

VYNPS

3.6 LIMITING CONDITIONS FOR OPERATION

C. Coolant Leakage

- 1a. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant leakage into the primary containment shall not exceed 25 gpm.
- b. While in the run mode, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm within any 24 hour period.
2. Both the sump and air sampling systems shall be operable during power operation. From and after the date the one of these systems is made or found inoperable for any reason, reactor operation is permissible only during succeeding seven days.
3. If these conditions cannot be met, initiate an orderly shutdown and the reactor shall be in the cold shutdown condition within 24 hours.

D. Safety and Relief Valves

1. During reactor power operating conditions and whenever the reactor coolant pressure is greater than 120 psig and temperature greater than 350°F, both safety valves shall be operable. The relief valves shall be operable, except that if one relief valve is inoperable, reactor power shall be immediately reduced to and maintained at or below 95% of rated power.
2. If Specification 3.6.D.1 is not met, initiate an orderly shutdown and the reactor coolant pressure shall be below 120 psig and 350°F within 24 hours.

4.6 SURVEILLANCE REQUIREMENTS

C. Coolant Leakage

1. Reactor coolant system leakage, for the purpose of satisfying Specification 3.6.C.1, shall be checked and logged once per shift, not to exceed 12 hours.

D. Safety and Relief Valves

1. Operability testing of Safety and Relief Valves shall be in accordance with Specification 4.6.E. The lift point of the safety and relief valves shall be set as specified in Specification 2.2.B.

VYNPS

3.6 LIMITING CONDITIONS FOR OPERATION

E. Structural Integrity and Operability Testing

The structural integrity and the operability of the safety-related systems and components shall be maintained at the level required by the original acceptance standards throughout the life of the plant.

4.6 SURVEILLANCE REQUIREMENTS

E. Structural Integrity and Operability Testing

1. Inservice inspection of safety-related components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i). Inservice inspection of piping, identified in NRC Generic Letter 88-01, shall be performed in accordance with the staff positions on schedule, methods, and personnel and sample expansion included in the Generic Letter.
2. Operability testing of safety-related pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

3.6 & 4.6 (Continued)

greater than the limit specified for unidentified leakage; the probability is small that imperfections or cracks associated with such leakage would grow rapidly. Leakage less than the limit specified can be detected within a few hours utilizing the available leakage detection systems. If the limit is exceeded and the origin cannot be determined in a reasonably short time the plant should be shut down to allow further investigation and corrective action.

The 2 gpm increase limit in any 24 hour period for unidentified leaks was established as an additional requirement to the 5 gpm limit by Generic Letter 88-01, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping."

The removal capacity from the drywell floor drain sump and the equipment drain sump is 50 gpm each. Removal of 50 gpm from either of these sumps can be accomplished with considerable margin.

D. Safety and Relief Valves

Parametric evaluations have shown that only three of the four relief valves are required to provide a pressure margin greater than the recommended 25 psi below the safety valve actuation settings as well as a MCPR > 1.06 for the limiting overpressure transient below 98% power. Consequently, 95% power has been selected as a limiting power level for three valve operation. For the purpose of this limiting condition a relief valve that is unable to actuate within tolerance of its set pressure is considered to be as inoperable as a mechanically malfunctioning valve.

Experience in safety valve operation shows that a testing of 50% of the safety valves per refueling outage is adequate to detect failures or deterioration. The tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code as $\pm 1\%$ of design pressure. An analysis has been performed which shows that with all safety valves set 1% higher the reactor coolant pressure safety limit of 1375 psig is not exceeded.

E. Structural Integrity and Operability Testing

A pre-service inspection of the components listed in Table 4.2-4 of the FSAR will be conducted after site erection to assure freedom from defects greater than code allowance; in addition, this serves as a reference base for further inspections. Prior to operation, the reactor primary system will be free of gross defects. In addition, the facility has been designed such that gross defects should not occur throughout plant life. The inservice inspection and testing programs are performed in accordance with 10CFR50, Section 50.55a(g) except where specific relief has been granted by the NRC. These inspection and testing programs provide further assurance that gross defects are not occurring and ensure that safety-related components remain operable.

3.6 & 4 (CONT'D)

The type of inspection planned for each component depends on location, accessibility, and type of expected defect. Direct visual examination is proposed wherever possible since it is sensitive, fast, and reliable. Magnetic particle and liquid penetrant inspections are planned where practical, and where added sensitivity is required. Ultrasonic testing and radiography shall be used where defects can occur on concealed surfaces.

Generic Letter 88-01 established the NRC position for in-service inspection of BWR austenitic stainless steel piping susceptible to Intergranular Stress Corrosion Cracking (IGSCC).

The in-service inspection and testing programs presented at this time are based on a thorough evaluation of present technology and state-of-the-art inspection and testing techniques.

F.

Jet Pumps

Failure of a jet pump nozzle assembly hold down mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within $\pm 5\%$, the flow rates in both recirculation loops will be verified by main Control Room monitoring instruments. If the two flow rate values do not differ by more than 10%, riser and nozzle assembly integrity has been verified. If they do differ by 10% or more the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured value of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10% or more (with the measured value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the plant shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115% to 120% for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3% to 6%) in the total core flow measure. This decrease, together with the loop flow increase, would result in a leak of correlation between measured and derived core flow rate.