Attachment A

Proposed Technical Specification Change

Proposed Change

Reference is made to Pilgrim Nuclear Power Station Unit #1 Technical Specification Appendix A, Section 3.7 "Limiting Conditions for Operation", Items 3.7.A.1.i, 3.7.A.1.j, 3.7.A.1.k and 3.7.A.1.m. These changes are shown on Page 152A attached. Also change Section 4.7 "Surveillance Requirements" by adding Item 4.7.A.1.f as shown on Page 152A attached. A change to the "Bases 3.7 and 4.7" will be necessary as shown on Pages 166 and 171 attached.

Reason for Change

The reason for this change is that the ventheader downcomers and internal piping are to be cut to conform to NRC Acceptance Criteria, Appendix A to NUREG 0661. By cutting the 96 ventheader downcomers the submerged drag loads will be reduced and by cutting the internal piping the pool swell and drag loads will be reduced. This change will necessitate a decrease in the drywell-suppression chamber differential pressure setpoint.

Safety Considerations

Compliance with NUREG 0661, 10 CFR 50.55A and ASME Code Section III and Section XI, 1977 Edition Summer 1978 Addenda will reduce the loads on the torus and associated components. A safety evaluation has been done for the proposed work and it is not considered to constitute an unreviewed safety question. These changes have been reviewed and approved by the Operations Review Committee and reviewed by the Nuclear Safety Review and Audit Committee.

Clarification

This change is necessary because 9" are being cut from each of the downcomers as part of the Mark I Containment LTP modifications. This work will be accomplished during the 1981 fall Refueling Outage.

Schedule of Change

This change will be put into effect upon receipt of approval by the Commission.

Fee Determination

Pursuant to 10 CFR Section 170.12, Boston Edison proposes this change as a Class III.

3.7 CONTAINMENT SYSTEMS (Cont'd)

h. During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cool down rates if the pool temperature reaches 120°F.

i. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.17 psid, except as specified in j and k.

j. The differential pressure shall be established within 24 hours of placing the reactor in the run mode following a shutdown. The differential pressure may be reduced to less than 1.17 psid 24 hours prior to a scheduled shutdown.

k. The differential pressure may be reduced to less than 1.17 psid for a maximum of four (4) hours for maintenance activities on the differential pressure control system and during required operability testing of the HPCI system, the relief valves, the RCIC system and the drywell-suppression chamber vacuum breakers.

l. If the specifications of Item i, above, cannot be met, and the differential pressure cannot be restored within the subsequent six (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition in twenty-four (24) hours.

m. Suppression chamber water level shall be maintained between -6 to -3 inches on torus level instrument which corresponds to a downcomer submergence of 3.00 and 3.25 feet respectively.

4.7 CONTAINMENT SYSTEMS (Cont'd)

e. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift when the differential pressure is required.

f. Suppression chamber water level shall be recorded at least once each shift when the differential pressure is required.

BASES:

3.7.A & 4.7.A Primary Containment

The integrity of the primary containment and operation of the core standby cooling system in combination limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time, thus greatly reducing the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth such that a rod drop would not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep off-site doses well below 10 CFR 100 limits.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1035 psig. Since all of the gases in the drywell are purged into the pressure supression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 45 psig which is below the maximum of 62 psig. Maximum water volume of 94,000 ft^3 results in a downcomer submergency of 4'-0" and the minimum volume of 84,000 ft^3 results in a submergence approximately 12-inches less. Mark I Containment Long Term Program Quarter Scale Test Facility (QSTF) testing at a downcomer submergency of 3.25 feet and 1.17 psi wetwell to drywell pressure differential shows a significant suppression chamber load reduction and Long Term Program analysis and modifications are based on the above submergence and ΔP.

Should it be necessary to drain the suppression chamber, provision will be made to maintain those requirements as described in Section 3.5.F BASES of this Technical Specification.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the pressure suppression pool is maintained below 160°F during any period of relief-valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high pressure suppression chamber loadings.

BASES:

3.7.A & 4.7.A Primary Containment (Cont'd)

The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least twice a week the oxygen concentration will be determined as added assurance. Mark I Containment Long Term Program testing showed that maintaining a drywell to wetwell pressure differential to keep the suppression chamber downcomer legs clear of water significantly reduced suppression chamber post LOAC hydrodynamic loads. A pressure of 1.17 psid is required to sufficiently clear the water legs of the downcomers without bubbling nitrogen into the suppression chamber at the 3.00 ft. downcomer submergence which corresponds to approx. 84,000 $ft.^3$ of water. Maximum downcomer submergence is 3.25 ft. at operating suppression chamber water level. The above pressure differential and submergence number will be used in the Pilgrim I Plant Unique Analysis to be submitted to the NRC.