3.1.2 PRESSURIZATION, HEATUP, AND COOLDOWN LIMITATIONS

Specification

3.1.2.1 Inservice Leak and Hydrostatic Tests:

Pressure temperature limits for the first five EFP years of inservice leak and hydrostatic tests are given in Figure 3.1.2-3. Heatup and cooldown rates shall be restricted according to the rates specified in Figure 3.1.2-3.

3.1.2.2 Heatup Cooldown:

For the first five EFP years of power opeations, the reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1.2-1 and Figure 3.1.2-2 respectively. Heatup and cooldown rates shall not exceed the rates stated on the associated figure.

3.1.2.3 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is below 130°F.

3.1.2.4 The pressurizer heatup and cooldown rates shall not exceed 100°F in any 1-hour period.

3.1.2.5 The spray shall not be used if the temperature difference between the pressurizer and spray fluid is greater than 410°F.

3.1.2.6 Prior to exceeding five effective full power years of operation, Figures 3.1.2-1, -2, and -3 shall be updated for the next service period in accordance with 10 CFR 50, Appendix G, Section V.B. The highest predicted adjusted reference temperature of all the beltline materials shall be used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction shall be submitted for NRC staff review in accordance with Specification 3.1.2.7.

3.1.2.7 The updated proposed technical specifications 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. Appropriate additional NRC review time shall be allowed for proposed technical specifications submitted in accordance with 10 CFR 50, Appendix G, Section V.C.

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3.1.2.8 Emergency/Faulted Operation:

In the emergency/faulted condition when there is no forced or natural circulation in the reactor coolant system and there is high pressure injection and/or makeup addition, the Reactor Coolant System temperature and pressure shall be limited in accordance with the limit line shown on Figure 3.1.2-4. Under the above emergency/faulted conditicns, Figure 3.1.2-2 will not apply. 64 Rev.1

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The maximum allowable pressure is taken to be the lowest pressure of the three calculated pressures. The pressure limit is adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all reactor coolant pump combinations. The limit curves were prepared based upon the most limiting adjusted reference temperature of all the beltline region materials at the end of the fifth effective full power year.

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 installed near the inside wall of the reactor vessel in the core area. Because the neutron energy spectra at the specimen location and at the vessel inner wall location are essentially the same, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The limit curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The unirradiated impact properties of the beltline region materials, required by Appendices G and H to 10 CFR 50, were determined for those materials for which sufficient amounts of material were available. The adjusted reference temperatures are calculated by adding the radiation-induced ΔRT_{NDT} and the unirradiated RT_{NDT}. The predicted ΔRT_{NDT} are calculated using the respective neutron fluence and copper and phosphorus contents in accordance with Reg. Guide 1.99.

The assumed RT_{NDT} of the closure head region is 60°F and the outlet nozzle steel forgings is 60°F.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME code requirements.

The limitations during emergency/faulted operation when all reactor coolant flow and all feedwater flow is lost to the OTSG's are established to take into consideration that HPI gives false cold leg temperatures. This transient is controlled by Figure 3.1.2-4 and the vessel beltline temperature is calculated using incore thermocouples and subtracting 150°F for conservatism. When the coolant flow or feedwater flow is re-established, a four hour transition period will be allowed to progress from Figure 3.1.2-4 to Figure 3.1.2-2. 64 Rev.1

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3.4.2.2 Both auxiliary feed pumps are operable. One auxiliary feed pump may be out of service for maintenance for a period of 48 hours.

3.4.2.3 An auxiliary feedwater flow path is operable, otherwise the reactor shall not remain critical. 64 Rev.1

Bases

The feedwater system and the turbine bypass system are normally used for decay heat removal and cooldown above 280 F. Feedwater makeup is supplied by operation of a condensate pump and main feedwater pump. In the event of complete loss of electrical power, feedwater is supplied by a turbine driven auxiliary feedwater pump which takes suction from the condensate storage tank. Steam relief would be through the system's atmospheric relief valves.

If neither main feed pump is available, feedwater can be supplied to the steam generators by an auxiliary feedwater pump and steam relief would be through the turbine bypass system to the condenser.

In order to heat the reactor coolant system above 280 F the maximum steam removal capability required is 4-1/2 percent of rated power. This is the maximum decay heat rate at 30 seconds after a reactor trip. The requirement for two steam system safety valves per steam generator provides a steam relief capability of over 10 percent per steam generator (1,341,938 lb/h). In addition, two turbine bypass valves to the condenser or two atmospheric dump valves will provide the necessary capacity.

The 250,000 gallons of water in the condensate storage tank is the amount needed for cooling water to the steam generators for a period in excess of one day following a complete loss of all unit ac power.[1]

The minimum relief capacity of seventeen steam system safety valves is 13,329,163 lb/hr.[2] This is sufficient capacity to protect the steam system under the design overpower condition of 112 percent.[3]

REFERENCES

(1) FSAR paragraph 14.1.2.8.4

(2) FSAR paragraph 10.3.4

(3) FSAR Appendix 3A, Answer to Question 3A.5

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4.8 AUXILIARY FEEDWATER PUMP PERIODIC TESTING

Applicability

Applies to the periodic testing of the turbine and motor driven auxiliary feedwater pumps.

Objective

To verify that the auxiliary feedwater pump and associated valves are operable.

Specification

4.8.1 At least every 92 days on a staggered test basis at a time when the average reactor coolant system temperature is $\geq 305^{o}F$, the turbine/motor driven and motor driven auxiliary feedwater pumps shall be operated on recirculation to the condenser to verify proper operation. 64 Rev.1

The 92 day test frequency requirement shall be brought current within 72 hours after the average reactor coolant system temperature is $\geq 305^{o}F$.

Acceptable performance will be indicated if the pump starts and operates for fifteen minutes at the design flow of 780 gpm. This flow will be verified using tank level decrease and pump differential pressure.

4.8.2 All valves, including those that are locked, sealed, or otherwise secured in position, are to be inspected monthly to verify they are in the proper position. 64 Rev.1

Bases

The quarterly test frequency will be sufficient to verify that the turbine/motor driven and motor driven auxiliary feedwater pumps are operable. Verification of correct operation will be made both from the control room instrumentation and direct visual observation of the pumps.

REFERENCE

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