operator shall start to place the reactor in the hot shutdown condition utilizing normal operating procedures.

3.6.3 Containment Automatic Isolation Trip Valves

The following exceptions apply only to automatic containment isolation valves required to be closed during accident conditions and which are either redundant or installed in a line which is part of a closed system within containment.

With one or more of the automatic containment isolation trip valves inoperable, either:

a. Restore the inoperable valve(s) to operable status within 4 hours, or

b. Isolate the affected penetration(s) within 4 hours by use of a deactivated automatic valve(s) secured in the isolation position(s), or

c. Isolate the affected penetration(s) within 4 hours by use of a closed manual valve(s) or blind flange(s), or

d. Be in cold shutdown within the next 36 hours.

3.6.4 Containment Purge and Vent Valves

The Containment Purge and Vent Valves (V12-6, V12-7, V12-8, V12-9, V12-10, V12-11, V12-12, and V12-13) shall be capable of shutting in ≤ 2 seconds. If any valve is not capable of shutting in that time, the penetration will be isolated as in Specification 3.6.3b and the defective valve repaired, tested, and documented before being returned to normal operation. Valves V12-6, V12-7, V12-8, and V12-9 will not be opened greater than 70° while above cold shutdown.

Basis

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

The shutdown margins are selected based on the type of activities that are being carried out. The 10% $\Delta k/k$ shutdown margin during refueling precludes criticality under any circumstances, even though fuel is being moved. When the reactor head is not to be removed, the specified cold shutdown margin of 1% $\Delta k/k$ precludes criticality in any occurrence.

Regarding internal pressure limitations, the containment design pressure of 42 psig would not be exceeded if the internal pressure before a major loss-of-coolant accident were as much as 4 psig.[1] The containment is designed to withstand an internal vacuum of 2.0 psi.[2]

Containment Purge and Vent Valves must shut within the specified time limit to limit post LOCA thyroid dose and to limit the increase in peak clad temperature due to reduction in Containment internal pressure.

References

(1) FSAR Section 14.3.4
(2) FSAR Section 5.1.2.3

c. Notification of the pending test, either of a sample tendon or the containment structural test, along with detailed acceptance criteria shall be forwarded to the Nuclear Regulatory Commission two months prior to the actual test. Within six months of conducting the test, a report and evaluation shall be submitted to the NRC.

4.4.5 Containment Purge and Vent Valves

The Containment Purge Valves (V12-6, V12-7, V12-8, V12-9, V12-10, V12-11, V12-12, and V12-13) shall be tested at each refueling to determine their capability to shut within the time limit specified in Specification 3.6.4.

Basis

The containment is designed for an accident pressure of 42 psig.[1] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure and a maximum temperature of 120°F. With these initial conditions, the temperature of the steam-air mixture at the peak accident pressure of 42 psig is 263°F.

Prior to initial operation, the containment will be strength tested at 48.3 psig and then will be leak-tested. The acceptance criterion for this preoperational leakage rate test has been established as 0.08% per 24 hours of containment atmosphere at 42 psig. This acceptable leakage rate is equivalent to a 0.1% of the containment steam-air atmosphere per 24 hours at 42 psig and 263°F. This leakage rate is consistent with the construction of the containment,[2] which is equipped with a penetration pressurization system which pressurizes penetrations, double gasketed seals, and some isolation valve spaces, and which contains channels over all containment liner welds. These channels were independently leak-tested during construction.

The safety analysis has been performed on the basis of a leakage rate of 0.10% per 24 hours at 42 psig and 263°F. With this leakage rate and with minimum containment engineered safety features operating, the public exposure would not exceed 10 CFR 100 guideline values in the event of the design basis accident.[3]

The performance of a periodic integrated leak rate test during plant life provides a current assessment of potential leakage from the containment in case of an accident that would pressurize the interior of the containment.

In order to provide a realistic appraisal of the integrity of the containment under accident conditions, this periodic leakage rate test is to be performed without preliminary leak detection surveys or leak repairs and containment isolation valves are to be closed in the normal manner.

The test pressure of 21 psig for the periodic integrated leakage rate test is sufficiently high to provide an accurate measurement of the leakage rate and it duplicates the pre-operational leak rate test at 21 psig. The factor of 0.8 relates the measured leakage of air to the potential leakage of a steam-air mixture. The specification also allows for possible deterioration of the leakage rate between tests, by requiring that only 75% of the allowable leakage rates actually be measured. The basis for these deterioration allowances is arbitrary, but is believed to be conservative and will be confirmed or denied by periodic testing. If indicated to be necessary, the deterioration allowances will be altered based on experience.

The minimum duration of 24 hours for the integrated leakage rate test is established to attain the desired level of accuracy. This also allows for daily cyclic variations in temperature and thermal radiation.

Secondly, the penetration pressurization system is in-service continuously to monitor leakage from potential leak paths, such as penetrations, double gasketed seals, and spaces between certain containment isolation valves. Total leakage from the system is measured by summing the recorded flows in each of the four pressurization headers. A leak would be expected to build up slowly and would therefore be noted before design leakage limits are exceeded. Therefore, remedial action can be taken before the limit is reached.

A flow sensing device is located in each of the headers supplying make-up air to the four pressurized zones. A leakage rate alarm is provided in each of the four indicating channels to alert the operator in the control room. The flow measurement accuracy is within $\pm 1\%$. A flow of 0.04% of the containment volume per day at 42 psig is approximately 0.58 ft^3/minute (2.34 scfm). The flowmeters are capable of indicating leakage well within these limits.

The specified frequency of periodic integrated leak rate tests is based on the following major considerations. First is the low probability of leaks in the liner, because of (a) the test of the leak tightness of the welds during erection; (b) conformance of the complete containment to a low leakage rate limit at 42 psig during preoperational testing which is consistent with 0.1% leakage at design basis accident (DBA) conditions; and (c) absence of any significant stresses in the liner during reactor operation.

Containment isolation valves are designed to incorporate positive barriers to prevent or minimize leakage through the valves under design basis accident conditions. Several isolation valves are pressurized by the penetration pressurization system to prevent leakage. The remaining valves either receive Isolation Seal Water System water or are installed in systems that are part of a closed system within the containment or operate at system pressures greater than 42 psig in the post-accident condition. These design features provide positive means to prevent containment leakage through the containment isolation valves.

The limiting leakage rates from the recirculation heat removal system are judgment values based primarily on assuring that the components could operate without mechanical failure for a period on the order of 200 days after a design basis accident. The test pressure, 350 psig, achieved either by normal system operation or hydrostatically testing, gives an adequate margin over the highest pressure within the system after a design basis accident.

A recirculation heat removal system leakage of 2 gal/hr will limit off-site exposure due to leakage to insignificant levels relative to those calculated for leakage directly from the containment in the design basis accident.

In case of failure to meet the acceptance criteria for leakage from the recirculation heat removal system or the penetrations, it may be possible to effect repairs within a short time. If so, it is considered unnecessary and unjustified to shut down the reactor.

The emergency core cooling system sump suction line penetration consists of an expansion joint welded to a pipe and sleeve going through the containment wall. The penetration pressurization system continuously pressurizes the annulus between the suction line and the guard pipe. Failure of the suction line or penetration would be identified in the control room by a flow alarm on the penetration pressurization system. The bellows expansion joint is welded to the suction line and guard pipe and meets the requirements of Section III of the ASME Boiler and Pressure Vessel Code.(7)

The surveillance tendons consist of two tendons similar to the service tendons, but shorter in length. Each tendon consists of six- 1-3/8" bars in a 6-inch pipe sheath, with anchor plates, prestressing hardware and grout pipe identical except for length to that of the tendons installed in the containment. They are embedded in a section of concrete approximately the same environment as that of the service tendons. When a tendon is removed for inspection it will be sent to a commercial laboratory qualified to perform material tests and analyses. Visual inspection for corrosion and tensile tests will be performed to determine if any significant changes have occurred.(8)

The containment structural test pressure will be 42 psig. This pressure is selected for consistency with the initial acceptance test. The initial acceptance test pressure was selected so as to impose, insofar as practical with a static pressure test, maximum stresses on the principal strength elements reasonably consistent with those stresses imposed by the design basis conditions. The initial acceptance test permits verification that the structural response is consistent with the design. The periodic tests thereafter permits verification that the structural response is consistent with the initial response and will thus provide a demonstration of the continued integrity of the structure.

The structural test intervals selected concentrate the test program in the period during the life of the plant where corrosion of the bar tendons, as opposed to the more sensitive wire tendons, would be of greater concern. The two sample tendons provide a check on the possible presence of a corrosive mechanism not yet sufficiently advanced to affect the results of a pressure test. The sample tendons are capable of being removed at any time. The pressure tests may be coordinated with an in-service inspection planned at approximately the same time.

The requirements for structural tests and the acceptance criteria are subject to review and modification based upon the results obtained from the initial pre-service proof test. The results of this test shall be provided to the NRC following completion of the test. The report shall include a discussion of the criteria upon which the adequacy of the containment structure was judged.

Containment Purge and Vent Valves are tested to assure they close within the specified time. Closing time of the valves affects post LOCA thyroid dose and peak clad temperature.

References

(1) FSAR Section 5.1.2.3
(2) FSAR Section 5.6.2.2
(3) FSAR Section 14.3.5
(4) Deleted
(5) Deleted
(6) FSAR Section 5.2.2
(7) FSAR Volume 4, Tab VI, Question 6-5
(8) FSAR Volume 4, Question III.E.2