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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO INDIVIDUAL PLANT EXAMINATION

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

1.0 INTRODUCTION

On September 30, 1992, Boston Edison Company (BECo) submitted the Pilgrim Nuclear Power Station Individual Plant Examination (IPE) in response to Generic Letter (GL) 88-20 and associated supplements. On April 17, 1995, the staff requested additional information (RAI) from the licensee. The licensee responded in a letter dated December 28, 1995. In responding to the RAI, the licensee stated that they have updated their IPE (1995) and included in their response the updated accident sequences and dominant core damage contributors.

A "Step 1" review of the Pilgrim IPE submittal was performed and involved the efforts of Science & Engineering Associates, Inc., Scientech, Inc., and Concord Associates, in the front-end, back-end, and human reliability analysis (HRA), respectively. The Step 1 review focused on whether the licensee's method was capable of identifying vulnerabilities. Therefore, the review considered (1) the completeness of the information and (2) the reasonableness of the results given the Pilgrim design, operation, and history. A more detailed review, a "Step 2" review, was not performed for this IPE submittal. Details of the contractors' findings are in the attached technical evaluation reports (Appendices A, B, and C) of this staff evaluation report (SER).

In accordance with GL 88-20, Boston Edison proposed to resolve Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements." No other specific USIs or generic safety issues were proposed for resolution as part of the Pilgrim IPE.

2.0 EVALUATION

The Pilgrim Nuclear Power Station is a BWR 3 with Mark I containment. The licensee reported in its submittal (1992) a core damage frequency (CDF) of 5.8E-5/reactor-year. The contribution from internal flooding is 6E-8/reactor-year. Partial loss of offsite power (LOSP) contributed 48%, loss of feedwater 17%, total LOSP 13%, manual shutdown 8%, medium loss of coolant accident (LOCA) 4%, turbine trip 4%, and other initiators 6%. The CDF of the 1995 update is 2.8E-5/reactor-year with LOSP contributing 30%, manual shutdown 19%, full LOSP (345 & 23 kV) 10%, turbine trip and reactor trip 10%, loss of feedwater 8%, medium LOCA 6%, loss of condenser vacuum 4%, loss of direct current bus B 3%, mainstream isolation valve closure 3% and others 7%. The

biggest change in the percent contribution to the CDF is related to partial LOSP; the licensee explained in the response that this difference is primarily due to: elimination of the high pressure cooling injection (HPCI) system dependency on room cooling based on engineering evaluation; revision of the automatic depressurization system (ADS) success criteria (reduction of the number of safety relief valves required) based on additional analysis; and reduction of the failure rates of the HPCI and reactor core isolation cooling (RCIC) systems based on improved system performance.

The staff notes that the licensee, in order to identify potential initiating events, did a more thorough examination of plant design and history than typically seen in IPEs/PRAs. Thus, the Pilgrim IPE went beyond the typical initiators and modeled: manual scrams, shutdowns, partial-and-total loss of offsite power, vessel rupture, loss of coolant outside the containment, loss of instrument nitrogen, reactor water level reference line leak, loss of dc busses, loss of reactor building and turbine building cooling, and loss of instrument air. Also the licensee, to the extent possible, used plant-specific data to calculate component failure rates and unavailabilities.

Based on the licensee's IPE process used to search for decay heat removal (DHR) vulnerabilities, and review of the Pilgrim plant-specific features, the staff finds the licensee's DHR evaluation consistent with the intent of the USI A-45, Decay Heat Removal Reliability, resolution.

The licensee performed an HRA to document and quantify potential failures in human-system interactions and to quantify human-initiated recovery of failure events. The licensee performed a Fussel-Vesely importance calculation to identify the most important human actions. This analysis identified two actions as particularly important in the estimate of the CDF: operator fails to depressurize the reactor vessel; operator follows the loss of dc procedure when both dc buses are lost. Actions associated with standby liquid control, reactor water level control, fire water crosstie, and direct torus vent alignment were also identified as important based on the Fussel-Vesely importance.

The staff identified weaknesses in the licensee's HRA approach. Regarding pre-initiator event analysis, the licensee did not analyze human errors related to calibration of equipment. The licensee stated in their responses to staff's RAI that miscalibration error "is being considered for future revisions of the model." Although it is unlikely that the omission of calibration errors would critically impact the licensee's overall conclusions from the IPE, the licensee may have missed the opportunity to gain insights regarding contributors to plant safety.

Regarding post-initiator event analysis, the licensee used small (1E-3) screening human error probabilities (HEPs) to determine the most important human events. Typically, high values are used for screening analysis (e.g., 0.1 - 0.5). Also, it appears that the screening values were not modified to account for dependencies. Such a dependency could bring the HEP values for consecutive actions to 0.5 or even 1.0. The licensee stated in the responses

that the model quantification at a low truncation level (1.0E-9) and the use of "industry accepted ranges of human error rates provides high confidence that all important human actions/sequences were assessed." The staff believes that HEPs in the 1E-3 range are in the range of industry accepted values when a detailed HRA is performed and when dependencies are taken into consideration. The staff is unclear as to whether a truncation level at 1.0E-9 is low enough when values of 1E-3 are used for screening purposes. The use of low screening values can affect the results of the screening analysis in terms of important events and important sequences.

Furthermore, the licensee took 100% credit for inhibiting ADS under anticipated transient without scram (ATWS) and it appears that they took 100% credit for two human actions in the back-end analysis (initiating drywell sprays and initiating containment venting). For those human actions that the licensee performed a detailed quantification, it appears that in some instances the method used (NUREG/CR-1278) has been misapplied. Although, the post-initiator event analysis to some degree treated factors such as the need to diagnose an event, time available versus time needed for an action, plant specific factors, and the influence of the accident progression on human performance, it does not appear that these factors were evaluated via a systematic and comprehensive examination to ensure that important aspects of human performance under severe accidents were not missed.

The licensee, however, did model post-initiator human actions typically seen in IPEs/PRAs for BWRs 3&4 and the human error probabilities used for those events appear to be reasonable. Thus, the Pilgrim IPE results are in-line with similar BWRs 3&4 IPE results in both the most dominant initiators and the most dominant accident sequences. Therefore, the staff believes that, in spite of these weaknesses in the HRA, the Pilgrim IPE is capable of identifying vulnerabilities associated with the Pilgrim operation and practices. Therefore the staff concludes that the licensee's IPE meets the intent of GL 88-20. However, these weaknesses will limit the use of the Pilgrim IPE for regulatory purposes other than GL 88-20.

The licensee evaluated and quantified the results of the severe accident progression through the use of the relatively small containment phenomena event tree, supported by fault trees tailoring each question to the specific plant damage state being evaluated, and considered uncertainties in containment response through the use of sensitivity analyses.

The licensee's back-end analysis appeared to have considered important severe accident phenomena. Among the Pilgrim conditional containment failure probabilities, early containment failure is 22%, late containment failure is 61%, bypass is 0.4%. The containment remains intact 17% of the time. Early radiological releases are dominated by ATWS, LOSP and transients and late releases are dominated by LOSP and transients (primarily loss of feedwater and turbine trip). The licensee's response to Containment Performance Improvement Program recommendations is consistent with the intent of GL 88-20 and associated Supplement 3.

Some insights and unique plant safety features identified by the licensee at Pilgrim are:

1. Although Pilgrim has a higher than average loss (and non-recovery) of offsite power frequency because is located in a region with a higher frequency of severe weather occurrence, station blackout contributes only about 3% of the total CDF. This is because: (a) Pilgrim has a 14 hour battery capacity (with load shedding), (b) a third diesel for station blackout, (c) availability of fire water for vessel injection, (d) availability of separate 23 kV offsite power, and (e) credit for recovery of failed diesel.

2. Pilgrim has significantly lower conditional probability of early containment failure than other BWRs with Mark I containment (e.g. Peach Bottom, NUREG-1150), and a higher probability of late containment failure. It appears that the lower early containment failure is driven by the IPE's assumptions regarding early drywell failure due to overpressurization and liner meltthrough while the relative high late containment failure probability is driven by Pilgrim's containment flooding strategy. It is stated in the submittal that the licensee is considering alternatives to the current procedures for containment flooding and reactor pressure vessel venting, because the sensitivity analyses showed that the current procedure has a negative impact on containment performance.

The licensee defined vulnerabilities by examining whether the results suggested that there are "new or unusual means by which core damage or containment failures could occur," or that "the Pilgrim core damage frequency would not be able to meet the NRC's safety goal for core damage." The licensee concluded that neither of these criteria lead to the identification of vulnerabilities for Pilgrim.

It is stated in the submittal that BECo performed several improvements prior to the issuance of GL 88-20, as part of its Safety Enhancement Program. These changes included:

1. Installation of a hardened vent path,

2. Fire water cross-tie,

3. Implementation of revision 4 of the BWR Emergency Operating Procedures,

4. Installation of a third diesel,

5. Installation of a backup nitrogen supply system,

6. Modification of ADS initiation logic, and

7. Installation of an ADS inhibit switch.

The licensee derived several insights regarding potential improvements as a result of the IPE; improvements[1] that appear to have been implemented are:

1. Modification of loss of dc procedures to allow the operator judgment for load shedding of alternating current buses associated with dc supplies, and

2. Modification of procedures to allow operators to use fire water for drywell sprays.

3.0 CONCLUSION

Based on the above findings, the staff notes that: (1) the licensee's IPE is complete with regard to the information requested by GL 88-20 (and associated guidance in NUREG-1335), and (2) the IPE results are reasonable given the Pilgrim design, operation, and history. As a result, the staff concludes that the licensee's IPE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and therefore, that the Pilgrim IPE has met the intent of GL 88-20. The staff, however, identified weaknesses in the licensee's evaluation of human errors during severe accidents that will limit the use of the IPE for purposes other than GL 88-20.

It should be noted, that the staff's review primarily focused on the licensee's ability to examine Pilgrim for severe accident vulnerabilities. Although certain aspects of the IPE were explored in more detail than others, the review is not intended to validate the accuracy of the licensee's detailed findings (or quantification estimates) that stemmed from the examination. Therefore, this SER does not constitute NRC approval or endorsement of any IPE material for purposes other than those associated with meeting the intent of GL 88-20. However, because the licensee intends to continue to use and maintain its IPE, the staff encourages the licensee to improve the Pilgrim IPE in order to make it a more valuable tool for other regulatory applications.

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[1] The adequacy of the improvements was not part of the staff review.