ATTACHMENT TO LICENSE AMENDMENT NO. 123

TO FACILITY COMBINED LICENSE NO. NPF-92

DOCKET NO. 52-026

Replace the following pages of the Facility Combined License No. NPF-92 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Combined License No. NPF-92

REMOVE	INSERT
7	7

Appendix A to Facility Combined License Nos. NPF-91 and NPF-92

REMOVE	INSERT
5.6-2	5.6-2
5.6-3	5.6-3
5.6-4	5.6-4
5.7-4	5.7-4

(7) Reporting Requirements

(a) Within 30 days of a change to the initial test program described in FSAR Section 14, Initial Test Program, made in accordance with 10 CFR 50.59 or in accordance with 10 CFR Part 52, Appendix D, Section VIII, “Processes for Changes and Departures,” SNC shall report the change to the Director of NRO, or the Director’s designee, in accordance with 10 CFR 50.59(d).

(b) SNC shall report any violation of a requirement in Section 2.D.(3), Section 2.D.(4), Section 2.D.(5), and Section 2.D.(6) of this license within 24 hours. Initial notification shall be made to the NRC Operations Center in accordance with 10 CFR 50.72, with written follow up in accordance with 10 CFR 50.73.

(8) Incorporation

The Technical Specifications, Environmental Protection Plan, and ITAAC in Appendices A, B, and C, respectively of this license, as revised through Amendment No. 123, are hereby incorporated into this license.

(9) Technical Specifications

The technical specifications in Appendix A to this license become effective upon a Commission finding that the acceptance criteria in this license (ITAAC) are met in accordance with 10 CFR 52.103(g).

(10) Operational Program Implementation

SNC shall implement the programs or portions of programs identified below, on or before the date SNC achieves the following milestones:

(a) Environmental Qualification Program implemented before initial fuel load;

(b) Reactor Vessel Material Surveillance Program implemented before initial criticality;

(c) Preservice Testing Program implemented before initial fuel load;

(d) Containment Leakage Rate Testing Program implemented before initial fuel load;

(e) Fire Protection Program

1. The fire protection measures in accordance with Regulatory Guide (RG) 1.189 for designated storage building areas (including adjacent fire areas that could affect the storage area) implemented before initial receipt

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

2.1.1, “Reactor Core SLs”;
3.1.1, “SHUTDOWN MARGIN (SDM)”;
3.1.3, “Moderator Temperature Coefficient (MTC)”;
3.1.5, “Shutdown Bank Insertion Limits”;
3.1.6, “Control Bank Insertion Limits”;
3.2.1, “Heat Flux Hot Channel Factor ($F_Q(Z)$) (Constant Axial Offset Control (CAOC) W(Z) Methodology)”;
3.2.2, “Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)”;
3.2.3, “AXIAL FLUX DIFFERENCE (AFD) (Constant Axial Offset Control (CAOC) Methodology)”;
3.2.5, “On-Line Power Distribution Monitoring System (OPDMS)-Monitored Parameters”;
3.4.1, “RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits”; and
3.9.1, “Boron Concentration.”

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, “Westinghouse Reload Safety Evaluation Methodology,” July 1985 (Westinghouse Proprietary) and WCAP-9273-NP-A (Non-Proprietary).

   (Methodology for Specifications 3.1.1 - Shutdown Margin (SDM), 3.1.3 - Moderator Temperature Coefficient, 3.1.5 - Shutdown Bank Insertion Limits, 3.1.6 - Control Bank Insertion Limits, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor, 3.2.3 - AXIAL FLUX DIFFERENCE, 3.4.1 - RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits, and 3.9.1 - Boron Concentration.)

2a. WCAP-8385, “Power Distribution Control and Load Following Procedures - Topical Report,” September 1974 (Westinghouse Proprietary) and WCAP-8403 (Non-Proprietary).

   (Methodology for Specification 3.2.3 - AXIAL FLUX DIFFERENCE (Constant Axial Offset Control).)

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

2b. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 - Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.

(Methodology for Specification 3.2.3 - AXIAL FLUX DIFFERENCE (Constant Axial Offset Control).)

2c. NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

(Methodology for Specification 3.2.3 - AXIAL FLUX DIFFERENCE (Constant Axial Offset Control).)

3. WCAP-10216-P-A, Revision 1A, “Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification,” February 1994 (Westinghouse Proprietary) and WCAP-10217-A (Non-Proprietary).

(Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology).)

4a. WCAP-16009-P-A, “Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM),” Revision 0, January 2005 (Westinghouse Proprietary) and WCAP-16009-NP-A (Non-Proprietary).

(Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor and Specification 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor.)

4b. APP-GW-GLE-026, “Application of ASTRUM Methodology for Best-Estimate Large-Break Loss-of-Coolant Accident Analysis for AP1000,” Revision 1, February 2009 (Westinghouse Proprietary) and APP-GW-GLE-026-NP (Non-Proprietary).

(Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor and Specification 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor.)

5. WCAP-12472-P-A, (Westinghouse Proprietary) and WCAP-12473-A (Non-Proprietary), “BEACON Core Monitoring and Operations Support System,” August 1994, Addendum 1, May 1996, and Addendum 2, March 2001 and WCAP-12472-P-A (Westinghouse Proprietary) and WCAP-12472-NP-A (Non-Proprietary) Addendum 4, September 2012.

(Methodology for Specification 3.2.5 - OPDMS - Monitored Parameters.)

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. APP-GW-GLR-137, Revision 1, “Bases of Digital Overpower and Overtemperature Delta-T (OPΔT/OTΔT) Reactor Trips,” Westinghouse Electric Company LLC.

   (Methodology for Specification 2.1.1 – Reactor Core Safety Limits.)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Passive Core Cooling Systems limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

   3.4.3, “RCS Pressure and Temperature (P/T) Limits”; and
   3.4.14, “Low Temperature Overpressure Protection (LTOP).”

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

   WCAP-14040-A, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves.” (Limits for LCO 3.4.3 and LCO 3.4.14).

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluency period and for any revision or supplement thereto.

5.6.5 Post Accident Monitoring Report

When a report is required by Condition B of LCO 3.3.17, “Post Accident Monitoring (PAM) Instrumentation,” a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the “As Low As is Reasonably Achievable” principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.