UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 232

TO FACILITY OPERATING LICENSE NO. DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY

THE CONNECTICUT LIGHT AND POWER COMPANY

THE WESTERN MASSACHUSETTS ELECTRIC COMPANY

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

1.0 INTRODUCTION

By letter dated December 28, 1998, as supplemented March 1 and 29, 1999, the Northeast Nuclear Energy Company, et al. (NNECO, or the licensee), submitted a request for changes to the Millstone Nuclear Power Station, Unit 2, Technical Specification (TS) regarding revised loss of normal feedwater analyses. Specifically, the reactor trip setpoints for low steam generator level, in TS Table 2.2-1, would be revised. The setpoint change results in an earlier reactor trip on decreasing steam generator level. In addition to the TS changes, the licensee has also modified the loss of normal feedwater (LONF) transient analysis in both chapter 10 and chapter 14 of the plant Final Safety Analysis Report (FSAR). These modifications were needed to permit lower auxiliary feedwater flow which is now, in part, offset by the earlier reactor trip signal. The supplemental submittals provided additional information that did not change the staff's proposed no significant hazards consideration determination.

2.0 EVALUATION

TS Table 2.2-1

A change to TS Table 2.2-1 has been requested to raise the reactor trip setpoint for steam generator water level to 48.5%, with an allowable value of 47.5%. The licensee has justified these values by performing an analysis with the trip assumed to occur at 43%. The analysis assumes the trip setpoint is lower than the TS setpoint to account for the appropriate setpoint and instrumentation uncertainties. The change results in the reactor trip on reducing steam generator level initiating earlier than it does now. The reason for the change is to partially offset a reduction in auxiliary feedwater (AFW) flow. The licensee has performed the transient analysis with the modified setpoints with the AFW flow and concluded that the acceptance criteria continues to be met.

Although increasing the setpoint may increase the likelihood of a reactor trip on steam generator level, the licensee does not expect the setpoint to be approached during normal plant operation and has stated that an unexpected plant event would be needed to cause the setpoint to be reached. Additionally, the licensee has adjusted the pretrip alarm in the control room to provide the operators with the same advanced notice of a steam generator low level condition. The staff finds the proposed changes acceptable.

FSAR Chapter 10

The licensee has modified FSAR Chapter 10. The modifications include a reference to a new best estimate of LONF analysis. The licensee has stated that the revised analysis now credits the atmospheric dump valves (ADVs) in lieu of the main steam safety valves to remove heat from the generator. Crediting the ADVs results in increased flow to the steam generators because the ADVs can be opened at lower pressure and the AFW system delivers more water to the steam generators at reduced pressure. The staff has determined that crediting the ADVs for the FSAR Chapter 10 analysis is acceptable. With the credit for the ADVs the licensee has stated that the loss of feedwater design basis continues to be met. As a result, the staff finds the proposed changes to be acceptable.

FSAR Chapter 14

The licensee has performed a reanalysis of the FSAR Chapter 14 LONF transient analysis. The analysis was performed at the new setpoints and reduced AFW, and shows acceptable results. In addition to the changes to the flow and setpoints, the licensee has made a number of other changes to the transient analysis. The analysis shows that for the most limiting LONF cases analyzed, assuming a single failure, the steam generators do not empty, the pressurizer does not go water solid, and the steam generators do not exceed 110% of the design pressure. The licensee has stated that another decrease in heat removal from the secondary system event, the loss of electric load or turbine trip event, continues to be more limiting from both the standpoint of minimum departure from nucleate boiling ratio (DNBR) and from a peak reactor coolant system (RCS) standpoint. As a result, these aspects of the LONF event do not need to be evaluated.

In the performance of the new analysis, the licensee has used a different NRC-approved evaluation model. The methodology is contained in the report ANF-89-151(P)(A) ANF-RELAP METHODOLOGY FOR PRESSURIZED WATER REACTORS: ANALYSIS OF NON-LOCA CHAPTER 15 EVENTS, and was approved by the staff in March of 1992. The methodology is appropriate for evaluating the LONF event. The licensee has analyzed five different cases to determine the most limiting conditions. The cases analyzed were chosen to maximize pressurizer water level and minimum steam generator water level and considered different combinations of the limiting single failures and the availability of offsite power. The limiting single failure was either a motor driven or turbine driven AFW pump. The analysis now credits automatic initiation of the turbine driven pump and conservatively assumes a minimum total AFW flow that includes instrument uncertainties and a 5 percent pump degradation. The initial conditions were also biased to maximize pressurizer water level and minimize steam generator water level. The initial steam generator and reactivity feedback values were conservatively selected in accordance with the approved topical report. The licensee also considered both the availability and unavailability of the normal plant controls and offsite power to be assured the limiting event was considered.

The licensee has also changed the way the main steam safety relief valve accumulation is modeled in the new analysis. By letter dated March 29, 1999, the licensee stated that rather than assuming the valve opens at the nominal setpoint plus 3 percent to account for drift, plus another 3 percent to account for accumulation, the licensee has modeled the valves to open at the nominal setpoint plus 3 percent to account for drift with a 0.1-second delay to account for valve accumulation. The licensee has justified this assumption by referencing a statement by the valve vendor which stated that the valve will go full open in about 20 to 30 milliseconds. Because the 0.1 second time assumed in the analysis is much higher than 20 to 30 milliseconds, the staff finds the modified assumption to be acceptable.

The analysis results for all cases show that the pressurizer does not go water solid, the steam generators do not empty, and decay heat is removed without exceeding 110 percent of the main design pressure. Because the licensee has demonstrated acceptable results for the LONF event by performing a reanalysis with an NRC-approved evaluation model that considers the limiting single failure and conservatively modeled the plant and the initial conditions, the staff finds the changes to be acceptable.

Instrument Setpoint Methodology

In a letter dated December 28, 1998, the licensee stated that the instrument setpoint methodology used to calculate trip setpoints and allowable values is consistent with the approach used in TS changes previously submitted by letters dated July 21 and October 6, 1998. The staff has previously reviewed these TS changes and has determined that the licensee's setpoint methodology is consistent with the guidance of Regulatory Guide 1.105, Rev. 2 and ISA Standard 67.04, 1982 and, is therefore, acceptable. The staff finds that the licensee's methodology for the proposed changes is acceptable. However, the licensee did not account for the effects of harsh environment on instrument drift because this instrumentation is not subjected to harsh environment. Feedwater system pipebreaks inside and outside containment are not included in the licensing basis for this plant (FSAR Section 14.2.8). In a conference call with the licensee, the staff requested the licensee to confirm that reactor trip on low SG water level has not been credited in any other event resulting in a harsh environment. In a letter dated March 1, 1999, the licensee documented that this instrumentation will not be required to trip the reactor when subjected to a harsh environment. Based on this documentation, the staff finds the FSAR and TS changes related to reactor trip on low SG water level to be acceptable.

The licensee has also revised the FSAR and TS Bases Section 2.2.1 for thermal margin/low pressure reactor trip setpoint from 1850 psia to 1865 psia to account for the uncertainties caused by the harsh environment. The staff finds the proposed change to be acceptable because it properly accounts for the uncertainties caused by the harsh environment.

Credit for Auto Initiation of Auxiliary Feedwater Start Signal

The revised analysis of LONF, documented in the licensee's submittal dated December 28, 1998, takes credit for the automatic initiation of Motor Driven Auxiliary Feedwater Pumps, within 4 minutes, after steam generator water level reaches the automatic auxiliary feedwater actuation setpoint. FSAR Section 7.3, "Engineered Safety Features Actuation System", which includes the Auxiliary Feedwater Automatic Initiation System (AFAIS), states that this system is designed to meet the requirements of Institute of Electrical and Electronics Engineers (IEEE)

Standard 279-1971, "Criteria for Nuclear Generating Station Protection Systems." Therefore, since this system meets the requirements of IEEE-279, the licensee can take credit for the system in the LONF event.

The licensee included conforming Bases pages with the amendment request. The NRC does not review and approve the Bases but they are included to maintain a current Authority File.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Connecticut State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (64 FR 6701, February 10, 1999). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: C. Jackson
H. Garg
V. Ordaz

Date: April 8, 1999