SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO.59 TO FACILITY OPERATING LICENSE NO. NPF-21

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

WPPSS NUCLEAR PROJECT NO. 2

DOCKET NO. 50-397

1.0 INTRODUCTION

By letter dated March 7, 1988 (Ref. 1), Washington Public Power Supply System, the licensee, proposed to amend Facility Operating License NPF-21 to support Cycle 4 operation of their Nuclear Plant No. 2 (WNP-2) with Advanced Nuclear Fuels Corporation (ANF) reload fuel. In support of the Cycle 4 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 changes (Ref. 5). The proposed Technical Specification page changes applicable to this amendment were omitted from the March 7, 1988 letter but were provided by supplemental letter dated April 12, 1988. The proposed page changes included with the March 8 submittal were withdrawn by the April 12 submittal.

2.0 EVALUATION

2.1 Reload Description

The WNP-2 Cycle 4 reload will incorporate a total of 152 unirradiated ANF 8x8C fuel assemblies which replace 152 of the General Electric (GE) initial core fuel assemblies. Twenty-four of these assemblies have an average enrichment of 2.72 weight percent U-235 while the remaining 128 are enriched to an assembly average of 2.64 weight percent U-235. The remainder of the core is comprised of 148 ANF 8x8C assemblies loaded for Cycle 3, 128 ANF 8x8C assemblies loaded for Cycle 2 and 336 GE 8x8RP assemblies remaining from the initial core.

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 4 core were approved for operation in previous cycles.

The 128 8x8C ANF reload fuel assemblies manufactured for loading in Cycle 4 are essentially identical to the 24 8x8C ANF reload assemblies originally fabricated for reload in Cycle 3 in all major physical characteristics except U-235 enrichment. Although minor differences in end plug design exist between these two assembly designs, they are essentially interchangeable. All of the reload fuel assemblies are essentially identical to the 8x8C ENC fuel approved for use in Cycle 2 (Ref. 9). Based on this, and on the fuel mechanical design analysis and results which used approved methodologies (Ref. 10), the staff finds the mechanical design of the ANF 8x8C reload fuel for the WNP-2 Cycle 4 reload acceptable.

2.3 Thermal-Hydraulic Design

The ANF thermal-hydraulic methodology and criteria used for the Cycle 4 design and analysis is the same as the previous WNP-2 reloads. These previous reviews concluded that hydraulic compatibility between GE and ANF 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 4 is based on ANF's revised critical power methodology (Ref. 11) 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. 12). Therefore, the proposed safety limit MCPR of 1.06 for all fuel types in this reload is acceptable.

The WNP-2 Technical Specifications include surveillance requirements for detecting and suppressing power oscillations. The staff has required that detect and suppress surveillance be used in regions which have code calculated decay ratios of 0.75 or greater and that operation be forbidden in regions having calculated decay ratios of 0.90 and greater. The NRC Generic Letter 86-02 addressed both GE and ANF stability calculation methodology and concluded that regions of potential instability constituted calculated decay ratios of 0.80 and greater using GE methodology and 0.75 and greater using the ANF methodology. Using the COTRAN code (Ref. 13), ANF has determined that the worst case value of decay ratio is less than 0.75 in the area of the power/flow map bounded by the APRM rod block line at 45% rated flow. In addition, the worst case decay ratio is no greater than 0.90 in the area of allowable low flow operation (detect and suppress region). The bounding power/flow points in the detect and suppress region are the APRM rod block line at 27.6% core flow (46%

power - minimum allowable two pump flow) and the APRM rod block line at 23.8% core flow (42% power - natural circulation flow). The COTRAN calculated decay ratios for these two state points were 0.88 and 0.82, respectively. The licensee's analysis included a 3% margin from the rod block line in order to bound future vendor stability calculations. The power boundary has been linearized between two points, (24% flow, 39% power) and (45% flow, 62% power). The staff finds the stability analyses and surveillance requirements acceptable.

2.4 Nuclear Design

The nuclear design for Cycle 4 has been performed with ANF methodologies previously reviewed and approved (Ref. 13). The fuel loading pattern is given in Figure 4.1 of Reference 3. The beginning-of-cycle (BOC) shutdown margin (SDM) is 1.06% 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 3.46% 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 4 reload core is acceptable.

2.5 Transient Analyses

Core wide transients were analyzed with the COTRANSA computer code (Ref. 14) 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 XCOBRA-T hot channel analysis model (Ref. 15) 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 4 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 was 0.24 for ANF fuel and 0.25 for GE fuel assuming normal scram speed resulting in MCPR limits of 1.30 and 1.31 for ANF fuel and GE fuel, respectively. With Technical Specification scram times, these values become 1.36 for ANF fuel and 1.38 for GE fuel. 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, the MCPR operating limits are 1.35 (ANF fuel) and 1.38 (GE fuel) with an inoperable RPT. For Technical Specification scram times, the MCPR limits are 1.41 (ANF fuel) and 1.44 (GE fuel) with an inoperable RPT. These values are bounded by the proposed Cycle 4 MCPR operating limits and are, therefore, acceptable.

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 licensee has also determined the limiting local transient to be the control rod withdrawal error (CRWE) and has calculated the MCPR operating limit as a function of the rod block monitor (RBM) setpoint. The CRWE was found to be most limiting from BOC up to 3750 MWD/MTU. The delta CPR for the CRWE with a 106% RBM setpoint was 0.17 (ANF fuel) and 0.21 (GE fuel), 0.18 (ANF fuel) and 0.22 (GE fuel) for a 107% RBM setpoint, and 0.20 (ANF fuel) and 0.23 (GE fuel) for a 108% RBM setpoint. Therefore, operation with a 108% RBM setting requires a MCPR limit of 1.26 for ANF fuel and 1.29 for the GE fuel which are bounded by the proposed Cycle 4 MCPR operating limits between BOC and 3750 MWD/MTU. At higher exposures, the CRWE delta CPR values are bounded by the LRWB transient as shown above.

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 (Ref. 16). GE has also performed analyses for the reactor internals and for the GE fuel assembly which showed satisfactory operation at this increased flow (Ref. 17). Based on these analyses and on the similarity between the two fuel types utilized in Cycle 4, the staff concludes that WNP-2 can operate safely with extended core flow up to 106% of rated core flow during Cycle 4. Thus, this increased core flow is acceptable.

2.6 Postulated Accidents

The control rod drop accident (CRDA) yields a value of 149 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 4 residual and reload ANF fuel. These LOCA analyses have covered an acceptable range of conditions, have been performed with approved

methodology and the resulting Technical Specification MAPLHGR values for the ANF fuel are acceptable. Since ANF 8x8C fuel is hydraulically and neutronically compatible with the GE fuel in Cycle 4, the existing GE LOCA analysis and MAPLHGR limits remain applicable to the GE fuel.

3.0 TECHNICAL SPECIFICATION CHANGES

Specification 3/4.2.3: Table 3.2.3-1 and Figure 3.2.3-1 have been revised to reflect Cycle 4 MCPR operating limits. These new limits are based on the Cycle 4 reload safety analysis which has been evaluated and approved in Section 2 and are, therefore, acceptable.

The proposed changes on Table 3.2.3-1 included new operating limits which would apply if the staff were to approve separate license amendment applications for feedwater temperature reduction and for extended conditions for single loop operations. Since this Safety Evaluation Report does not address those separate amendment applications, the new operating limits are not included at this time.

4.0 CORRECTIONS TO THE TECHNICAL SPECIFICATIONS

The license amendment application for the cycle two refueling included several proposed changes which were reviewed by the staff and approved (Amendment 28 issued May 23, 1986) but were omitted from the license amendment as issued.

The licensee applied for removal of section 3/4.1.3.3, Control Rod Average Scram Insertion Times, by letter dated April 24, 1986 (G02-86-367). This deletion also necessitated revision of page v, of the table of contents. The Safety Evaluation issued with Amendment 28 found removal of this specification acceptable, However, the revised pages were not included in the Amendment when issued. Therefore this amendment deletes Section 3/4.1.3.3 and revises page v.

At three places on Table 3.2.3-1 (pages 3/4 2-7) of the Technical Specifications, specification 3.1.3.4 is referred to and the corresponding page number is indicated as page 3/4 1-7. The correct page number is 3/4 1-8. This correction is also made by this amendment.

The licensee requested deletion of Table B 2.1.2-2 (Page B 2-4) by letter dated February 26, 1988 (G02-86-173). Amendment 28 instructed the licensee to remove Page B 2-4. However no replacement page was issued. To provide continuity in the page numbers, a blank page B 2-4 should be inserted in the Technical Specifications.

Page B 2-2 is revised to correct two typographical errors introduced with the issuance of Amendment Number 45. On the fifth line from the bottom of the page and in footnote b, the correct number of the report providing the basis for the uncertainity in the XN-3 correlation is XN-NF-512(A), Rev. 1.

5.0 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 tha 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.

6.0 CONTACT WITH STATE OFFICIAL

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (53 FR 15920) on May 4, 1988 and consulted with the State of Washington. No public comments were received and the State of Washington did not have any comment.

7.0 CONCLUSION

The staff has reviewed the reports submitted for the Cycle 4 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 fourth cycle 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.

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Dated: June 9, 1988

8.0 REFERENCES

1. Letter from G. C. Sorensen (WPPSS) to NRC, Request for Amendment to Technical Specifications - Reload License Amendment (Cycle 4), March 7, 1988.

2. WPPSS-EANF-119, "WNP-2 Cycle 4 Reload Summary Report," February 1988.

3. XN-NF-88-02, "WNP-2 Cycle 4 Reload Analysis Report," January 1988.

4. XN-NF-88-01, "WNP-2 Cycle 4 Plant Transient Analysis Report," January 1988.

5. Attachment to Reference 1, Proposed Changes to WNP-2 Technical Specifications.

6. ANF-87-129(P), "WNP-2 Reload XN-3(WPB3), Cycle 4 Design Report," December 1987.

7. XN-NF-81-21(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," September 1982.

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

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

10. XN-NF-85-67(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," July 1985.

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

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

13. 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.

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

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

16. Letter from J. B. Edgar (ENC) to WPPSS, Supplemental Analysis Results, ENNP-86-0067, April 15, 1986.

17. NEDC-31107, "Safety Review of WPPSS Nuclear Project No. 2 at Core Flow Conditions Above Rated Flow Throughout Cycle 1 and Final Feedwater Temperature Reduction," February 1986.