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BECo Ltr. 94-071

U.S. Nuclear Regulatory Commission
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SUBJECT: **BOSTON EDISON COMPANY PEER REVIEW OF NRC PRELIMINARY ACCIDENT SEQUENCE PRECURSOR ANALYSIS OF A PILGRIM EVENT**

We have reviewed the preliminary Accident Sequence Precursor (ASP) program analysis of an operational event which occurred at Pilgrim on March 13, 1993. As requested in your April 13, 1994 letter, we reviewed for technical adequacy of the analyses including the depiction of plant equipment and equipment capabilities. Based on our review, we do not agree that this event is appropriately classified as a Loss of Offsite Power (LOOP) initiated reactor scram and as such, should not be considered as an accident sequence precursor event. The NRC model does not include key plant specific information regarding our offsite and onsite emergency power sources, nor does it credit additional capabilities for long term core cooling. This information is included in Attachment A.

We recommend this information also be used for other Pilgrim events that may be under consideration as preliminary accident sequence precursors. Should you have the need for additional questions on this material, please contact Mr. Clem Littleton (508) 830-7790 of our Systems and Safety Analysis Division.

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Attachment

JK/lam/RAP94/ASP

A001

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ATTACHMENT A

Plant Specific Information Not Included in NRC ASP Analysis

SUMMARY

- The scram was caused by a switchyard flashover, not a loss of offsite power. Preferred offsite power was available for 42 minutes following the scram.

- No credit was given for manual starting and loading the Emergency Diesel Generators (EDG) prior to losing the Startup Transformer (SUT).

- Offsite power remained available from the 23Kv source as automatic backup to the EDGs.

- Only Residual Heat Removal (RHR) is credited for providing long term core/ containment cooling whereas the main condenser could have been recovered, and in extreme cases, the direct torus vent is available for use.

DISCUSSION

4160V busses A-5 and A-6 are designated emergency service system busses. They power essential loads during abnormal operational transients and accidents, as well as during all normal operations.

The unit auxiliary transformer (UAT) powers these two vital busses during normal operations with the main generator on line. The start-up transformer (SUT) provides power when the main generator is not on line. During conditions when neither the UAT nor the SUT are available, the diesel generators will power the emergency busses. If the diesel generators fail to pick up the emergency busses within 10 seconds, the shutdown transformer (SDT) will automatically power these busses after an additional 2 second time delay. If the SDT fails to pick up the emergency busses, the blackout diesel may be started.

NUREG/CR-4674 Appendix A defines a BWR LOOP as corresponding to any situation in which power from both the auxiliary and startup transformers is lost. The NRC modeling assumption categorizes the Pilgrim March 13, 1993 event as a LOOP initiated reactor scram. The NRC bases this conclusion as being consistent with LOOPs described in NUREG-1032, "Evaluation of Station Blackout Accidents in Nuclear Power Plants". However, the March 13, 1993 Pilgrim reactor scram occurred as a result of load rejection caused by switchyard insulator flashovers due to wind packed snow deposited during a severe coastal storm. The actual loss of preferred offsite power did not occur until 42 minutes after the reactor scram event, during which time the startup transformer (SUT) remained available. Also, the EDGs were manually started and loaded 27 minutes after the scram in anticipation of losing the SUT. No credit for this action appears in the NRC analysis. Further, the design capability of the SDT to automatically backup the EDGs is not included. The SDT remained available throughout the event.

Based on these design details, we consider the NRC analysis classifying this event as a LOOP is incorrect and therefore not applicable as a preliminary ASP event.

COMMENTS ON CONDITIONAL CORE DAMAGE PROBABILITY CALCULATION

The NRC conditional core damage probability calculations are based on an event tree included in Appendix A of NUREG/CR-4674. The event tree identifies five risk significant sequences (identified by number) which lead to core damage. As described below, the sequences are not considered risk significant.

The ASP analysis is dominated by a LOOP initiated Station Blackout sequence, number 83.

> "Unable to recover long-term electric power following a loss of offsite power, failure of emergency power, and successful reactor scram."

This sequence is considered unlikely due to the LOOP not being the initiating event, the manual loading of the EDGs not credited and the automatic capability of the SDT to provide backup to the EDGs.

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The next most probable sequence number 40, deals with the loss of long term core cooling.

> "Unavailability of long-term core cooling (failure of residual heat removal in shutdown and suppression cooling modes) following a loss of offsite power with successful emergency power, reactor scram, safety relief vaive challenge and reseat, and successful high-pressure coolant injection."

Only the RHR system is credited in the analysis for this function. During the event, the SUT was recovered within 6 hours after the scram. The main condenser could have been recovered in the event of an unrecoverable RHR failure. Additionally, the analysis doesn't credit the capability of crossing the fire water system to RHR (FSAR Section 10.8) nor the direct torus vent for performing the long term core cooling function. The fire water system has no reliance on AC power and could be placed in service in a few hours. The direct torus vent could prevent violation of the primary containment pressurization limit if all other heat removal means fail. Thus, this sequence is considered unlikely.

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The next sequence, number 55, involves a stuck open safety relief valve (SRV) followed by loss of high pressure injection and failure to depressurize.

> "Unavailability of vessel depressurization following a loss of offsite power with successful emergency power and reactor scram. Safety relief valve challenge and unsuccessful reseat, and failure of the high-pressure coolant injection system."

If a SRV sticks open, depressurization would be accomplished in time to prevent core damage whether or not high pressure injection is successful. This was verified through use of the Modular Accident Analysis Program (MAAP) as referenced in our IPE report sent to the NRC on September 20, 1992. Therefore, this sequence is not a core damage sequence.

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The last two sequences of importance, numbers 67 and 69, deal with a loss of high pressure injection leading directly to core damage.

67 "Unavailability of long-term core cooling (failure of the residual heat removal system in shutdown and suppression pool cooling modes) following a loss of offsite power, failure of emergency power, successful reactor scram, successful long-term recovery of electric power, safety relief valve challenge and reseat, with failures of high-pressure coolant injection and reactor core isolation cooling."

69 "Failure of high-pressure coolant injection following a loss of offsite power, with emergency power failure, successful reactor scram, safety relief valve challenge and unsuccessful reseat."

Neither sequence challenges the depressurization function nor the subsequent low pressure injection function. Depressurization would have been initiated if high pressure injection had failed in this event. In fact, sequence 69 is not applicable for the same reason as sequence 55 in that depressurization would be successful for a stuck open relief valve. Therefore, these sequences do not represent the plant design at PNPS and are not applicable.