Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Rope Ferry Road
Waterford, CT 06385

Dominion

JUL 11 2001

Docket Nos. 50-336
50-423
B18429

RE: 10 CFR 50.54(f)

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Millstone Nuclear Power Station, Unit Nos. 2 and 3
Response to a Request for Additional Information Regarding
Resolution of Issues Related to Generic Letter 96-06

This letter provides Dominion Nuclear Connecticut, Inc. (DNC) response to a request for additional information regarding the resolution of issues related to Generic Letter (GL) 96-06[1] for Millstone Unit No. 2. GL 96-06 included a request for Licensee to evaluate cooling water systems that service Containment air coolers to assure that they are not vulnerable to water hammer and two-phase flow conditions.

Millstone Unit No. 2 responses to the requested actions of GL 96-06[1] and a previous request for additional information[2] were provided in letters dated January 28, 1997,[3] and January 12, 1999.[4]

---

(1) Thomas T. Martin letter to Regulatory Affairs, "Generic Letter 96-06, Assurance of Equipment Operability and Containment Integrity During Design - Basis Accident Conditions," dated September 30, 1996.

(2) D. G. McDonald letter to M. L. Bowling, "Request for Additional Information Regarding Resolution of Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design - Basis Accident Conditions," dated September 30, 1996," dated May 5, 1998.

(3) M. L. Bowling letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 2, Response to Requested Actions of Generic Letter 96-06, Assurance of Equipment Operability and Containment Integrity During Design - Basis Accident Conditions," dated January 28, 1997.

(4) M. L. Bowling letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 2, Responses to the Request for Additional Information Regarding Resolution of Issue Related to Generic Letter 96-06 (TAC No. M96833)," dated January 12, 1999.

A072

On October 6, 2000,[5] a request for additional information was received via fax containing five (5) questions. Four (4) questions are related to Millstone Unit No. 2 and one (1) question is related to Millstone Unit No. 3. The request was followed by several teleconferences with the Nuclear Regulatory Commission (NRC) staff.

The purpose of this letter is to transmit the response to the Millstone Unit No. 2 questions (Attachment 1). A response to the Millstone Unit No. 3 question was submitted on December 11, 2000.[6]

In a conference call on May 23, 2001, the NRC indicated that they have completed their review of the Millstone Unit No. 3 response[6] and are satisfied with the evaluation of all susceptible locations, except for penetration Z-56 and 3FPW*CTV49, and commented that a long term action is required to insure continued protection against overpressurization. DNC has written a Condition Report (CR-01-05586) on May 24, 2001, to capture the NRC comment and determine further corrective action.

There are no regulatory commitments contained within this letter.

If you should have any questions on the above, please contact Mr. Ravi Joshi at (860) 440-2080.

Very truly yours,

DOMINION NUCLEAR CONNECTICUT, INC.

J. Alan Price, Vice President
Nuclear Technical Services - Millstone

Sworn to and subscribed before me

this 11 day of July, 2001

Donna Lynne Williams
Notary Public

My Commission expires Nov 30, 2001

cc: See next page

[5] Fax from Robert Pulsifier to Paul Russell, "Draft RAI to Generic Letter 96-06," dated October 6, 2000, (A15513).

[6] R. P. Necci letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 3, Response to a Request for Additional Information Regarding Resolution of Issues Related to Generic Letter 96-06," dated December 11, 2000.

Attachment (1)

Enclosure (1)

cc: H. J. Miller, Region I Administrator
J. T. Harrison, NRC Project Manager, Millstone Unit No. 2
S. R. Jones, Senior Resident Inspector, Millstone Unit No. 2
V. Nerses, NRC Senior Project Manager, Millstone Unit No. 3
A. C. Cerne, Senior Resident Inspector, Millstone Unit No. 3

Docket Nos. 50-336
50-423
B18429

Attachment 1

Millstone Nuclear Power Station, Unit Nos. 2 and 3

Millstone Nuclear Power Station, Unit No. 2
Response to a Request for Additional Information Regarding
Resolution of Issues Related to Generic Letter 96-06

Millstone Nuclear Power Station, Unit No. 2
Response to a Request for Additional Information Regarding
Resolution of Issues Related to Generic Letter 96-06

Question 1 *In the 1-28-97 submittal, the licensee states that 9 penetrations are to be reviewed and corrective actions implemented prior to startup. Provide a description of how the thermal pressurization concern has been addressed for each of these penetrations.*

Response: An evaluation was performed for the potential for thermal overpressurization in isolated piping inside Containment for Millstone Unit No. 2 in the event of a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB). The evaluation concluded that nine (9) out of the 89 penetrations (penetrations 67 and 68 were evaluated as one penetration) reviewed are susceptible to thermal overpressurization.

In addition to the Containment penetrations, the evaluation addressed isolated piping segments inside Containment. All modifications and procedure changes to eliminate thermal overpressurization are summarized in Table 1 below. Based on the results of the evaluations, and the completed actions summarized in the table below, it is concluded that the integrity of the safety-related systems are maintained and their ability to perform their intended safety functions will not be adversely impacted due to thermal overpressurization.

Table 1
Evaluation of Potential Thermal Overpressurization of Isolated Piping Inside Containment

Penetration	System	Description	Resolution	Modification/ Procedure Change
2	Chemical & Volume Control (CVC)	Let-Down line to Purification Demineralizer Piping between 2-CH-516 and 2-CH-089	These valves trap hot fluid (550° F) during Containment isolation and eliminate the potential for overpressure.	None
	CVC	Let-Down line to Purification Demineralizer Piping between 2-CH-515 and 2-CH-516	This line is significantly cooled by the charging flow during LOCA and MSLB transients eliminating the thermal overpressure conditions.	None
10	Shut-Down Cooling (SDC)	SDC suction piping between 2-SI-651 and 2-SI-709	Addition of thermal insulation will prevent the fluid temperature from rising above 200° F. Procedure OP 2201, "Plant Heatup" dictates closing these valves only when the SDC temperature is greater than 200° F and less than 240° F.	Complete

Penetration	System	Description	Resolution	Modification/ Procedure Change
14	Clean Liquid Radwaste (CLR)	Containment Sump to Aerated Drain Tank piping between 2-SSP-16.1 and 2-SSP-16.2	These are globe valves with flow under the seat. Lifting of outboard isolation valve 2-SSP-16.2 relieves the pressure build-up.	None
	CLR	Containment Sump to Aerated Drain tank piping between 2-SSP-115A/B and 2-SSP-16.1	Although 2-SSP-16.1 which is a globe valve with flow under the seat, a relief valve was installed in this line per Design Change Request M2-97026.	Complete
21	Reactor Coolant System (RCS)	RCS sample branch lines between 1. 2-RC-045 and 2-RC-003, 2. 2-RC-045 and 2-RC-002, 3. 2-RC-045 and 2-RC-001, and 4. 2-RC-045 to 2-LRR-61.1.	2-RC-045 and 2-RC-003 are normally open at all times and trap hot fluid (550° F) during Containment isolation and eliminate the potential for overpressure. The other branch lines which are connected between 2-RC-045 and 2-RC-003 relieve the pressure buildup to the already cooled, trapped hot fluid line.	None
35	CLR	Discharge piping from the Primary Drain Tank between Containment isolation valves 2-LRR-43.1 and 2-LRR-43.2	These valves are globe valves with flow under the seat. Thus, overpressurization would relieve through the outboard valve 2-LRR-43.2. 2-LRR-69 is normally open and provides a relief path downstream of 2-LRR-43.2.	2-LRR-69 is normally open during plant operation.
43	Charging System (CHS)	Control bleed-off line between 2-CH-506 (inside Containment) and 2-CH-198 (outside Containment)	Only a very small portion of the piping is inside Containment (2.625 ft) as compared to the total length of the isolated piping (31 ft). Pipe stresses due to thermal overpressurization are within the code allowable limits.	None
	CHS	Control bleed-off line between 2-CH-506 (inside Containment) and 2-CH-505 (outside Containment)	Only a very small portion of the piping is inside Containment (2.625 ft) as compared to the total length of the isolated piping (31 ft). Pipe stresses due to thermal overpressurization are within the code allowable limits.	None
49	Fire Protection Water	Procedure OP-2341A calls for closing the inside Containment valve (2-FIRE-120) and outside Containment Isolation valve (2-FIRE-108), thus isolating a portion of the water filled fire protection piping.	A procedure change has been implemented to leave 2-FIRE-120 open since this is not a Containment isolation valve, thus eliminating thermal overpressurization concerns.	Complete

Penetration	System	Description	Resolution	Modification/ Procedure Change
67 & 68	Spent Fuel Pool Cooling	Refueling Pool cooling & purification supply and return lines between Containment isolation valves 2-RW-63 and 2-RW-154, 2-RW-21 and 2-RW-232, and between 2-RW-22 and 2-RW-34.	These lines are not in service during normal operation. Procedure changes to OP 2305 to partially drain these isolated segments when taking out of service will eliminate the potential for thermally induced overpressurization.	Complete

Question 2 *RELAP5 is a computer code with a largely empirical basis for its closure relations. Therefore, RELAP5 must be assessed against experimental data that is applicable to the present analysis. Please provide the RELAP5 assessment that was performed that qualifies it for the present application. Describe how the range of conditions in the experiments correspond to the water hammer calculations.*

Response: The response to Question 2 was developed by HOLTEC International,[7] and is provided as Enclosure 1.

Question 3 *The RELAP5 output is used in structural load calculations. Provide the assessment of the overall load methodology against applicable experimental data that qualify it for the present application. Describe how the range of conditions in the experiments correspond to the water hammer calculations.*

Response: The response to Question 3 was developed by HOLTEC International, and is provided as Enclosure 1.

Question 4 *Also, in your response to Questions 7 of the submittal, you indicated that the "procedural change would be made to instruct operators to delay restarting an idle RBCCW pump (restarting of the pump will be based on the existing post-accident Containment conditions)." The decision to start the pump should be based on the worst conditions that were experienced in Containment, not the existing conditions.*

Response: In response to GL 96-06, an analysis was performed to determine if the cooling water supply and return lines for the Millstone Unit No. 2 Containment Air Recirculation (CAR) coolers is susceptible to either water hammer or two phase flow conditions during postulated accident conditions. The postulated accident conditions are either a LOCA or

[7] Letter from Kalyan K. Niyogi of HOLTEC International to Jack Deluna, "Response to NRC RAI on Generic Letter 96-06 for Millstone Unit No. 2," dated June 25, 2001.

MSLB inside containment concurrent with Loss of Normal Power (LNP). Following the LNP, the CAR System will be restarted within 26 seconds as required by the facility Technical Requirements Manual, Section 4.0, Table 3.3-5. This time includes the Emergency Diesel startup delay time and the restart sequencing time for the Reactor Building Closed Cooling Water (RBCCW) pumps. Our analysis concluded that there is a potential for flashing and voiding to occur in the vicinity of the CAR and Control Element Drive Mechanism (CEDM) coolers and surrounding piping following an RBCCW pump trip. The pressure surges in the CAR coolers and surrounding piping could be as high as 310 psia. Our analyses shows that the CEDM and CAR coolers are within acceptable limits for these pressure surges.

Our analysis also indicated that the longer the delay in RBCCW pump restart, the larger the volume of steam voids formed in the RBCCW System, which consequently results in higher pressure transients. Further evaluation of the system was performed for two additional scenarios.

1) The manual restart of an idle RBCCW pump following LOCA with the CAR fans continuing to operate.
2) The manual start of the Emergency Diesel Generator due to failure to start automatically on a Safety Injection Actuation Signal in the event of a LOCA or MSLB. The associated RBCCW pump and the CAR fans restart upon manual restart of the Emergency Diesel Generator.

It was shown that in either of the cases mentioned above, the temperature of the steam void inside the headers closely follows the temperature of the Containment. The steam void in the CAR coolers and surrounding piping reaches a maximum size and then decreases in size as the Containment temperature decreases. It was established that the pressure transient responses are lower in both scenarios with a restart time of 45 minutes than the pressure transient for the 26 second pump restart. It was also concluded that restarting the pump in less than 45 minutes could result in significantly higher pressure transients than those evaluated for the design basis case.

Analyses were performed to determine the maximum loads which could be experienced by the RBCCW system on pump restart within the 45 minute time frame due to steam voids, assuming that the CAR fans are forcing high temperature Containment air on to the stagnant water in the CAR coolers. The calculated maximum loads for this case are higher than those analyzed for the 26 second pump restart case. Containment pressure rather than temperature is recommended as an input for making a decision to start an idle RBCCW pump.

Therefore, conservatively a Containment pressure of 20 psig is chosen since this pressure corresponds to a maximum Containment temperature of 215°F (well below 249°F) based on the predicted Containment pressure/temperature profiles for all cases analyzed. The minimum time to reach 20 psig, based on the pressure and temperature profiles for all cases analyzed is greater than an hour. This limitation on pump restart ensures that the fluid inside the CAR/CEDM coolers and the surrounding piping is well below the saturation temperature by transferring heat back to the Containment. The Containment temperature profile used as input is conservative and envelopes the current Containment analysis of record.

Docket Nos. 50-336
50-423
B18429

Enclosure 1

Millstone Nuclear Power Station, Unit No. 2
Letter from HOLTEC International Addressing RAI Questions 2 and 3

HOLTEC
INTERNATIONAL

Holtec Center, 555 Lincoln Drive West, Marlton, NJ 08053
Telephone (856) 797-0900
Fax (856) 797-0909

June 25, 2001

Mr. Jack Deluna
Millstone Unit 2
Dominion Nuclear CT
P.O. Box 128
Waterford, CT 06385

Subject: Response to NRC RAI on Generic Letter 96-06
P.O. No. 03000654 / Holtec Project No. 1173

Dear Mr. Deluna:

Please find enclosed Holtec response to Item #2 and #3 of the NRC Request for Additional Information on Generic Letter 96-06 for Millstone Unit 2.

If you have any questions, please call me at (856) 797-0900 x 644.

Thank you for the opportunity to serve you.

Sincerely yours,

Kalyan K. Niyogi, Ph.D., P.E.
Director - Technology Consulting Group

Enclosure: **GENERIC LETTER 96-06 RESPONSE**

Item #2

RELAP5 is a computer code with a largely empirical basis for its closure relations. Therefore, RELAP5 must be assessed against experimental data that is applicable to the present analysis. Please provide the RELAP5 assessment that was performed that qualifies it for the present application. Describe how the range of conditions in the experiments correspond to the water hammer calculations.

Response

RELAP5/MOD3 is a computer code [1] recommended by the USNRC for thermal-hydraulic analysis of transients and pipe-break type accidents in light-water nuclear power plants. Although the program is a primary means in establishing the mass and energy release from a spectrum of pipe breaks required to be addressed in safety analyses for nuclear power plants, it has been extensively used as an analytical tool to predict fluid transients in a piping system to develop a one-dimensional, two-fluid model, consisting of steam and water, with the possibility of the vapor phase containing a noncondensible component and the liquid phase containing a nonvolatile solute.

The fundamental single-phase equations used in the RELAP5 program are based on standard classical fluid mechanics constitutive relations and interface balance equations. Well established empirical correlations have been used in the phasic conservation equations primarily in predicting phase separation behavior in two-phase flow problems. The "closure problem" has been recognized in RELAP5 solution techniques. Closure models are required for single-phase continuum models when derived by averaging from more elementary kinetic theory models. Additional closure models are required for the Reynolds average model and also the two-fluid model. The closure problem for the single-phase continuum model consists of providing models for properties such as viscosity, conductivity, and diffusivity. For the Reynolds average model, additional constitutive models are required for the Reyonolds stresses and the Reynolds heat flux. The two-fluid cases require models for interphase interactions as well as interface models.

The present application of RELAP5 for the prediction of the fluid transient in the RBCCW system following a pump trip and subsequent restart is described briefly as follows.

1. Void Formation:

   Following the trip the pump flow drops considerably in 5 seconds and then coasts down at a relatively slower rate. At 15 seconds the flow practically ceases. Due to heat transfer from the containment atmosphere the water inside the Containment Air Cooler (CAR) units gets heated and starts boiling. Due to the generation of steam, surrounding water is pushed out. The rate of void formation drops as the CAR cooling coils become empty, since the heat transfer drops drastically.

2. Pump Restart:

The pump restarts at 26 seconds, reaches full speed fast and flow acclerates. The flow downstream of the void does not acclerate as rapidly. Consequently, the void gets compressed and shrinks.

3. Void Collapse:

Finally, the void collapses and the upstream and the downstream flow fronts hit each other, resulting in pressurization.

4. Wave Propagation:

The Pressure surge propagates both upstream and downstream as waves in subcooled liquid in the piping.

The uncertainty in the closure models are expected to have very little effect on the overall outcome of the transient. The vapor formation is affected by the heat transfer from the coils. However, as soon as the CAR is empty the effect drops out. The pump flow is well defined by its characteristics. During the collapse of the void, the closure model of the interphase heat/mass transfer is important. However, flow acceleration dominates the phenomena. The pressure surge due to meeting of two fluid columns is a well established phenomenon. Wave propagation in subcooled liquid subsequent to pressure surge is also well known.

It is clear from the above that the closure problem does not have any significant role in the prediction of the transient results in the present scenario. Furthermore, the closure and the constitutive models used in RELAP5/MOD3 are based on standard industry practices. A simplistic approximate calculation can establish the validity of the results (surge pressure, etc.) predicted by RELAP5 for the present problem.

The RELAP5 program went through an extensive validation program [1]. This included a large number of tests and separate effects experiments. However, most of them deal with two-phase flow conditions and a relatively high pressure range.

Item #3

The RELAP5 output is used in structural load calculations. Provide the assessment of the overall load methodology against applicable experimental data that qualify it for the present application. Describe how the range of conditions in the experiments correspond to the water hammer calculations.

Response

The unbalanced transient forces on the piping segments are calculated by a postprocessor [2] using the results from RELAP5. The time-dependent liquid and gas densities and velocities following a precipitating event such as a sudden valve closure or pump start and vapor bubble collapse are assembled into output files. The net unbalanced force, or wave force, on each segment of the piping system is obtained for each time step by determining the momentum change across the volumes in each segment.

EPRI performed a series of full-scale tests [3] on a set of relief and safety valves using a simplified piping system to acquire data with which to assess the performance of RELAP5. Instrumentation was employed which allowed measuring of transient fluid loads as well as physical properties. Model data comparisons were performed for five different typical tests: steam, two for steam with a cold loop seal, steam with a warm loop seal and saturated liquid. It was found that RELAP5 results were adequate for calculation of safety and relief valve discharge piping hydrodynamic loads. It was found to be a valuable tool to predict hydrodynamic loads in piping systems.

In the present application of RELAP5, for the calculation of piping loads, the fluid condition in the piping remained subcooled following the void collapse. The fluid conditions of the EPRI experiments included subcooled water. Although the actual pressure and temperature conditions are different, the tests establish the adequacy of the methodology for the calculation of the piping unbalanced loads.

REFERENCES

1. NUREG/CR-5535, RELAP5/MOD3 Code Manual, Idaho National Engineering Laboratory, August 1995.

2. Holtec International Report No. HI-971809, Rev. 0, RFPP Computer Program.

3. EPRI NP-2479, Application of RELAP5/MOD1 for Calculation of Safety and Relief Valve Discharge Piping Hydrodynamic Loads, December 1982.