ATTACHMENT TO LICENSE AMENDMENT NO. 151

TO FACILITY COMBINED LICENSE NO. NPF-91

DOCKET NO. 52-025

Replace the following pages of the Facility Combined License No. NPF-91 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-91

REMOVE	INSERT
4	4
7	7

Appendix C to Facility Combined License No. NPF-91

REMOVE	INSERT
C-72	C-72
C-79	C-79

D. The license is subject to, and SNC shall comply with, the conditions specified and incorporated below:

(1) Changes during Construction

(a) SNC may request use of a preliminary amendment request (PAR) process, for license amendments, at any time before a Commission finding under 10 CFR 52.103(g). To use the PAR process, SNC shall submit a written request to the Office of New Reactors (NRO) in accordance with COL-ISG-025, "Changes during Construction under Part 52."

(b) Before NRO's issuance of a written PAR notification, SNC shall submit the license amendment request (LAR). Thereafter, NRO will issue a written PAR notification, setting forth whether SNC may proceed in accordance with the PAR, LAR, and COL-ISG-025. If SNC elects to proceed and the LAR is subsequently denied, SNC shall return the facility to its current licensing basis.

(2) Pre-operational Testing

(a) SNC shall perform the design-specific pre-operational tests identified below:

1. Pressurizer Surge Line Stratification Evaluation (first plant test as identified in UFSAR Section 14.2.9.1.7 Item (d));

2. Automatic Depressurization System Blowdown Test (first three plants test as identified in UFSAR Section 14.2.9.1.3 Item (s)).

(b) SNC shall review and evaluate the results of the tests identified in Section 2.D.(2)(a) of this license and confirm that these test results are within the range of acceptable values predicted or

(7) Reporting Requirements

(a) Within 30 days of a change to the initial test program described in UFSAR 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. 151, 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

2.1.3 Reactor System

Design Description

The reactor system (RXS) generates heat by a controlled nuclear reaction and transfers the heat generated to the reactor coolant, provides a barrier that prevents the release of fission products to the atmosphere and a means to insert negative reactivity into the reactor core and to shutdown the reactor core.

The reactor core contains a matrix of fuel rods assembled into fuel assemblies using structural elements. Rod cluster control assemblies (RCCAs) are positioned and held within the fuel assemblies by control rod drive mechanisms (CRDMs). The CRDMs unlatch upon termination of electrical power to the CRDM thereby releasing the RCCAs. The fuel assemblies and RCCAs are designed in accordance with the principal design requirements.

The RXS is operated during normal modes of plant operation, including startup, power operation, cooldown, shutdown and refueling.

The component locations of the RXS are as shown in Table 2.1.3-3.

1. The functional arrangement of the RXS is as described in the Design Description of this Section 2.1.3.
2. a) The reactor upper internals rod guide arrangement is as shown in Figure 2.1.3-1.
   b) The rod cluster control and drive rod arrangement is as shown in Figure 2.1.3-2.
   c) The reactor vessel arrangement is as shown in Figure 2.1.3-3.
3. The components identified in Table 2.1.3-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.
4. Pressure boundary welds in components identified in Table 2.1.3-1 as ASME Code Section III meet ASME Code Section III requirements.
5. The pressure boundary components (reactor vessel [RV], control rod drive mechanisms [CRDMs], and incore instrument QuickLoc assemblies) identified in Table 2.1.3-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.
6. The seismic Category I equipment identified in Table 2.1.3-1 can withstand seismic design basis loads without loss of safety function.
7. The reactor internals will withstand the effects of flow induced vibration.
8. The reactor vessel direct injection nozzle limits the blowdown of the reactor coolant system (RCS) following the break of a direct vessel injection line.
9. a) The Class 1E equipment identified in Table 2.1.3-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
   b) The Class 1E components identified in Table 2.1.3-1 are powered from their respective Class 1E division.

Table 2.1.3-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
77	2.1.03.06.iii	Not used per Amendment No. 85		
78	2.1.03.07.i	7. The reactor internals will withstand the effects of flow induced vibration.	i) Not Used per Amendment No. 151. ii) A pre-test inspection, a flow test and a post-test inspection will be conducted on the as-built reactor internals.	i) Not Used per Amendment No. 151. ii) The as-built reactor internals have no observable damage or loose parts.
		10. The reactor lower internals assembly is equipped with holders for at least eight capsules for storing material surveillance specimens.	Inspection of the reactor lower internals assembly for the presence of capsules will be performed.	At least eight capsules are in the reactor lower internals assembly.
79	2.1.03.07.ii	Not used per Amendment No. 113		
80	2.1.03.08	8. The reactor vessel direct vessel injection nozzle limits the blowdown of the RCS following the break of a direct vessel injection line.	An inspection will be conducted to verify the flow area of the flow limiting venturi within each direct vessel injection nozzle.	The throat area of the direct vessel injection line nozzle flow limiting venturi is less than or equal to 12.57 in^2.
81	2.1.03.09a.i	Not used per Amendment No. 85		
82	2.1.03.09a.ii	Not used per Amendment No. 85		
83	2.1.03.09b	9.b) The Class 1E components identified in Table 2.1.3-1 are powered from their respective Class 1E division.	Testing will be performed by providing simulated test signals in each Class 1E division.	A simulated test signal exists for Class 1E equipment identified in Table 2.1.3-1 when the assigned Class 1E division is provided the test signal.
84	2.1.03.09c	Not used per Amendment No. 85		
85	2.1.03.10	Not used per Amendment No. 113		
86	2.1.03.11	11. The RPV beltline material has a Charpy upper-shelf energy of no less than 75 ft-lb.	Manufacturing tests of the Charpy V-Notch specimen of the RPV beltline material will be performed.	A report exists and concludes that the initial RPV beltline Charpy upper-shelf energy is no less than 75 ft-lb.