NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
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October 11, 1985

Docket No. 50-245
B11795

Director of Nuclear Reactor Regulation
Attn: Mr. Christopher I. Grimes, Chief
Systematic Evaluation Program Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

References: (1) J. F. Opeka letter to C. I. Grimes, dated May 17, 1985.

(2) H. L. Thompson letter to J. F. Opeka, dated July 31, 1985.

Gentlemen:

Millstone Nuclear Power Station, Unit No. 1
Integrated Safety Assessment Program

In Reference (1), Northeast Nuclear Energy Company (NNECO) provided a proposed scope for the Integrated Safety Assessment Program (ISAP) review of Millstone Unit No. 1. In Reference (2), the Staff formally issued the results of the ISAP screening review process, establishing the scope of ISAP for Millstone Unit No. 1 and initiating issue-specific evaluations. Reference (1) also indicated that for each issue or topic included in ISAP, NNECO would provide a discussion of the safety objective and an evaluation of the plant design with respect to the issue being addressed to identify specific items to be considered in the integrated assessment. In accordance with this commitment, reviews for the following ISAP topics are attached:

- ISAP Topic 1.06 - "Seismic Qualification of Safety-Related Piping"
- ISAP Topic 1.13 - "BWR Vessel Water Level Instrumentation"

If you have any questions concerning the attached reviews, please contact us.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

J. F. Opeka

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By: W. F. Fee
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ISAP TOPIC NO. 1.06

SEISMIC QUALIFICATION OF SAFETY-RELATED PIPING

ISAP Topic No. 1.06

Seismic Qualification of Safety-Related Piping

I. Introduction

I&E Bulletin 79-14 required licensees to verify that seismic analyses performed on safety-related piping systems apply to the actual configuration of safety-related piping systems. NRC I&E Bulletin 79-02 required licensees to verify the adequacy of pipe support base plates and expansion anchor bolts utilized in supporting safety-related piping.

II. Review Criteria

(1) I&E Bulletin 79-02
(2) I&E Bulletin 79-14
(3) Regulatory Guide 1.29
(4) Millstone Unit No. 1 FSAR

III. Related Topics/Interfaces

ISAP Topic No. 1.17

IV. Evaluation

In response to I&E Bulletins 79-02, Pipe Support Designs Using Concrete Expansion Anchor Bolts, and 79-14, Seismic Analysis for As-Built Safety-Related Piping Systems, NNECO initiated an extensive reanalysis and modification effort to qualify safety-related piping for safe shutdown earthquake loadings. The scope of this effort involves the following systems:

- Service Water
- RBCCW
- Isolation Condenser
- Stand-By Liquid Control
- Secondary Cooling Water
- Core Spray
- Containment Cooling
- Shutdown Cooling
- Condensate
- Feedwater
- Reactor Water Cleanup
- Standby Gas Treatment
- Control Rod Drive
- Reactor Head Cooling
- Atmospheric Control
- Main Steam
- Demineralized Water and Condensate Storage and Transfer
- Primary Containment Cooling Service Water
- Diesel Generator Fuel Oil
- Reactor Vessel Vent and Drain

For Millstone Unit 1, all piping covered by I&E Bulletin 79-14 has been field verified. The differences found between as-built and the design assumed in the analysis were analyzed utilizing FSAR criteria for acceptable stress levels. Because of the low acceptable stress level, the FSAR method provided conservative results and showed that several of the piping supports needed modifications. These modifications were grouped in two categories, as discussed below:

- Priority Modifications

  Those modifications which were needed to qualify the piping for the Operating Basis Earthquake or OBE (ground acceleration $\leq$ 0.07g) were classified as priority modifications. All these modifications have been completed.

- Upgrading Modifications

  Those modifications which are needed to qualify the piping for the Safe Shutdown Earthquake or SSE (ground acceleration $\leq$ 0.17g) were classified as the upgrading modification. Many of these modifications have been completed.

Approximately 1,100 modifications have been identified, of which approximately 370 remain. NNECO currently plans to implement 100 pipe-support modifications during the 1985 outage. The remaining modifications are scheduled for the 1987 and 1989 outages.

In a letter dated August 23, 1985 (Reference 4), NNECO documented the results of our public safety risk-oriented analysis of the remaining modifications. The analysis concluded that no overall significant reduction in public safety would be gained through implementation of the remaining modifications. Reference 4 provides a detailed discussion of the risk-oriented analysis of the remaining modifications to safety-related piping.

The current work scope of the upcoming outage reflects the results of our public safety risk-oriented analyses. Although the overall benefit in implementing the remaining modifications was very low (0.1 out 10 - Reference 4), the high priority modifications identified by the PRA review have been incorporated into the 1985 outage work scope. At the completion of the 1985 outage it is anticipated that the piping modifications associated with the following systems will be completed:

- Isolation Condenser Supply and Return
- Service Water (Note 1)
- Containment Cooling (LPCI)
- Core Spray
- Condensate (Note 2)
- Feedwater (Note 2)
- Diesel Fuel Oil
- Secondary Cooling (Note 2)
- Containment Cooling Service Water
- Recirculation
- Atmospheric Control (Note 2)

Note 1: Service water will be complete with the exception of the small bore bearing cooling lines for the service water pumps. These modifications will be available for post-outage, accessible area construction.

Note 2: System completion is dependent upon the quantity of work accomplished in this 1985 refueling as the current outage schedule calls for a very short (approximately 33 days) outage.

V. Conclusions

Based on the above, it is apparent that NNECO has taken the initiative in assuring quality in our seismic assessments of Millstone Unit No. 1 and has implemented modifications to piping when it has been determined that such modifications would lead to an increase in safety. This has been acknowledged by the NRC as outlined in a recent inspection report (Reference 3).

"Based on the above observations, the inspector concludes that licensee has demonstrated satisfactory management involvement and approach to resolution of technical issues from a safety standpoint."

With respect to the remaining modifications, NNECO's position is since analysis has shown that the remaining modifications will not provide any significant improvement to safety, a priority for implementation and an evaluation of the associated resource burden should be evaluated in the integrated assessment.

VI. References

1. I&E Bulletin 79-02, Pipe Support Plate Designs Under Concrete Expansion Anchor Bolts.

2. I&E Bulletin 79-14, Seismic Analyses for As-Built Safety-Related Piping Systems.

3. I&E Inspection Report 50/245/85-04, dated April 18, 1985.

4. J. F. Opeka letter to C. I. Grimes, "Integrated Safety Assessment Program Summaries of Public Safety Impact Model Project Analyses," dated August 23, 1985.

5. SEP Topic III-6, Seismic Design Considerations

ISAP TOPIC NO. 1.13

BWR VESSEL WATER LEVEL INSTRUMENTATION

ISAP Topic No. 1.13
BWR Vessel Water Level Instrumentation

I. Introduction

The water level instrumentation in a BWR is relied upon for controlling feedwater, actuating emergency systems and for providing the operators information which is used as a basis for actions to assure adequate core cooling. Many of the actions in the emergency procedures guidelines are keyed to reactor water level.

NUREG-0737 Item II.F.2 and Generic Letter 84-23 require that licensees modify or supplement existing equipment to assure accurate indication of vessel water level.

II. Review Criteria

NUREG-0737, Item II.F.2
Generic Letter 84-23

III. Related Topics/Interfaces

ISAP Topic No. 1.09

IV. Evaluation

The reactor pressure vessel (RPV) water level indication which is used to mitigate accidents is based on a number of level gauges which utilize one of two reference legs located within the drywell. Unusually high drywell temperature in conjunction with RPV depressurization can affect reactor water level instrumentation by causing water in the reference legs to flash when the drywell temperature reaches RPV saturation condition. This results in erroneous readings that indicate a higher reactor water level than the actual level which is present in the vessel. If the operator places undue reliance on the indicated water level the potential exists that makeup could be throttled to the extent that the core is uncovered. Such scenarios are explicitly considered in the Millstone Unit 1 emergency procedures. (See EOP 508, Part 3.4, in Appendix 2-B of the Probabilistic Safety Study (PSS).)

The outcome of reference leg flashing is explicitly addressed in the Millstone Unit 1 PSS in the treatment of cognitive errors involving failure to control water level. Of particular interest are the LOCA scenarios which cause a release of high temperature steam to the drywell at the same time that the RPV is depressurizing and thus provide the conditions for reference leg flashing.

In non-LOCA transients, prolonged loss of drywell cooling also heats up the reference legs. However, in these scenarios, the reference leg temperature is maintained below RPV saturation condition and therefore no flashing occurs. At a higher reference leg temperature, the indicated water level is slightly higher than the actual water level due to the

decrease in water density in the reference leg. The difference between indicated and actual water level is only a few inches and is therefore negligible compared to the margin that exists before core uncovery takes place (the normal water level is about 15 feet above the top of the core). Consequently, operator decisions based on indicated level do not jeopardize the core cooling.

V. Conclusions

Reference (1) reported the results of NRC Staff review of a BWR Owners Group Report, Reference (2), which generically evaluated reactor vessel water-level instrumentation. Reference (2) concluded, in part, that some inaccuracies can be introduced through heating of level instrument reference legs in the drywell as a result of post-LOCA environmental conditions. Allowance was made for these inaccuracies in the BWR Owners Group's development of emergency procedure guidelines. Reference (2) recommended physical improvements to further reduce the effect of post-LOCA environment in containment.

Reference (1) reported an NRC conclusion that the changes identified in the emergency procedure guidelines are adequate for short term. In the longer term, however, NRC concluded that permanent physical improvements, along the lines recommended in Reference (2) should be made "to reduce the burden on the operator." Reference (1) requested BWR licenses to submit a description of plans to implement these improvements and a proposed schedule.

NNECO is currently investigating several proposed consultant approaches to resolve the reference leg flashing issue for Millstone Unit No. 1 and expects to initiate detailed investigations in the next few months. The need for additional instrumentation should be evaluated in the integrated assessment.

VI. References

1. Generic Letter 84-23, "Reactor Vessel Water Level Instrumentation in BWRs."

2. S. Levy Inc. Report SLI-8211 "Review of BWR Reactor Vessel Water Level Measurement System."

3. J. F. Opeka letter to C. I. Grimes, "Integrated Safety Assessment Program Summaries of Public Safety Impact Model Project Analyses", dated September 6, 1985.