies which replace 136 of the General Electric (GE) initial core fuel assemblies. The remainder of the core is comprised of 152 ANF 8x8C assemblies loaded for Cycle 4, 148 ANF 8x8C assemblies loaded for Cycle 3, 128 ANF 8x8C assemblies loaded for Cycle 2 and 200 GE P8x8R assemblies remaining from the initial core. The licensee is also requesting approval for loading four 9x9 ANF lead fuel test assemblies (LTAs).

2.2 Fuel Design

The mechanical design of the ANF 8x8C reload fuel is described in References 6, 7 and 8. The remaining fuel types to be returned to the Cycle 5 core were approved for operation in previous cycles.

The 136 8x8C ANF reload fuel assemblies manufactured for loading in Cycle 5 are essentially identical to the 8x8C ANF reload assemblies originally

fabricated for reload in Cycle 2 in all major physical characteristics except U-235 enrichment. All of the reload fuel assemblies are essentially identical to the 8x8C ENC fuel approved for use in Cycle 2 (Ref. 5). Based on this, and on the fuel mechanical design analysis which used approved methodologies (Ref. 8) and yielded acceptable results, the staff finds the mechanical design of the ANF 8x8C reload fuel for the WNP-2 Cycle 5 reload acceptable.

2.3 Thermal-Hydraulic Design

The ANF thermal-hydraulic methodology and criteria used for the Cycle 5 design and analysis is the same as the previous WNP-2 reloads. These previous reviews concluded that hydraulic compatibility between ANF and GE fuel is satisfactory and the calculation of core bypass flow and the safety limit minimum critical power ratio (MCPR) is acceptable. The methodology for Cycle 5 is based on ANF's revised critical power methodology (Ref. 9) which incorporates a constant flow MCPR formulation for BWR applications and has been approved by the staff. The XN-3 correlation used to develop the MCPR limit has been approved for application to both the ANF 8x8C and GE 8x8R fuel types (Ref. 10). Therefore, the proposed safety limit MCPR of 1.06 for all fuel types in this reload is acceptable.

2.4 Thermal Hydraulic Stability

The licensee, by letter dated March 3, 1989 (G02-89-030), submitted its response to IE Bulletin 88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors." WPPSS' letter dated March 31, 1989 (G02-89-051) also requested Technical Specification changes for power flow instability and neutron flux noise monitoring. This request is being reviewed separately and a separate document will be issued addressing the thermal hydraulic stability concerns. Completion of the thermal hydraulic review is not a prerequisite for the Cycle 5 reload.

2.5 Nuclear Design

The nuclear design for Cycle 5 has been performed with ANF methodologies previously reviewed and approved (Ref. 11). The fuel loading pattern is given in Figure 4.2 of Reference 3. The beginning-of-cycle (BOC) shutdown margin (SDM) is 1.32% delta-k, well in excess of the required 0.38% delta-k. The standby liquid control system (SLCS) was calculated to provide a SDM of 3.67% delta-k for cold conditions with all control rods in their full power positions. This fully meets shutdown requirements. Since these results have been obtained with previously approved methods and fall within the expected range, the staff concludes that the nuclear design of the Cycle 5 reload core is acceptable.

2.6 Transient Analyses

Core wide transients were analyzed with the COTRANSA computer code (Ref. 12) which includes a one-dimensional neutron kinetics model for evaluation of the axial power shape response during pressurization transients (generator

load rejection and feedwater controller failure). The referenced report has been reviewed by the staff and the methods for calculating the system transient response were found to be acceptable.

Calculation of the change in critical power ratio (CPR) during the core wide transient events involves the use of COTRANSA system results which serve as input to a XOBRA-T hot channel analysis model (Ref. 13) used to calculate the delta CPR values. The XCOBRA-T model has been reviewed by the staff and found to be acceptable. The licensee evaluated several categories of potential core wide transients for Cycle 5 and provided specific results for the three limiting transients: load rejection without bypass (LRWB), feedwater controller failure (FWCF), and loss of feedwater heating (LOFH). For operation at rated power in the range of EOC-2000 (3750)MWD/MTU to EOC, the LRWB is identified as the limiting transient. The calculated delta CPR, assuming WNP-2 measured scram speed, was 0.25 for ANF fuel and 0.29 for GE fuel resulting in MCPR limits of 1.31 and 1.35 for ANF fuel and GE fuel, respectively.

If the recirculation pump trip (RPT) should become inoperable, the limiting transient between 3750 MWD/MTU and EOC is still the LRWB. Assuming normal scram speeds, at end-of-cycle exposures, the MCPR operating limits are 1.38 (ANF fuel) and 1.42 (GE fuel) with an inoperable RPT. For technical specification scram times, the MCPR limits are 1.42 (ANF fuel) and 1.48 (GE fuel) with an inoperable RPT. These values are bounded by the proposed Cycle 5 MCPR operating limits and are, therefore, acceptable.

The control rod withdrawal error (CRWE) was found to be most limiting from BOC up to 3750 MWD/MTU. The delta CPR for the CRWE was 0.17 for both ANF fuel and GE fuel. At higher exposures, the CRWE delta CPR values are bounded by the LRWB transient as shown above.

The most limiting event for reactor vessel over-pressurization is the main steamline isolation valve (MSIV) closure without direct scram (single failure) on valve position. The maximum value of the sensed pressure in the steam dome was 1286 psig which corresponds to a maximum vessel pressure of 1315 psig at the lower plenum. These values are less than the Technical Specification limit of 1325 psig as measured by the steam dome pressure indicator and the 1375 psig ASME vessel pressure limit. This is acceptable.

The limiting plant system transients mentioned above were all analyzed at an increased core flow of 106% of rated core flow. ANF has performed analyses which demonstrate that the ANF 8x8C fuel assembly can operate satisfactorily from a mechanical standpoint at this increased flow. GE has also performed analyses for the reactor internals and for the GE fuel assembly which showed satisfactory operation at this increased flow. Based on these analyses and on the similarity between the two fuel types utilized in Cycle 5, the staff concludes that WNP-2 can operate safely with extended core flow up to 106% of rated core flow during Cycle 5.

The licensee reviewed the recirculation flow run-up analysis for Cycle 2 and concluded that the Cycle 2 analysis is applicable to Cycle 5 except for the six degree reduction in feedwater temperature at full power conditions. Thus, the reduced flow MCPR for Cycle 5 is changed on the conservative side from earlier cycles. The reduced flow MCPR operating limit for Cycle 5 presented in T/S Figure 3.2.3-1 is acceptable.

2.7 Postulated Accidents

The control rod drop accident (CRDA) yields a value of 121 cal/gm for the maximum deposited fuel rod enthalpy. This is well below the NRC required limit of 280 cal/gm and is, therefore, acceptable.

The loss of coolant accident (LOCA) analysis for Cycle 2 was performed for a full core of ANF 8x8C fuel and remains applicable for the Cycle 5 residual and reload ANF fuel. These LOCA analyses have covered an acceptable range of conditions, have been performed with approved methodology and the current technical specification MAPLHGR values for the ANF fuel were found acceptable. Since ANF 8x8C fuel is hydraulically and neutronically compatible with the GE fuel in Cycle 5, the existing GE LOCA analysis and MAPLHGR limits remain applicable to the GE fuel.

2.8 Single Loop Operation (SLO)

Single Loop Operation was approved in Amendment 62 for Cycle 4. The description and evaluation of SLO in Amendment 62 is applicable for Cycle 5 (Ref. 14, 15, 16) and thus, SLO is acceptable.

2.9 9x9-IX and 9x9-9X Lead Test Assemblies (LTAs)

The licensee intends to load the 9x9 lead fuel test assemblies (LTAs) in core locations which have been analyzed to have sufficient margin such that the LTAs are not expected to be the limiting assemblies in the core on neither a nodal or a bundle power basis. This approach is intended to prevent the 9x9 LTAs from ever being the limiting fuel bundles. Evaluations were performed consistent with ANF methodolgoy (Ref. 11) to establish a licensing basis for two ANF 9x9-IX and two ANF 9x9-9X LTA in the WNP-2 Cycle 5 core.

The insertion of only four ANF 9x9 LTAs in the Cycle 5 core will have negligible effects upon core wide transient performance. However, 9x9 LTA specific analyses were performed to assure that the Cycle 5 operating limits are also applicable to the LTAs. Fuel specific LHGR and MAPLHGR limits were developed for these LTAs.

The dynamic response of the LTAs is expected to be almost identical to that of the 8x8 already in the core. This is due to the fact that the fuel assembly stiffness is provided by the assembly channel, which is the same in both designs. The mass of the LTAs is very close to that of the 8x8's. It thus follows that the dynamic response should be the same.

The 9x9 LTAs are hydraulically compatible with the co-resident ANF 8x8 fuel assemblies based on a comparison of fuel component hydraulic resistances. Steady state thermal hydraulic analysis has shown that even though the ANF 9x9 LTA design has a somewhat smaller flow area than the ANF 8x8 design, no reduction in thermal margin is expected in the Cycle 5 core. This is due to the increased critical power performance of the ANF 9x9 LTA design relative to the ANF 8x8 design at WNP-2 Cycle 5 conditions.

The average enrichment and enrichment distribution for the 9x9-IX and 9x9-9X fuel assemblies have been selected to match, as closely as possible, the neutronic performance of the four 8x8 XN-3 2.64 w/o U-235 reload assemblies included in the Cycle 5 reload. The fuel assembly average enrichment is 2.53 w/o U-235 for the 9x9-IX design and 2.59 w/o U-235 for the 9x9-9X design. The average enrichment of the 138 inch fuel assembly is 2.69 w/o U-235 for the 9x9-IX and 2.75 w/o U-235 for the 9x9-9X. Each 9x9 assembly contains six fuel rods containing Gd_2O_3 blended with 2.51 w/o U-235. The 9x9 fuel assembly contains 72 fuel rods and one central water channel displacing nine rod positions.

The nuclear characteristics of the 9x9 LTAs are similar to the characteristics of the ANF 8x8 fuel. The effect of replacing four ANF 8x8 assemblies with the four ANF 9x9 LFAs on the Cycle 5 core neutronics is negligible. The maximum cold uncontrolled non-voided k of the 9x9 fuel is 1.215 compared to the maximum k of 1.229 for the XN-3 8x8 fuel; thus the 9x9 fuel is compatible with the 8x8 fuel for fuel storage in the spent fuel pool.

Analyses of the WNP-2 Cycle 5 limiting transients have been performed for ANF 8x8, ANF 9x9 LTAs, and GE P8x8R fuel. It has been shown that using the XN-3 ANF CHF correlation, the bundle power required to produce transition boiling in an ANF 9x9 LTA is higher than that for an ANF 8x8 bundle. That is, when an ANF 9x9 LTA bundle is modeled as an 8x8 bundle with equivalent conditions, there is margin to the MCPR safety limit during all transients. The Cycle 5 Safety Limit Analysis considered the LTAs such that the MCPR safety limit of 1.06 is also applicable to the LTAs. Therefore, the ANF 9x9 LTAs can be monitored to the ANF 8x8 fuel limits.

Since heatup is primarily a planar and not an axial phenomena, the appropriate bundle power limit that is derived from a LOCA analysis is the peak bundle planar power. The ANF 9x9 LTAs have better cooling during LOCA conditions relative to an ANF 8x8 fuel assembly due to the lower stored energy in the fuel rods, a greater surface area provided by the larger number of fuel rods, and more inert surface from the central water channel. Thus, a LOCA analysis for the ANF 9x9 LTAs would yield lower Peak Cladding Temperatures (PCTs) and metal-water reactions than an ANF 8x8 assembly at the same bundle peak planar power. The MAPLHGR limits for the ANF 9x9 LTAs restrict the peak bundle planar power to that analyzed for the ANF 8x8 fuel and assure that the criteria are met for the ANF 9x9 LTAs in Cycle 5.

The fuel loading error was analyzed for the ANF 9x9 LTAs. Results show that if the loading error went undetected, the offsite consequences would remain well within the guidelines specified in 10 CFR Part 100.

All operational limits used for ANF 8x8 fuel are applicable to the ANF 9x9 LTAs except for fuel type specific MAPLHGR limits and the 9x9-IX and 9x9-9X LHGR limits. The LHGR limits for the 9x9-IX and 9x9-9X LFAs are shown in T/S Figures 3.2.4-2 and 3.2.4-3 respectively, and the MAPLHGR limits for the LTAs are shown in T/S Figure 3.2.1-6. These are acceptable.

3.0 TECHNICAL SPECIFICATION CHANGES

(a) T/S 3.1.3.4: Average Scram insertion time for the four control rods are changed to agree with the values assumed in the analysis. The proposed change is acceptable.

(b) T/S 3/4.2.1: This specification is amended to include reference to Figure 3.2.1-6 giving the specific limits applicable to the lead fuel assemblies. The associated Bases section is similarly revised. As stated above under section 2.9, these changes are acceptable.

(c) T/S 3/4.2.3: Table 3.2.3-1 and Figure 3.2.3-1 have been revised to reflect Cycle 5 MCPR operating limits. These new limits are based on the Cycle 5 reload safety analysis which has been evaluated and approved in Section 2 and are, therefore, acceptable.

(d) T/S 3/4.2.4: This specification is amended to include reference to Figures 3.2.4-2 and 3.2.4-3, which give the limits for the linear heat generation rate applicable to the lead fuel assembkies. As stated above under section 2.9, these changes are acceptable.

(e) Bases Section 2.1.2: The third paragraph of this Bases section is revised to delete reference to Bases Table B2.1.2-2. Table B2.1.2-2 was removed by Amendment 28. Deletion of this reference is administrative and is acceptable.

(f) Bases section 3/4.2.1: This section is amended such that the reference cited in the final line of the section is identified. This is an editorial change and is acceptable.

(g) T/S 5.3: The description of the fuel assemblies in the reactor core is amended to show that the reload fuel may employ a nine by nine array of fuel rods. This change is to accommodate the lead fuel assemblies. Since the limits applicable to the lead fuel assemblies have been reviewed and found acceptable in Section 2.9, this change is acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation and use 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.

5.0 CONTACT WITH STATE OFFICIAL

The Commission made a proposed determination that the amendment involves no significant hazards consideration (54 FR 13771, April 5, 1989) and consulted with the State of Washington. No public comments were received, and the State of Washington did not have any comment.

6.0 CONCLUSION

The staff has reviewed the reports submitted for the Cycle 5 reload of WNP-2 with ANF fuel using ANF methodology and analysis. Based on this review, the staff concludes that appropriate material was submitted and that the fuel design, nuclear design, thermal-hydraulic design and transient and accident analyses are acceptable. The Technical Specification changes submitted for this reload suitably reflect the use of acceptable methodology and the operating limits associated with those changes and reload parameters. The proposed operation of WNP-2 for a fifth cycle including the use of four 9x9 LTAs, is therefore, acceptable. 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.

Principal Contributor: George Thomas, SRXB

Dated: June 7, 1989

7.0 REFERENCES

1. Letters from G. C. Sorensen (WPPSS) to NRC, Request for Amendment to Technical Specifications - Reload License Amendment (Cycle 5), March 3, 1989, April 20, 1989 and June 1, 1989.

2. WPPSS-EANF-124, "WNP-2 Cycle 5 Reload Summary Report," February 1989.

3. ANF -89-02, "WNP-2 Cycle 5 Reload Analysis Report," Rev. 1, March 1989.

4. ANF-89-01, "WNP-2 Cycle 5 Plant Transient Analysis Report," Rev. 1, March 1989.

5. WPPSS-C-ANF-101, "WNP-2 Cycle 2 Reload Summary Report," February 1986.

6. XN-NF-81-21(A), Rev. 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, January 1982.

7. XN-NF-81-21(A), Revision 1, Supplement 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," March 1985.

8. XN-NF-85-67(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, September 1986.

9. XN-NF-524(P)(A), Revision 1, "Exxon Nuclear Critical Power Methodology for BWRs," November 1983.

10. Letter from H. Bernard (NRC) to G. F. Owsley (ENC), "Acceptance for Referencing of Topical Report XN-NF-512, Revision 1," July 22, 1982.

11. XN-NF-80-19(A), Volume 1 and Volume 1 Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors: Neutronic Methods for Design and Analysis," November 1981.

12. XN-NF-79-71(P), Revision 2 (as supplemented), "Exxon Nuclear Power Plant Transient Methodology," November 1981.

13. XN-NF-84-105(A), Volume 1 (as supplemented), "XCOBRA-T: A Computer Code for BWR Transient Thermal Hydraulic Core Analysis," February 1987.

14. ANF-87-119, "WNP-2 Single Loop Operation Analysis," September 1987.

15. ANF-87-118, "WNP-2 LOCA Analysis for Single Loop Operation," September 1987.

16. WPPSS-EANF-115, "WNP-2 Single Loop Operation Summary Report," February 1988.