NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

General Offices • Selden Street Berlin, Connecticut

P.O. BOX 270
HARTFORD, CONNECTICUT 06141-0270
(203) 665-5000

September 13, 1985

Docket No. 50-245
B11695

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, the review for the following ISAP topic is attached:

o ISAP Topic 1.29 - "Response to Generic Letter 81-34"

If you have any questions concerning the attached reviews, please do not hesitate to contact us.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

J. F. Opeka

J. F. Opeka
Senior Vice President

cc: J. A. Zwolinski

A001
1/1

ISAP TOPIC NO. 1.29

RESPONSE TO GENERIC LETTER 81-34

I. Introduction

As a result of the Browns Ferry Unit 3 control rod partial insertion failure on June 28, 1980, the NRC initiated an investigation of safety concerns associated with postulated pipe breaks in the boiling water reactor (BWR) scram system. On April 3, 1981, the NRC published draft NUREG-0785, "Safety Concerns Associated with Pipe Breaks in the BWR Scram System." Subsequently, on April 10, 1981, the NRC issued a letter to all BWR licensees requiring generic and plant-specific evaluations of the safety concerns discussed in draft NUREG-0785.

On July 7, 1981, the NRC informed all BWR licensees that the generic review of this issue had been completed and that a NUREG describing the results of the review would be issued. Generic Letter 81-34, "Safety Concerns Associated with Pipe Breaks in the BWR Scram System," was issued to all BWR licensees on August 31, 1981 and transmitted NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping."

Briefly, the NRC concluded that the generic BWR scram discharge volume (SDV) design was acceptable, provided certain requirements that were met on a plant-specific basis. The NRC also concluded that safety concerns associated with a postulated scram discharge system piping failure do not represent a dominant contribution to core melt risk, provided certain generic risk assessment assumptions were validated on a plant-specific basis. NUREG-0803 provided the NRC's guidance regarding plant-specific response.

Specifically, BWR licensees were requested to provide information related to:

1) piping integrity;

2) capability to mitigate a scram discharge volume break and

3) environmental qualification of equipment required to mitigate a scram discharge volume break.

The objective of ISAP Topic 1.29 is to evaluate the Millstone Unit No. 1 SDV design against the guidelines in NUREG-0803.

II. Review Criteria

1. 10CFR50.2(v) and 50.46

2. 10CFR50, Appendix A, General Design Criteria 35 and 55

3. Generic Letter 81-34, dated August 31, 1981

4. NUREG-0803, dated August 1981

III. Evaluation

A. Millstone Unit No. 1 compliance with NUREG-0803

Reference (4) forwarded NUREG-0803 (Reference (5)) which concluded that SDV piping design was acceptable provided certain conditions were satisfied and information was verified on a plant-specific basis. Licensees were requested to provide information related to piping integrity, capability to mitigate an SDV break and environmental qualification of equipment required to mitigate an SDV break.

The BWR Owners Group requested General Electric (GE) to perform analyses relating to the possibility of an SDV pipe break. Initial results were reported in NEDO-22209, "Analysis of Scram Discharge Volume System Piping Integrity" submitted by Reference (6). In response to NRC questions in Reference (7), additional analyses were performed with results reported by Reference (8) and (10). This BWR Owners Group effort concluded, using both probabilistic and mechanistic analyses, that the risk associated with SDV pipe breaks is very small. Northeast Nuclear Energy Company (NNECO) participated in these BWR Owners Group efforts and supports their conclusion that the probability of core damage initiated by an SDV break is sufficiently small as to preclude the necessity of qualifying equipment specifically to detect and/or mitigate the consequences of such a break.

Concurrently, and for different reasons, major modifications were made to the Millstone Unit No. 1 SDV. During the 1982 refueling outage, the system was modified including replacement of all piping downstream of the hydraulic control unit header. The previous single SDV was replaced with two larger volumes, and the hydraulic coupling between the SDV and the header was improved by elimination of all two inch diameter piping. To the extent practical, the new SDV system piping (except that downstream of the vent and drain valves) meets the design, fabrication, installation, testing, and quality assurance (QA) requirements of ASME Section III, Class 2, 1977 Edition through Summer, 1979 Addenda (excluding an N-stamp). The piping was installed in full conformance with the Northeast Utilities QA program in effect in late 1982. All threaded joints were eliminated and the piping and supports were designed to seismic Category I criteria.

NNECO has reviewed the summary description of BWR response to an SDV break contained in NUREG-0803. This description does not apply to Millstone Unit No. 1, since the plant does not incorporate a High Pressure Coolant Injection (HPCI) or Reactor Core Isolation Cooling (RCIC) system. These systems, and their capability to operate in the environment resulting from an SDV break were a principal concern in NUREG-0803. At Millstone Unit No. 1, high pressure makeup is provided by the Feedwater Coolant Injection (FWCI) system, which is located in the turbine building. This system uses one of the safety-related feedwater (FW) trains and can be powered from the gas turbine generator on loss of offsite power. A detailed description of plant response to an SDV break and a discussion of related equipment qualification issues is contained the following section.

In response to NRC concerns, the BWR Owners Group proposed that plants classifying SDV piping as ASME Section III, Class 2 perform a post-scram walkdown once each cycle to inspect for evidence of leakage. NNECO considers such a walkdown unnecessary at Millstone Unit No. 1. As noted above, essentially all system piping was replaced in 1982 utilizing full QA controls. For this reason, NNECO considers that pressurization of the system and inspection for leakage once during each in-service inspection period (approximately every 3 1/3 years) is adequate. NNECO does not intend to implement this BWR Owners Group recommendation.

Finally, NUREG-0803 requested that BWR licensees submit Technical Specification changes adopting primary coolant iodine limits of the Standard Technical Specifications. This change was proposed by Reference (9).

B. Millstone Unit No. 1 Response to a Postulated SDV Pipe Break

The design of Millstone Unit No. 1 does not incorporate a RCIC or HPCI system. The functions that these systems perform at other BWRs are performed by the Isolation Condenser (IC) and the FWCI system respectively. Plant response to a postulated pipe break in the scram discharge volume would thus differ from the scenario described in NUREG-0803.

Operator response to a scram at Millstone Unit No. 1 would include taking the mode switch out of RUN to prevent closure of the main steam isolation valves (MSIV) and allowing piping system pressure to be controlled by bypassing steam to the main condenser. This is the same initial action described in section 4.1.3 of NUREG-0803. Regardless of whether the MSIVs are open, however, water level would be maintained using the control rod drive (CRD) pumps, if running, and the FW system. The only significant indication of a leak would likely be high radiation alarms, although use of symptom based operating procedures would not require the operator to identify the leak location to take proper action.

Scenarios for actions following an SDV break can be divided into 3 types: 1) Normal Power; 2) Loss of Offsite Power; 3) Loss of FW/FWCI. Each scenario is described separately below.

Normal Power

Following a postulated SDV break with normal power, reactor vessel water level would be maintained with the FW system through either the low flow regulating valve or the main FW regulating valves. Pressure control would be maintained by bypassing steam to the main condenser if the MSIVs remain open, or by use of the Isolation Condenser (preferred) or safety relief valve actuation, if the MSIVs are closed.

Short-term decay heat removal would be accomplished by the IC. Long-term core cooling would be accomplished using the Low Pressure Coolant Injection System (LPCI) and the containment cooling heat exchangers.

Loss of Offsite Power

Following a postulated SDV break with a coincident loss of offsite power, reactor vessel water level would be maintained initially using FWCI (powered by the gas turbine). The emergency condensate transfer pump is located in the Reactor Building and is not qualified for a 212°F/100% humidity environment (see below), and makeup to the condenser hotwell therefore may not be available. However, the normal condensate transfer pumps, which are located outside of the reactor building and thus would not be exposed to the harsh environment, could be used. If no makeup pumps were available, FWCI would continue to deliver water to the reactor vessel until the pumps trip on low suction pressure.

If FWCI flow lasts long enough and the IC has reduced pressure sufficiently, LPCI and/or Core Spray can then be used to maintain water level following loss of FWCI. If the reactor pressure remains too high for use of these systems, manual depressurization would occur. Use of the symptom-based Emergency Operating Procedures assures that this manual depressurization would be performed by the operator if needed.

As in the case for normal power, short term decay heat removal would be via the IC with long-term core cooling accomplished using the LPCI system and the containment spray cooling heat exchangers.

Loss of FW/FWCI

The loss of FW/FWCI case would be similar to the loss of offsite power case after FWCI trip. Manual depressurization would definitely be required and would occur sooner. Following depressurization, LPCI and/or core spray can be used to supply water to the core.

Decay heat removal and long-term cooling would be identical to the other two cases.

Environmental Qualification of Equipment Used

Following the worst possible SDV pipe break, the environment in the Reactor Building could reach saturated conditions (212°F/100% humidity). Analyses performed by GE for the BWR Owners Group and previously provided to NRC have demonstrated the probability of such a break to be extremely low. Leakage from a small crack, a more likely failure mechanism, would not have any appreciable effect on the Reactor Building environment.

The following list describes the qualification of potential mitigating systems in the highly unlikely event of a major SDV failure (212°F/100% humidity/100 rads integrated exposure).

1. Isolation Condenser: Required components of this system are qualified for both the normal power and loss of offsite power cases.

2. FW System: No portion of this system (other than piping) is located in the Reactor Building except the emergency condensate transfer pump. This pump is not qualified for the possible environment; however, the normal condensate transfer pumps, which are not in the Reactor Building and thus would not be exposed to the harsh environment, could be used. If these pumps fail FWCI can deliver water to the reactor until the supply in the condenser hotwell is exhausted. Manual depressurization and other qualified systems would then be used as noted above.

3. Automatic Depressurization System/Safety Relief Valves: Required components of this system are qualified for the normal and loss of offsite power cases.

4. LPCI System: Required components are qualified for both the normal and loss of offsite power cases.

5. Low Pressure Core Spray System: Required components are qualified for both the normal and loss of offsite power cases.

6. CRD pumps: These pumps are not qualified for this environment. If they continue to operate they provide additional makeup/cooling but are not needed for mitigation.

7. Shutdown Cooling: Required components are not qualified. Long term cooling can be provided using LPCI.

Summary

For Millstone Unit No. 1, an SDV break accident is not a major core uncovery or risk concern. FW will maintain reactor vessel level in the event of loss of the FW system, manual depressurization and other qualified systems would keep the core covered and provide core cooling.

IV. Conclusions

Based on the information provided above, the concerns enumerated in NUREG-0803 have been completely addressed for Millstone Unit No. 1. Accordingly, further analyses or SDV modifications are unnecessary.

V. References

1. NUREG-0785, dated April 1981.
2. Generic Letter 81- 10 dated April 10, 1981.
3. NEDO-24342, dated April 1981.
4. Generic Letter 81-34, dated August 31, 1981.
5. NUREG-0803, dated August 1981.
6. NEDO-22209, dated October 1982.
7. D. G. Eisenhut letter to BWR Owners Group, dated July 25, 1983.
8. BWR Owners Group letter to D. G. Eisenhut, dated November 18, 1983.

9. W. G. Counsil letter to D. M. Crutchfield, dated December 28, 1983.

10. BWR Owners Group letter to D. G. Eisenhut, dated April 1984.