TECHNICAL SPECIFICATIONS

DEFINITIONS

The following terms are defined for uniform interpretation of these Specifications.

REACTOR OPERATING CONDITIONS

Rated Power

A steady state reactor core output of 1524 MWt.

Reactor Critical

The reactor is considered critical for purposes of administrative control when the neutron flux logarithmic range channel instrumentation indicates greater than 10^{-4}% of rated power.

Power Operation Condition (Operating Mode 1)

The reactor is in the power operation condition when it is critical and the neutron flux power range instrumentation indicates greater than 2% of rated power.

Hot Standby Condition (Operating Mode 2)

The reactor is considered to be in a hot standby condition if the average temperature of the reactor coolant (Tavg) is greater than 515°F, the reactor is critical, and the neutron flux power range instrumentation indicates less than 2% of rated power.

Hot Shutdown Condition (Operating Mode 3)

The reactor is in a hot shutdown condition if the average temperature of the reactor coolant (Tavg) is greater than 515°F and the reactor is subcritical by at least the amount defined in Paragraph 2.10.2.

Action statements (5)b. and c. include the removal of power from a closed block valve to preclude any inadvertent opening of the block valve at a time the PORV may not be closed due to maintenance. However, the applicability requirements of the LCO to operate with the block valve(s) closed with power maintained to the block valve(s) are only intended to permit operation of the plant for a limited period of time not to exceed the next refueling shutdown (Mode 5), so that maintenance can be performed on the PORV(s) to eliminate the seat leakage condition.

To determine the maximum steam flow, the only other pressure relieving system assumed operational is the main steam safety valves. Conservative values for all systems parameters, delay times and core moderator coefficients are assumed. Overpressure protection is provided to portions of the reactor coolant system which are at the highest pressure considering pump head, flow pressure drops and elevation heads.

If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve lift pressure would be less than half of the capacity of one safety valve. This specification, therefore, provides adequate defense against overpressurization when the reactor is subcritical.

Performance of certain calibration and maintenance procedures on safety valves requires removal from the pressurizer. Should a safety valve be removed, either operability of the other safety valve or maintenance of at least one nozzle open to atmosphere will assure that sufficient relief capacity is available. Use of plastic or other similar material to prevent the entry of foreign material into the open nozzle will not be construed to violate the "open to atmosphere" provision, since the presence of this material would not significantly restrict the discharge of reactor coolant.

The total relief capacity of the ten main steam safety valves is 6.606×10^6 lb/hr. If, following testing, the as found setpoints are outside +/-1% of nominal nameplate values, the valves are set to within the +/-1% tolerance. The main steam safety valves were analyzed for a total loss of main feedwater flow while operating at RATED POWER[3] to ensure that the peak secondary pressure was less than 1100 psia, the ASME Section III upset pressure limit of 10% greater than the design pressure. At RATED POWER, sufficient relief valve capacity is available to prevent overpressurization of the steam system on loss-of-load conditions.[4] These analyses are based on a minimum of four-of-five operable main steam safety valves on each main steam header.

The power-operated relief valve low setpoint will be adjusted to provide sufficient margin, when used in conjunction with Technical Specification Sections 2.1.1 and 2.3, to prevent the design basis pressure transients from causing an overpressurization incident. Limitation of this requirement to scheduled cooldown ensures that, should emergency conditions dictate rapid cooldown of the reactor coolant system, inoperability of the low temperature overpressure protection system would not prove to be an inhibiting factor. The effective full flow area of an open PORV is 0.94 in^2.

Removal of the reactor vessel head provides sufficient expansion volume to limit any of the design basis pressure transients. Thus, no additional relief capacity is required.

References

(1) Article 9 of the 1968 ASME Boiler and Pressure Vessel Code, Section III
(2) USAR, Section 14.9
(3) USAR, Section 14.10
(4) USAR, Sections 4.3.4, 4.3.9.5

Basis

The containment is designed for an accident pressure of 60 psig.[2] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure and a maximum temperature of about 120°F. With these initial conditions the temperature of the steam-air mixture at the peak accident pressure of 60 psig is 288°F.

Prior to initial operation, the containment was strength-tested at 69 psig and then was leak tested. The design objective of the pre-operational leakage rate test has been established as 0.1% by weight for 24 hours at 60 psig. This leakage rate is consistent with the construction of the containment, which is equipped with independent leak-testable penetrations and contains channels over all inaccessible containment liner welds, which were independently leak-tested during construction.

Safety analyses have been performed on the basis of a leakage rate of 0.1% of the free volume per day of the first 24 hours following the maximum hypothetical accident. With this leakage rate, at RATED POWER, and with minimum containment engineered safety systems for iodine removal in operation (one air cooling and filtering unit), the public exposure would be well below 10 CFR Part 100 values in the event of the maximum hypothetical accident.[3] The performance of an integrated leakage rate test and performance of local leak rate testing of individual penetrations at periodic intervals during plant life provides a current assessment of potential leakage from the containment.

The reduced pressure (5 psig) test on the PAL is a conservative method of testing and provides adequate indication of any potential containment leakage path. The test is conducted by pressurizing between two resilient seals on each door. The test pressure tends to unseat the resilient seals which is opposite to the accident pressure that tends to seat the resilient seals. A periodic test ensures the overall PAL integrity at 60 psig.

The integrated leakage rate test (Type A test) can only be performed during refueling shutdowns.