Duke Power Company
Oconee Nuclear Station

Proposed Technical Specification Revision

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3.1.2 Pressurization, Heatup, and Cooldown Limitation

Specification

3.1.2.1 The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited as follows:

Heatup:

Heatup rates and allowable combinations of pressure and temperature shall be limited in accordance with Figure 3.1.2-1A Unit 1
3.1.2-1B Unit 2
3.1.2-1C Unit 3

Cooldown:

Cooldown rates and allowable combinations of pressure and temperature shall be limited in accordance with Figure 3.1.2-2A Unit 1
3.1.2-2B Unit 2
3.1.2-2C Unit 3

3.1.2.2 Leak tests required by Specification 4.3 and ASME Section XI shall be limited to the heatup and cooldown rates and allowable combinations of pressure and temperature provided in Figure 3.1.2-3A Unit 1
3.1.2-3B Unit 2
3.1.2-3C Unit 3

3.1.2.3 For leak test of connected systems required by License Condition 3.H. outlined in Section 4.5.4.2, where the reactor coolant system allowable pressure-temperature limits are controlling, the RCS may be pressurized to the limits set forth in Specification 3.1.2.2.

3.1.2.4 For thermal steady state system hydro tests required by ASME Section XI the system may be pressurized to the limits set forth in Specification 2.2 and 3.1.2.2.

3.1.2.5 The secondary side of the steam generator shall not be pressurized above 237 psig if the temperature of the vessel shell is below 110°F.

3.1.2.6 The pressurizer heatup and cooldown rates shall not exceed 100°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 410°F.

3.1.2.7 Prior to exceeding six (Unit 1)
fifteen (Unit 2)
fifteen (Unit 3)

effective full power years of operation.

Figures 3.1.2-1A (Unit 1), 3.1.2-2A (Unit 1)
3.1.2-1B (Unit 2), 3.1.2-2B (Unit 2)
3.1.2-1C (Unit 3), 3.1.2-2C (Unit 3)

and 3.1.2-3A (Unit 1)
3.1.2-3B (Unit 2)
3.1.2-3C (Unit 3)

and Technical Specification 3.1.2.1, 3.1.2.2 and 3.1.2.3 shall be updated for the next service period in accordance with 10 CFR 50, Appendix G, Section V.B.

3.1.2.8 The updated proposed technical specification referred to in 3.1.2.6 shall be submitted for NRC review at least 90 days prior to the end of the service period for Units 1, 2 and 3.

Bases - Units 1, 2 and 3

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, startup and shutdown operations, and inservice leak and hydrostatic tests. The various categories of load cycles used for design purposes are provided in Table 4.8 of the FSAR.

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10 CFR 50. Results of this analysis, including the actual pressure-temperature limitations of the reactor coolant pressure boundary, are given in BAW-1699 and BAW-1697.

The Figures specified in 3.1.2.1, 3.1.2.2 and 3.1.2.4 prevent the pressure-temperature limit curves for normal heatup, normal cooldown and hydrostatic tests respectively. The limit curves are applicable up to the indicated effective full power years of operation. These curves are adjusted by 25 psi and 10°F for possible errors in the pressure and temperature sensing instruments. The pressure limit is also adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all operating reactor coolant pump combinations.

The cooldown limit curves are not applicable to conditions of off-normal operation (e.g., small LOCA and extended loss of feedwater) where cooling is achieved for extended periods of time by circulating water from the HPI through the core. If core cooling is restricted to meet the cooldown limits under other than normal operation, core integrity could be jeopardized.

The pressure-temperature limit lines shown on the figures specified in 3.1.2.1 for reactor criticality and on the figures referred to in 3.1.2.4 for hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice hydrostatic testing.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating, in accordance with Appendix H to 10 CFR 50, reactor vessel material irradiation surveillance specimens which are installed near the inside wall of this or a similar reactor vessel in the core region, or in test reactors.

The limitation on steam generator pressure and temperature provide protection against nonductile failure of the secondary side of the steam generator. At metal temperatures lower than the RT_{NDT} of +60°F, the protection against nonductile failure is achieved by limiting the secondary coolant pressure to 20 percent of the preoperational system hydrostatic test pressure. The limitations of 110°F and 237 psig are based on the highest estimated RT_{NDT} of +40°F and the preoperational system hydrostatic test pressure of 1312 psig. The average metal temperature is assumed to be equal to or greater than the coolant temperature. The limitations include margins of 25 psi and 10°F for possible instrument error.

The requirements to perform leakage tests of systems outside of containment which could potentially contain radioactivity was established by the NRC following TMI. Oconee performs the leak test of LPI by establishing RCS pressure at about 300 psig and with LPI at this same pressure, checking for leakage. Such a test is within the scope of testing upon which the curves referenced in Specification 3.1.2.2 are based--that is, they are not routine evolutions, such as heatup and cooldown, but rather infrequent leak tests conducted on a refueling outage basis. As such, the hydrostatic/leak test pressure-temperature limitations are applicable for the RCS when performing leak tests of the LPI system.

The spray temperature difference is imposed to maintain the thermal stresses at the pressurized spary line nozzle below the design limit.

REFERENCES

(1) Analysis of Capsule OCII-A from Duke Power Company Oconee Unit 2 Reactor Vessel Materials Surveillance Program, BAW-1699, December 1981.

(2) Analysis of Capsule OCIII-B from Duke Power Company Oconee Unit 3 Reactor Vessel Materials Surveillance Program, BAW-1697, October 1981.

TABLE 4.4-1
LIST OF PENETRATIONS WITH 10CFR50,
APPENDIX J TEST REQUIREMENTS

PENETRATION NUMBER	SYSTEM	TYPE A TEST SYSTEM CONDITION	LOCAL LEAK TEST	REMARKS
22	LPSW from RC Pump motors and lube oil coolers outlet	Not Vented	None required	Note 7b, 9
23	RC Pump seal injection	Not Vented	Type C	Note 5, 7d, 9
24	RB H_2 Analyzer Train A	Note 1	Type C	Note 7c
25	OTSG B Feedwater line	Not Vented	None required	Note 5
26	OTSG A Main steam line	Not Vented	None required	Note 5, MS Stop valve leak test performed
27	OTSG A Feedwater line	Not Vented	None required	Note 5
28	OTSG B Main steam line	Not Vented	None required	Note 5, MS Stop valve leak test performed
29	Quench tank drain line	Note 1	Type C	Note 3, 7b, 9
30 31 32	LPSW for RB Cooling units inlet line	Not Vented	None required	Note 5
33 34 35	LPSW for RB cooling units outlet line	Not Vented	None required	Note 5

TABLE 4.4-1
LIST OF PENETRATIONS WITH 10CFR50,
APPENDIX J TEST REQUIREMENTS

PENETRATION NUMBER	SYSTEM	TYPE A SYSTEM CONDITION	LOCAL LEAK TEST	REMARKS
36 37	RB emergency sump recirculation line	Not Vented	None required	Note 5
38	Quench tank cooler inlet line	Note 1	Type C	Note 2, 7d
39	HP Nitrogen supply	Note 1	None required	Note 3 (manual valves)
(Unit 2, 3 only)	CFT Vent line	Note 1	None required	Note 3 (manual valves)
40	RB emergency sump drain line	Note 1	None required	
41	Instrument air supply & ILRT verification line	Note 1	None required	Note 3 (manual valves)
42	RB H_2 Analyzer Train B	Note 1	Type C	Note 7c
43	OTSG A drain line	Note 1	None required	Note 7b
44	Component cooling to control rod drive inlet line	Note 1	Type C	Note 3, 7d
45	ILRT instrument line	Not Vented	Type C	Note 3, 7a
46	Reactor head-wash filtered water inlet	Note 1	Type B	Note 3, 6a

Reactor Coolant System, verification shall be made that the check and isolation valves in the core flooding tank discharge lines operate properly.

b. The test will be considered satisfactory if control board indication of core flood tank level verifies that all valves have opened.

4.5.1.2 Component Tests

4.5.1.2.1 Pumps

The high pressure and low pressure injection pumps shall be started and operated to verify proper operation in accordance with the requirements of Specification 4.0.4. Acceptable performance will be indicated if the pump starts, operates for 15 minutes, and the discharge pressure and flow are within ± 10 percent of a point on the pump head curve. (Figures 4.5.1-1 and 4.5.1-2)

4.5.1.2.2 Valves - Power Operated

a. Valves LP-17, -18, shall only be tested every cold shutdown unless previously tested during the current quarter.

b. During each refueling outage, low pressure injection pump discharge (engineered safety features) valves, low pressure injection discharge throttling valves, and low pressure injection discharge header crossover valves shall be cycled manually to verify the manual operability of these power-operated valves.

4.5.1.2.3 Check Valves

Periodic individual leakage testing (a) of valves CF-12, CF-14, LP-47 and LP-48 shall be accomplished prior to power operation after every time the plant is placed in the cold shutdown condition for refueling, after each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed. Whenever integrity of these valves cannot be demonstrated, the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily. For the allowable leakge rates and limiting conditions for operation, see Technical Specification 3.1.6.10

Bases

The Emergency Core Cooling Systems are the principle reactor safety featues in the event of loss of coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

(a) To satisfy ALARA requirements, leakge may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

(2) Verification of the engineered safety features function of the Low Pressure Service Water System which supplies coolant to the reactor building coolers shall be made to demonstrate operability of the coolers.

(b) The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly, the appropriate pump breakers have completed their travel, fans are running at half speed, LPSW flow through each cooler exceeds 1400 GPM and air flow through each fan exceeds 40,000 CFM.

4.5.2.2 Component Tests

4.5.2.2.1 Pumps

The reactor building spray pumps shall be started and operated to verify proper operation in accordance with the requirements of Specification 4.0.4. Acceptable performance will be indicated if the pump starts, operates for 15 minutes, and the measured discharge pressure and flow results in a point above the pump head curve. (Figure 4.5.2-1).

Bases

The Reactor Building Coolant System and Reactor Building Spray System are designed to remove heat in the containment atmosphere to control the rate of depressurization in the containment. The peak transient pressure in the containment is not affected by the two heat removal systems. Hence, the basis for the spray pump flow acceptance test is the flow rate required during recirculation (1,000 gpm).

The delivery capability of one reactor building spray pump at a time can be tested by opening the valve in the line from the borated water storage tank, opening the corresponding valve in the test line, and starting the corresponding pump. Pump discharge pressure and flow indication demonstrate performance.

With the pumps shut down and the borated water storage tank outlet closed, the reactor building spray injection valves can each be opened and closed by operator action. With the reactor building spray inlet valves closed, low pressure air or fog can be blown through the test connections of the reactor building spray nozzles to demonstrate that the flow paths are open.

The equipment, piping, valves, and instrumentation of the Reactor Building Cooling System are arranged so that they can be visually inspected. The cooling units and associated piping are located outside the secondary concrete shield. Personnel can enter the Reactor Building during power operations to inspect and maintain this equipment. The service water piping and valves outside the Reactor Building are inspectable at all times. Operational tests and inspections will be performed prior to initial startup.

4.9 EMERGENCY FEEDWATER PUMP AND VALVE PERIODIC TESTING

Applicability

Applies to the periodic testing of the turbine-driven and motor-driven emergency feedwater pumps and associated valves.

Objective

To verify that the emergency feedwater pumps and associated valves are operable.

Specification

4.9.1 Pump Test

The turbine-driven and motor-driven feedwater pumps shall be operated on recirculation to the upper surge tank for a minimum of one hour in accordance with the requirements of Specification 4.0.4.

4.9.2 Valve Test

Automatic valves in the emergency feedwater flow path will be determined to be operable in accordance with the requirements of Specification 4.0.4.

4.9.3 System Flow Test

Prior to Unit operation above 25% Full Power following any modifications or repairs to the emergency feedwater system which could degrade the flow path and at least once per refueling cycle, the emergency feedwater system shall be given either a manual or an automatic initiation signal.

4.9.4 Acceptance Criteria

These tests shall be considered satisfactory if control board indication and visual observation of the equipment demonstrates that all components have operated properly. In addition, during operation of the System Flow Test (Item 4.9.3 above), flow to the steam generators shall be verified by control room indication.

Bases

The monthly testing frequency is sufficient to verify that the emergency feedwater pumps are operable. Verification of correct operation is made both from the control room instrumentation and direct visual observation of the pumps. The parameters which are observed are detailed in the applicable edition of the ASME Boiler and Pressure Vessel Code, Section XI. The System Flow Test verifies correct total system operation following modifications or repairs.

REFERENCES

(1) FSAR, Section 10.2.2

(2) FSAR, Section 14.1.2.8.3