Attachment 3

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 Table 3.1-1 and 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 Table 3.1-2 and 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 Tables 3.1-1, 3.1-2 and Figure 3.1.2-3A Unit 1
3.1.2-3B Unit 2
3.1.2-3C Unit 3

3.1.2.3 Reserved for new specification previously submitted and currently under NRC review.

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 fifteen (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. and V.E.

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.

3.1.2.9 When the temperature of one or more of the RCS cold legs is less than or equal to 325°F, except when the reactor vessel head is removed, then at least one of the following low temperature overpressure protection systems shall be operable:

a. Both Train A and Train B of HP injection shall be inoperable by:

1. For Train A by shutting and deactivating valves HP-26, -409, and -410 by tagging open the valve breakers and tagging the valves in the closed position, or by deactivating pumps HP-A and HP-B and tagging the pump breakers open.

2. For Train B by shutting and deactivating valves HP-27 and -409 by tagging open the valve breakers and tagging the valves in the closed position, or by deactivating pump HP-C and tagging the pump breaker open.

b. The power operated relief valve (PORV) with a lift setting, of less than or equal to 500 psig, a steam bubble or nitrogen blanket in the pressurizer with a pressurizer level less than or equal to 260 inches, and an RCS pressure less than 400 psig.

If neither overpressure protection system is operable then within 1 hour restore at least one system to operable status, or depressurize and establish an RCS vent equivalent to 1 inch ID within the next 12 hours.

c. In the event either the PORV or the RCS vent equivalent to 1 inch ID is used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.6.3 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent(s) on the transient and any corrective action necessary to prevent recurrence.

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 spray line nozzle below the design limit.

The low temperature overprotection systems for Oconee consist of either deactivating both trains of high pressure injection or by having the PORV operable with the condition of the RCS as specified. If either of these is inoperable, the RCS must be depressurized and a vent path equivalent to the PORV flow capability established.

<u>REFERENCES</u>

(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.

(3) Analysis of Capsule OCI-E from Duke Power Company Oconee Unit 1 Reactor Vessel Materials Surveillance Program, BAW-1436, September 1977.

3.6 REACTOR BUILDING

Applicability

Applies to the containment when the reactor is in conditions other than refueling shutdown.

Objective

To assure containment integrity during shutdown (other than refueling shutdown), startup and operation.

Specification

3.6.1 Containment integrity shall be maintained whenever all three (3) of the following conditions exist:

a. Reactor coolant pressure is 300 psig or greater
b. Reactor coolant temperature is 200°F or greater
c. Nuclear fuel is in the core

3.6.2 Containment integrity shall be maintained whenever the reactor is subcritical by less than 1% $\Delta k/k$ or whenever positive reactivity insertions are being made which would result in the reactor being subcritical by less than 1% $\Delta k/k$.

3.6.3 Exceptions to 3.6.1 and 3.6.2 shall be as follows:

a. If either the personnel or emergency hatches become inoperable, except as a result of an inoperable door gasket, the hatch shall be restored to an operable status within 24 hours, or the reactor shall be in cold shutdown within the next 36 hours.

   If a hatch is inoperable due to an inoperable door gasket:

   1. The remaining door of the affected hatch shall be closed and sealed. If the inner door gasket is inoperable, momentary passage (not to exceed 10 minutes for each opening) is permitted through the outer door for repair or test of the inner door, provided that the outer door gasket is leak tested within 24 hours after opening of the outer door.

   2. The hatch shall be restored to operable status within seven days or the reactor shall be in cold shutdown within the next 36 hours.

b. The Reactor Building purge supply and exhaust isolation valves shall be closed except as allowed by Specification 3.6.3.b.1 and 3.6.3.6.2.

   1. The Reactor Building purge system may be operated, with the supply and exhaust isolation valves open, when the Reactor Coolant System temperature is below 250°F and pressure is below 300 psig.

2. For plant conditions when the Reactor Coolant System temperature is above 250°F and pressure is above 300 psig but the reactor is below hot shutdown, one Reactor Building Purge isolation valve on each penetration may be open for testing and/or maintenance per Specification 4.4.4.1 and 3.6.6.

3. For plant conditions other than contained in Specification 3.6.3.b.1, .2 above, with one or more Reactor Building purge valves open, the open valves shall be closed within one hour, or the plant shall be in hot shutdown within 12 hours and within an additional 24 hours, Reactor Coolant System temperature below 250°F and pressure below 300 psig.

c. A containment isolation valve, other than a Reactor Building Purge isolation valve, may be inoperable provided either:

1. The inoperable valve is restored to operable status within four hours.

2. The affected penetration is isolated within four hours by the use of a deactivated automatic valve secured and locked in the isolated position.

3. The affected penetration is isolated within four hours by the use of a closed manual valve or blind flange.

4. The reactor is in the hot shutdown condition within 12 hours and cold shutdown within 24 hours.

3.6.4 The reactor building internal pressure shall not exceed 1.5 psig or five inches of Hg if the reactor is critical.

3.6.5 Prior to criticality following refueling shutdown, a check shall be made to confirm that all manual containment isolation valves which should be closed are closed and tagged.

3.6.6 The combined leakage rate for all penetrations and valves shall be determined in accordance with Specification 4.4.1.2. If, based on the most recent surveillance testing results the combined leakage rate exceeds the specified value and containment integrity is required, then

1) corrective action of Specification 3.6.3.c is met, or

2) repairs shall be initiated immediately and conformance with specified value shall be demonstrated within 48 hours or the reactor shall be in cold shutdown within an additional 36 hours.

Bases

The Reactor Coolant System conditions of cold shutdown assure that no steam will be formed and hence no pressure buildup in the containment if the Reactor Coolant System ruptures.

The selected shutdown conditions are based on the type of activities that are being carried out and will preclude criticality in any occurrence.

The reactor building is designed for an internal pressure of 59 psig and an external pressure 3.0 psi greater than the internal pressure. The design external pressure of 3.0 psi corresponds to a margin of 0.5 psi above the differential pressure that could be developed if the building is sealed with an internal temperature of 120°F with a barometric pressure of 29.0 inches of Hg and the building is subsequently cooled to an internal temperature of 80°F with a concurrent rise in barometric pressure to 31.0 inches of Hg. The weather conditions assumed here are conservative since an evaluation of National Weather Service records for this area indicates that from 1918 to 1970 the lowest barometric pressure recorded is 29.05 inches of Hg and the highest of 30.85 inches of Hg.

Operation with a personnel or emergency hatch inoperable does not impair containment integrity since either door meets the design specifications for structural integrity and leak rate. Momentary passage through the outer door is necessary should the inner door gasket be inoperative to install or remove auxiliary restraint beams on the inner door to allow testing of the hatch. The time limits imposed permit completion of maintenance action and the performance of a local leak rate test when required or the orderly shutdown and cooldown of the reactor. Timely corrective action for an inoperable containment isolation valve is also specified.

When containment integrity is established, the limits of 10 CFR 100 will not be exceeded should the maximum hypothetical accident occur.

The Reactor Building purge system was designed to allow cleanup of the Reactor Building atmosphere. It is normally operated during a unit shutdown which will require entry into the Reactor Building. It is used to reduce the MPC levels within the Building atmosphere, thus reducing overall personnel exposure. At times, certain safety related functions necessitate entry into the Reactor Building prior to cold shutdown conditions. These include isolation of leaking primary coolant system valves and visual inspections following outages. Use of the purge system tends to minimize any personnel exposure while not significantly contributing to overall plant risk.

REFERENCES

FSAR, Section 3.8

4.2.5 The power operated relief valve (PORV) is used for low temperature overpressure protection of the RCS and shall be demonstrated operable by:

a. Performing an operability test prior to each startup from cold shutdown.

b. Performing a calibration of the actuation circuit each refueling outage.

c. Performing an inspection of the PORV at least once every two refueling cycles.

4.2.6 Each shift, the RCS vent(s) (as defined in Specification 3.1.2.9) shall be verified to be open, if the vent(s) is(are) being used for overpressure protection. If the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then these valves will be verified open at least once per 31 days.

Bases

The surveillance program has been developed to comply with the applicable edition of Section XI and addenda of the ASME Boiler and Pressure Vessel Code, Inservice Inspection of Nuclear Reactor Coolant Systems, as required by 10 CFR 50.55(a) to the extent practicable within limitations of design, geometry and materials of construction. The program places major emphasis on the area of highest stress concentrations and on areas where fast neutron irradiation might be sufficient to change material properties.

The number of reactor vessel specimens and the frequencies for removing and testing these specimens are provided to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

4.4.4 Reactor Building Purge System

Applicability

Applies to the Reactor Building Purge System.

Objective

To verify that the Reactor Building Purge System is operable.

Specification

4.4.4.1 Each shutdown, when the purge valves have been operated, leakage integrity tests shall be performed on the containment purge isolation valves after final closing and prior to going above hot shutdown. If the purge valves have not been operated, leakage integrity tests shall be performed prior to going above hot shutdown unless such tests have been conducted within the preceeding six months. If the acceptance criteria of Specification 4.4.1.2.3 are not met, Specification 3.6.6 shall apply. Unit shutdown to conduct the test and/or effect repairs is specifically not required.

4.4.4.2 Monthly, when the unit is above 250°F and 300 psig, the containment purge isolation valves shall be verified closed.

4.4.4.3 Each refueling the valve seals of the containment purge isolation valves shall be visually inspected and adjusted or replaced as appropriate.

4.4.4.4 Prior to use of the purge system at conditions between cold shutdown and 250°F and 300 psig, the isolation valves shall be exercise tested in accordance with the requirements (except test frequency) of the applicable edition of the ASME Boiler and Pressure Vessel Code, Section XI.

4.4.4.5 The pneumatically operated purge isolation valves shall be verified to close in response to a control signal from RIA-45 when the system is tested prior to refueling operations per Specification 3.8.10.

Bases

Leakage integrity tests of the purge supply and isolation valves are conducted in order to identify excessive degradation of the resilient seals. Excessive leakage past resilient seals is typically caused by severe environmental conditions and/or wear due to frequent use.

The pneumatically operated purge isolation valves are tested prior to refueling operations because the only automatic isolation system in service at refueling is through RIA-45, which only closes the pneumatic isolation valves.

TABLE 6.1-1
MINIMUM OPERATING SHIFT REQUIREMENTS
(With Fuel in the Three Reactor Vessels)

	One Unit Operating*	Two Units Operating*	All Units Operating*	All Units Shutdown
Shift Supervisor (SRO)	1	1	1	1
Additional SRO	1	2**	2	None
Shift Technical Advisor	1	1	1	None
Reactor Operator	4	5**	5	3
Nuclear Equipment Operator	2	5	4	3

* Above cold shutdown
** Only one SRO and four Reactor Operators required if both units are operated from one Control Room.

ADDITIONAL REQUIREMENTS

1. One licensed operator per unit shall be in the Control Room at all times when there is fuel in the reactor vessel.

2. Two licensed operators shall be in the Control Room during startup and scheduled shutdown of a reactor.

3. At least one licensed operator shall be in the reactor building when fuel handling operations in the reactor building are in progress.

4. An operator holding a Senior Reactor Operator license and assigned no other operational duties shall be in direct charge of refueling operations.

5. At least one person per shift shall have sufficient training to perform routine health physics requirements.

6. If the computer for a reactor is inoperable for more than eight hours, an operator, in addition to those required above, shall supplement the shift crew.

7. A fire brigade of 5 members shall be maintained on site at all times. This excludes 3 members of the minimum operating shift requirements that are required to be present in the control rooms.

8. An operator holding a Senior Reactor Operator's license shall be in in the Control Room from which the unit is operated whenever the unit is above cold shutdown.

9. Temporary deviations from the requirements of Table 6.1-1 may be allowed in cases of sudden illness, injury or other similar emergencies provided replacement personnel are notified immediately and are on site as soon as possible to return shift manning to minimum.