



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

PRIMARY COLD LEG PIPING LEAK-BEFORE-BREAK REVISED ANALYSIS

NORTHEAST NUCLEAR ENERGY COMPANY

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

1.0 INTRODUCTION

By letters dated June 25, 1998 (Ref. 1), and September 9, 1998 (Ref. 2), Northeast Nuclear Energy Company (NNECO/licensee) submitted a request for NRC review and approval of NNECO's reanalysis of the leak-before-break (LBB) status of the Millstone Nuclear Power Station, Unit No. 2 (MNPS-2), reactor coolant loop (RCL) piping. This reanalysis was necessary since upon replacement of the MNPS-2 steam generators in 1992, the loadings for sections of the piping (the RCL crossover and cold legs) were no longer bounded by the licensee's previous LBB evaluation. NNECO's submittals were based on the provision of General Design Criterion 4 (GDC 4) of Title 10 of the Code of Federal Regulations, Part 50 (10 CFR Part 50), Appendix A, which states in part,

[h]owever, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

For the purposes of this demonstration, the licensee submitted an LBB reanalysis prepared by Combustion Engineering (CE) for the crossover and cold leg portions of the RCL piping and reaffirmed the applicability of the previous staff approval for the RCL hot legs. These evaluations were based on the methodology in the previously approved CE Owners Group (CEOG) Topical Report CEN-367-A (Ref.3) and was mostly consistent with the methodology contained in NRC NUREG-1061, Volume 3 (Ref. 4), and/or Draft Standard Review Plan (DSRP) Section 3.6.3. Analyses consistent with this NRC staff guidance have been previously approved by the Commission as demonstration of an extremely low probability of piping system rupture.

2.0 REGULATORY REQUIREMENTS AND STAFF POSITIONS

Nuclear power plant licensees have, in general, been required to consider the dynamic effects that could result from the rupture of sections of high energy piping (fluid systems that during normal plant operations are at a maximum operating temperature in excess of 200 °F and/or a maximum operating pressure in excess of 275 psig). This requirement has been formally included in 10 CFR Part 50, Appendix A, GDC 4, which states, in part, "[s]tructures, systems, and components important to safety...shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from

9811160302 981109
PDR ADOCK 05000336
P PDR

Enclosure

equipment failures and from events and conditions outside the nuclear power unit." For facilities such as MNPS-2, which were licensed prior to the advent of the GDC, these requirements were included as part of plant-specific licensing reviews.

The philosophy of "leak-before-break" behavior for high energy piping systems was developed by the NRC in the early 1980s, was used in certain evaluations stemming from Unresolved Safety Issue A-2, "Asymmetric Blowdown Loads on PWR Primary Systems," and was subsequently expanded for application toward resolving issues regarding defined dynamic effects from high energy piping system ruptures. The methodology developed by the NRC for performing LBB analyses was thoroughly described in NUREG-1061, Volume 3, and summarized in DSRP Section 3.6.3, "Leak-Before-Break Evaluation Procedures," which was published for public comment in August 1987.

The licensee recently identified a condition at MNPS-2 in which sections of RCL piping were no longer bounded by the licensee's previous LBB evaluation after replacement steam generators were installed in 1992. The licensee informed the NRC of this issue via letter dated January 30, 1998, and as discussed in Section 1.0 of this SE, has addressed the problem by performing a LBB reanalysis of the RCL cold leg and crossover leg piping and reaffirming the LBB behavior of the MNPS-2 hot legs. In addition, the licensee requested an extension in its use of RCL LBB behavior. Previously, by letter dated September 1, 1992 (Ref. 5), the NRC had agreed that the licensee could credit RCL LBB behavior toward the design basis of the MNPS-2 neutron shield tank. In this submittal, the licensee requested that, in addition to the design of the neutron shield tank, NNECO be permitted to credit RCL LBB behavior toward the design of the facility's core barrel snubbers, core barrel support ledge, and core barrel stabilizer blocks.

3.0 LICENSEE'S DETERMINATION

The following discussion contains information supplied by NNECO in References 1, 2, and 3. The attachments to the June 25, 1998, letter included report N-PENG-EG-007, "Justification of Continued LBB Compliance for NU MP2," and calculation set MISC-ME-C-144, "Demonstration of Feasibility of Using JEST Program for MS-2 L.B.B. Analysis of Cold Leg," prepared by Asea Brown Boveri/CE for the licensee. Since reanalysis of the MNPS-2 hot legs was not required, the remainder of this section will only discuss those portions of the MNPS-2 RCL that were reanalyzed.

3.1 Identification of Analyzed Piping and Piping Material Properties

The licensee's submittals identified and analyzed the following sections of high energy piping for LBB behavior verification. For each MNPS-2 RCL, the licensee's submittals addressed the piping from the steam generator to the reactor coolant pumps (RCPs) (defined herein as the crossover leg) and the piping from the RCPs to the pressure vessel (the cold leg). This piping is shown in Figure 1 (attached).

The RCL crossover and cold legs were identified as having the following material components. The main piping sections and the welds connecting them were manufactured from American Society for Mechanical Engineers (ASME) SA-516 Grade 70 carbon steel and clad with stainless steel. In the CEN-367-A report, the RCPs were identified as being manufactured from Type 304 stainless steel with safe ends made from ASME SA-351 Grade CF-8M cast stainless steel (CSS). The safe ends were then welded to the ferritic piping using Inconel 182 weld filler metal.

For the material properties used in the LBB analysis, NNECO/CE used data that was originally submitted to the staff in the CEN-367-A report and was specific to the material being evaluated at a particular location. Weld locations in the carbon steel piping, based on inferior material properties and higher stresses when compared to base material, were chosen as analysis locations as well as the stainless steel safe ends at the RCP. Archival samples and/or test data specific to the MNPS-2 materials were not available. The stress-strain curves for the SA-516 Grade 70 submerged arc weld (SAW) materials were based on generic data available from CE piping material test programs (Ref. 6). The J-resistance (J-R) data for the SA-516 Grade 70 SAW material was taken from the work performed by Battelle Columbus Laboratory for the NRC and summarized in NUREG/CR-4082, BMI-2120, Vol. 4 (Ref. 7). For the stainless steel materials, the licensee stated that stress-strain curves and the J-R data for Type 304 stainless steel from the same data sources would be representative. Furthermore, since the SA-516 Grade 70 weld J-R data was conservative with respect to the Type 304 stainless steel J-R data, the licensee concluded that for this analysis it would be appropriate to use the SA-516 Grade 70 data as a conservative bound for all materials in the system.

3.2 General Aspects of the Licensee's LBB Analysis

In this analysis, the licensee sought to reaffirm the LBB behavior of the subject piping and the NNECO/CE analysis addressed those aspects of the analysis from the CEN-367-A report, which changed with the installation of the replacement steam generators. As such, the analysis directly examined the impact of the recalculated piping loads during normal operation (NOP) and safe-shutdown earthquake (SSE) conditions on the critical flaw margin and leakage flow stability criteria. The licensee's analysis made use of the CE code JEST for assessing the fracture mechanics behavior of the leakage flow and critical flow. In CEN-367-A, the use of the JEST code was found to be always conservative with respect to finite element modeling results.

3.3 Evaluation of Reactor Coolant Loop Crossover Leg and Cold Leg Piping

The NNECO/CE analysis in CEN-367-A and in the current submittal was initiated by determining the appropriate leakage flow size for the analysis. Without revisiting the complete bases for its conclusion, the licensee postulated, consistent with CEN-367-A, that the leakage from a through-wall flow in the RCL would be 250 gallons per square inch of crack opening area, that the crack width would be 0.1 inch, and therefore a 7.5-inch through-wall flow would be sufficient to provide 10 gallons per minute (gpm) of leakage at any piping location. Since the MNPS-2 containment leakage detection system has the capability of detecting 1 gpm of leakage in the course of 1 hour (consistent with NRC Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," guidance), a margin of 10 (consistent with the requirements of NUREG-1061, Vol. 3) on the detectable leakage would be met by use of the 7.5-inch leakage flow. Based on this, the assumed critical flow size for any location was then 15.0 inches (to provide the NUREG-1061, Vol. 3, recommended margin of 2 between the critical flow size and the leakage flow size). The licensee's reanalysis then addressed the two other objectives: demonstrate that this critical flow size was stable for all analyzed piping locations when it was subjected to (NOP+SSE) loads and demonstrate that the leakage size flow was stable under the $\sqrt{2}$ * (NOP+SSE) loads.

Since the 7.5-inch leakage flow and 15.0-inch critical flow were postulated for all locations, the licensee concluded that demonstrating that these flows were acceptable at the piping location of greatest NOP+SSE stresses would provide a bounding analysis for all locations. For each location, the NOP deadweight axial forces and NOP thermal axial forces were summed

algebraically while their bending moments were summed componentially by a square-root-of-the-sum-of-the-squares method (the torsional moment was, however, not included in the licensee's evaluation). These NOP sums were then added absolutely to the pressure-induced axial force and SSE axial forces and bending moments. To compare the locations, the licensee converted the total axial force into an equivalent bending moment based on equalizing the midwall stresses. Based on this methodology, the licensee identified location 6 (at the RCP suction nozzle) as the bounding location with an equivalent bending moment based on pressure+NOP+SSE loads of 54,273 inch-kips and $\sqrt{2}$ (pressure+NOP+SSE) of 76,753 in-kips. The licensee's graphical analysis of the flaw stability and margin criteria are shown in Figures 2 and 3. Failure of the critical flaw to be stable under pressure+NOP+SSE loads or failure of the leakage flaw to be stable under $\sqrt{2}$ (pressure+NOP+SSE) loads would be represented by the intersection of the loading curves with the SA-516 Grade 70 (dJ/da) vs. J material property curve. Since the graphs do not intersect, the licensee concluded that the margins required by NUREG-1061, Vol. 3, were met and the LBB behavior of the crossover and cold legs of the RCL was demonstrated.

4.0 STAFF CONFIRMATORY ANALYSIS

Based on the information provided by the licensee regarding the materials comprising the MNPS-2 RCL piping and the loads under NOP and SSE conditions, the staff independently assessed the compliance of this system with the LBB criteria established in NUREG-1061, Vol. 3. While the staff has concluded that the analyses submitted by the licensee were sufficient to demonstrate that LBB behavior would be expected from the subject piping, the following sections will focus on the differences between the details of the staff's confirmatory analysis, conducted in accordance with NUREG-1061, Vol. 3, and the licensee's. This piping system was demonstrated to be a candidate for LBB evaluation in the CEN-367-A report given the absence of degradation mechanisms or atypical loading events during the operating history of CE reactors. The replacement of the steam generators at MNPS-2 does not invalidate that conclusion.

4.1 Identification of Analyzed Piping and Piping Material Properties

The staff examined the list of materials identified for the RCL piping and concluded that the materials of primary interest for the LBB analysis would be the CSS safe ends because of their susceptibility to thermal aging or the SA-516 Grade 70 weld locations. The bimetallic welds between the CSS safe end and the ferritic piping were identified by the licensee as having been manufactured with Inconel 182 and, therefore, decarburization of the ferritic base metal is not a concern for these welds.

NUREG-1061, Vol. 3, specifies particular aspects that should be considered when developing materials property data for LBB analyses. First, data from the testing of the plant-specific piping materials is preferred. However, in the absence of such data, more generic data from the testing of samples having the same material specification may be used. More specifically, it was noted in Appendix A of the NUREG that "[m]aterial resistance to ductile crack extension should be based on a reasonable lower-bound estimate of the material's J-resistance curve," while section 5.2 of the NUREG stated that the materials data should include "appropriate toughness and tensile data, long-term effects such as thermal aging and other limitations."

The staff had previously concluded in its review of CEN-367 that the materials property data obtained from References 6 and 7 for SA-516 Grade 70 material was acceptable. However, concerning the CSS safe end material, the staff noted in its safety evaluation on the CEN-367 that the effects of thermal aging on the material properties of the CSS should be considered and that the staff would use information from NRC-sponsored work for its independent evaluation. This continues to be the staff position. Since no δ -ferrite compositions were provided for the CSS safe ends, the staff's evaluation assumed that conservatively high amounts were present (which increases the materials sensitivity to thermal aging). Results from work at Argonne National Laboratory (References 8 and 9), sponsored by the NRC, were used as the basis for developing the J-R and stress-strain curves for the CSS material.

4.2 General Aspects of the Staff's LBB Analysis

The staff's confirmatory analysis was performed in accordance with the guidance provided in NUREG-1061, Vol. 3. Based on the information submitted by the licensee, the staff determined the critical flaw size at the bounding location for the RCL using the codes compiled in the NRC's Pipe Fracture Encyclopedia (Ref. 10). For the purposes of the staff's evaluation, the critical location was defined by those locations at which materials with low postulated fracture toughness existed in combination with high ratios of SSE-to-NOP stresses. This was because high SSE stresses tend to reduce the allowable critical flaw size while low NOP stresses increase the size of the leakage flaw required to produce 10 gpm of leakage. In particular, when evaluating the critical flaw in thermally-aged CSS base materials, the staff used the LBB.ENG2 code developed by Brust and Gilles (Ref. 11). When evaluating pipe welds, the staff would use the LBB.ENG3 code developed by Battelle (Ref. 11) for that express purpose if a substantial difference in the tensile properties of the weld and base metal were expected. In this evaluation, however, the tensile properties of the SA-516 Grade 70 weld material were expected to be not significantly different from the tensile properties of the surrounding SA-516 Grade 70 piping and/or ferritic nozzle material and therefore the LBB.ENG2 code was also used to analyze the weld locations. The same criteria as discussed in Sections 3.3 and 3.4 with regard to the applied J exceeding the material J_{IC} and the applied dJ/da exceeding the material's $d(J-R)/da$ were used to identify the critical crack size.

The staff then compared the critical flaw at the bounding location to the leakage flaw which provided 10 gpm of leakage under NOP conditions to determine whether the margin of 2 defined in NUREG-1061, Vol. 3, was achieved. The leakage flaw size calculation was carried out using the Pipe Crack Evaluation Program (Revision 1) analytic code developed by the Electric Power Research Institute. The 10 gpm value was defined by noting that the compliance of the MNPS-2 containment leakage detection system with the positions in Regulatory Guide 1.45 indicates that this system would be able to detect a 1 gpm leak in the course of one hour and a factor of 10 is applied to this 1 gpm detection capability to account for thermohydraulic uncertainties in calculating the leakage through small cracks. The stability of the leakage flow under loadings a factor of $\sqrt{2}$ greater than the combination of SSE+NOP loads was subsequently evaluated to check the final acceptance criteria of NUREG-1061, Vol. 3.

4.3 Evaluation of the MNPS-2 Reactor Coolant Loop Crossover and Cold Legs

The staff's evaluation first examined the loadings submitted by the licensee. It was noted that the summation methodology utilized by the licensee was not completely consistent with the guidance provided by the staff in NUREG-1061, Vol. 3, or DSRP 3.6.3 for determining loads for LBB analyses. This inconsistency was apparent in two aspects of the licensee's analysis: one, the licensee did not include the torsional moments (as directed to in NUREG-1061, Vol. 3) in its

moment summation and, two, the licensee's load summation was a combination of the algebraic and absolute summation methodologies discussed in NUREG-1061, Vol. 3, and DSRP 3.6.3.

However, the licensee did include the torsional moment components for each piping location in its submittal so that information was available for the staff's evaluation and the licensee demonstrated to reference 2 that its summation methodology was at least as conservative as the algebraic summation method outlined by the staff.

Based on the staff's evaluation of the loadings supplied by the licensee, the staff concluded that the limiting locations for the RCL piping evaluation would be location 6 (at suction nozzle of the RCP) or location 8 (at the piping connection to the RPV inlet nozzle) as shown in Figure 1. At location 6, the staff's evaluation considered the possibility of a crack in the CSS safe end between the ferritic piping and the RCP. At location 8, the staff's evaluation postulated a crack in the SA-516 Grade 70 weld between the cold leg piping and the RPV nozzle safe end.

The staff first evaluated location 6. The material properties assumed by the staff for the CSS safe end material are shown in Table 1 and the loads used in the staff's leakage flow and critical flow analysis (which include the torsional moments) are shown in Table 2. The staff applied the PICEP code using the nominal piping dimensions and a crack surface roughness of $\epsilon = 0.0003$ inches. This produced a 10 gpm leakage flow of 6.1 inches in length. It should be noted that, consistent with the conclusions stated in the staff's SE on the CEN-367-A report, the staff still does not concur with the licensee's use of 250 gpm/in² for calculating the leakage rate based only on a consideration of crack opening area. The critical flow size determined by using the LBB ENG2 code was 22.2 inches for the CSS material. Therefore, the ratio of the critical-to-leakage-flow size for location 6 was $(22.2/6.1) = 3.63$, which exceeds the recommended margin of 2 in NUREG-1061, Vol. 3. The leakage flow was also shown to be stable under $\sqrt{2}$ times the summation of the NOP and SSE loads (as calculated by the licensee), which ensures that the margin on loading recommended by NUREG-1061, Vol. 3, was also achieved. Therefore, LBB behavior was demonstrated for location 6.

The staff then evaluated location 8. The methodology used was the same as used for location 6 and the material properties assumed by the staff for the SA-516 Grade 70 weld material are shown in Table 3 and the loads used in the staff's leakage flow and critical flow analysis (which include the torsional moments) are shown in Table 2. The 10 gpm leakage flow determined for location 8 was 8.6 inches in length, while the critical flow size was found to be 21 inches. Therefore, the ratio of the critical-to-leakage flow size for location 6 was $(21/8.6) = 2.44$, which exceeds the recommended margin of 2 in NUREG-1061, Vol. 3. The leakage flow was also shown to be stable under $\sqrt{2}$ times the summation of the NOP and SSE loads (as calculated by the licensee), which ensures that the margin on loading recommended by NUREG-1061, Vol. 3, was also achieved. Therefore, LBB behavior was also demonstrated for location 8 and consequently for all RCL crossover and cold leg piping.

5.0 CONCLUSION

Based on the information and analysis supplied by the licensee, the staff was able to independently assess the LBB status of the MNPS-2 RCL piping. The staff has concluded that the licensee has demonstrated that the LBB behavior of the MNPS-2 RCL hot legs is still covered by the analysis in CEN-367-A and that the RCL crossover and cold legs are addressed by the additional evaluations reviewed in this SE. Furthermore, the licensee is permitted to credit this conclusion for eliminating the dynamic effects associated with the postulated rupture of these

sections of piping from the MNPS-2 facility licensing basis, consistent with the provisions of 10 CFR Part 50, Appendix A, GDC 4, for the design basis of the neutron shield tank, core barrel snubbers, core barrel support ledge, and core barrel stabilizer blocks.

Attachments: References (1-11)
 Tables (1-3)
 Figures (1-3)

Principal Contributor: Matthew Mitchell

Date: November 9, 1998

REFERENCES

1. Bowling, M.L. (NNECO) to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station Unit No.2, Leak Before Break Revised Evaluation of the Primary Cold Leg Piping, Request for NRC Review for Continued Applicability of Report CEN-367-A," June 25, 1998.
2. Bowling, M.L. (NNECO) to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station Unit No.2, Additional Information Concerning Leak Before Break Evaluation of the Primary Cold Leg Piping," September 9, 1998.
3. Topical Report CEN-367-A, "Leak-Before-Break Evaluation of Primary Coolant Loop Piping in Combustion Engineering Designed Nuclear Steam Supply Systems," Combustion Engineering Owners Group, 1991.
4. NUREG-1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks," November 1984.
5. Vissing, G.S. (USNRC) to Opeka, J.F. (NNECO), "Application of RCS Leak Before Break Analysis," September 1, 1992.
6. EPRI Report NP-5057, "Analysis of Cracked Pipe Weldments," Ganta, B.R., Ayres, D.J., February, 1987.
7. NUREG/CR-4082, Vol. 4, "Degraded Piping Program - Phase II," Battelle's Columbus Division, Semi-Annual Report, October 1985 - March 1986.
8. Chopra, O.K., "Estimation of Fracture Toughness of Cast Stainless Steels during Thermal Aging in LWR Systems," NUREG/CR-4513, ANL-93/22, Rev.1.
9. Michaud, W.F., et al., "Tensile-Property Characterization of Thermally Aged Cast Stainless Steels," NUREG/CR-6142, ANL-93/35.
10. Pipe Fracture Encyclopedia, produced on CD-ROM by Battelle-Columbus Laboratory for the U.S. Nuclear Regulatory Commission, 1997.
11. Brust, F.W., et al., "Assessment of Short Through-Wall Circumferential Cracks in Pipes," NUREG/CR-6235, BMI-2179.

TABLES
(1 - 3)

Table 1: Parameters used in Staff Evaluation of Millstone 2 Cast Stainless Steel Safe Ends

Parameter	Value
Young's Modulus	25500 ksi
Yield Strength	32.8 ksi
Ultimate Tensile Strength	78.8 ksi
Sigma-zero	32.2 ksi
Epsilon-zero	0.00129
Ramberg-Osgood Alpha	1.276
Ramberg-Osgood n	6.6
C	2599 in-lb / in ²
n	0.31

Note: $J = C(\Delta a)^n$

Table 2: Loads Used in the Staff's Evaluation of Locations 6 and 8

	Location 6	Location 8
Pressure	408 kips	408 kips
Normal Ops. Axial	- 251 kips	- 36 kips
Normal Ops. Bending	18134 in-kips	4341 in-kips
SSE Axial	157 kips	115 kips
SSE Bending	19749 in-kips	35005 in-kips

Table 3: Parameters used in Staff Evaluation of Millstone 2 SA-516 Grade 70 Piping Welds

Parameter	Value
Young's Modulus	28000 ksi
Yield Strength	35.0 ksi
Ultimate Tensile Strength	75.0 ksi
Sigma-zero	35.0 ksi
Epsilon-zero	0.00125
Ramberg-Osgood Alpha	1.53
Ramberg-Osgood n	5.66

Note: J-Resistance curve used in the staff's analysis was a point-by-point representation of the lowest curve submitted as Figure 7.6 of CEN-367-A and has been included in this SE as Figure 3. No convenient parameterization for the J-R vs. Δa curve was developed.

FIGURES
(1 - 3)

Figure 1: Piping Layout of Millstone Unit 2 Main Coolant Loop

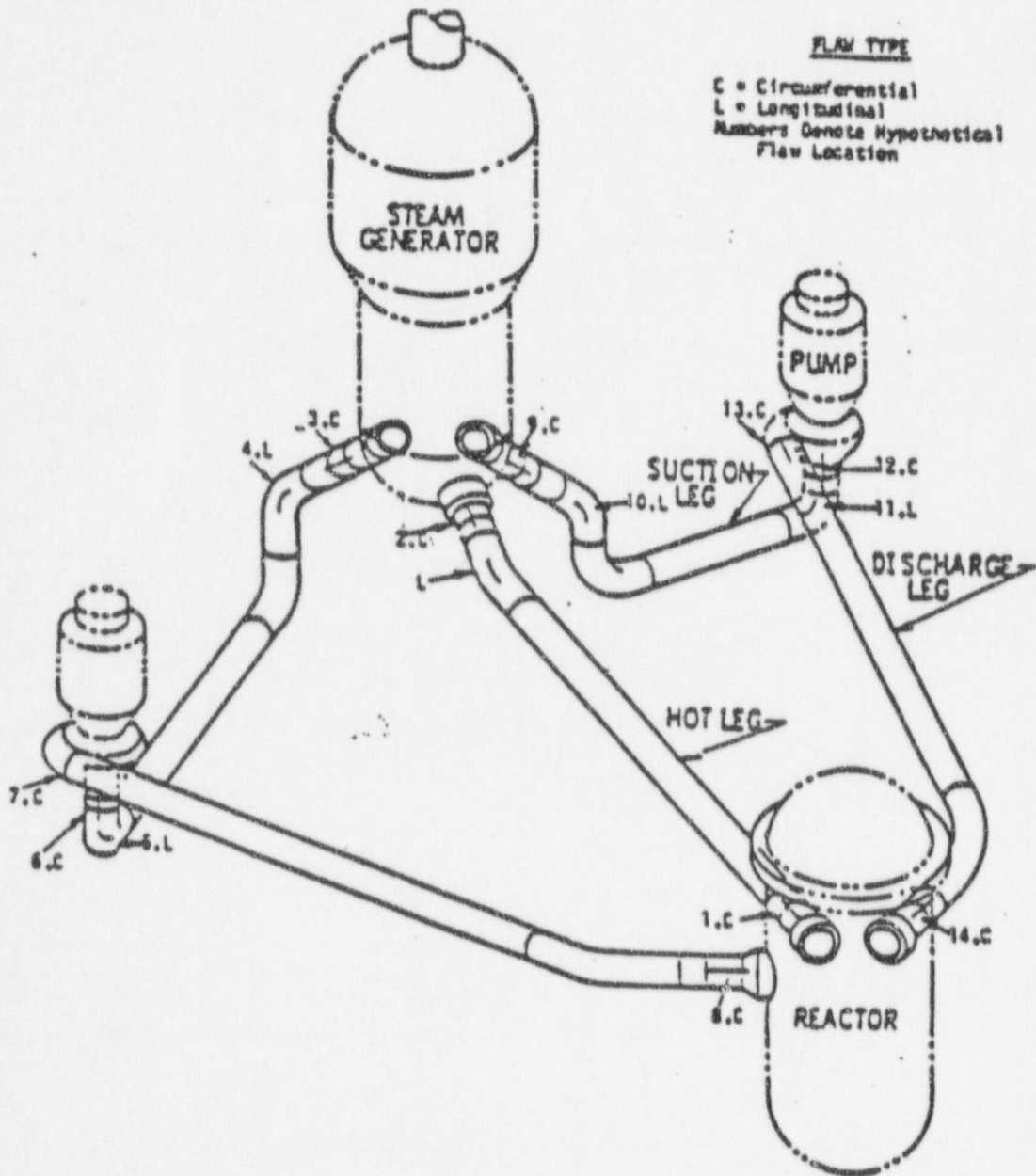


Figure 2: Licensee's Stability Evaluation for Flaws at Location 6
Using SA-516 Gr. 70 Tensile Properties

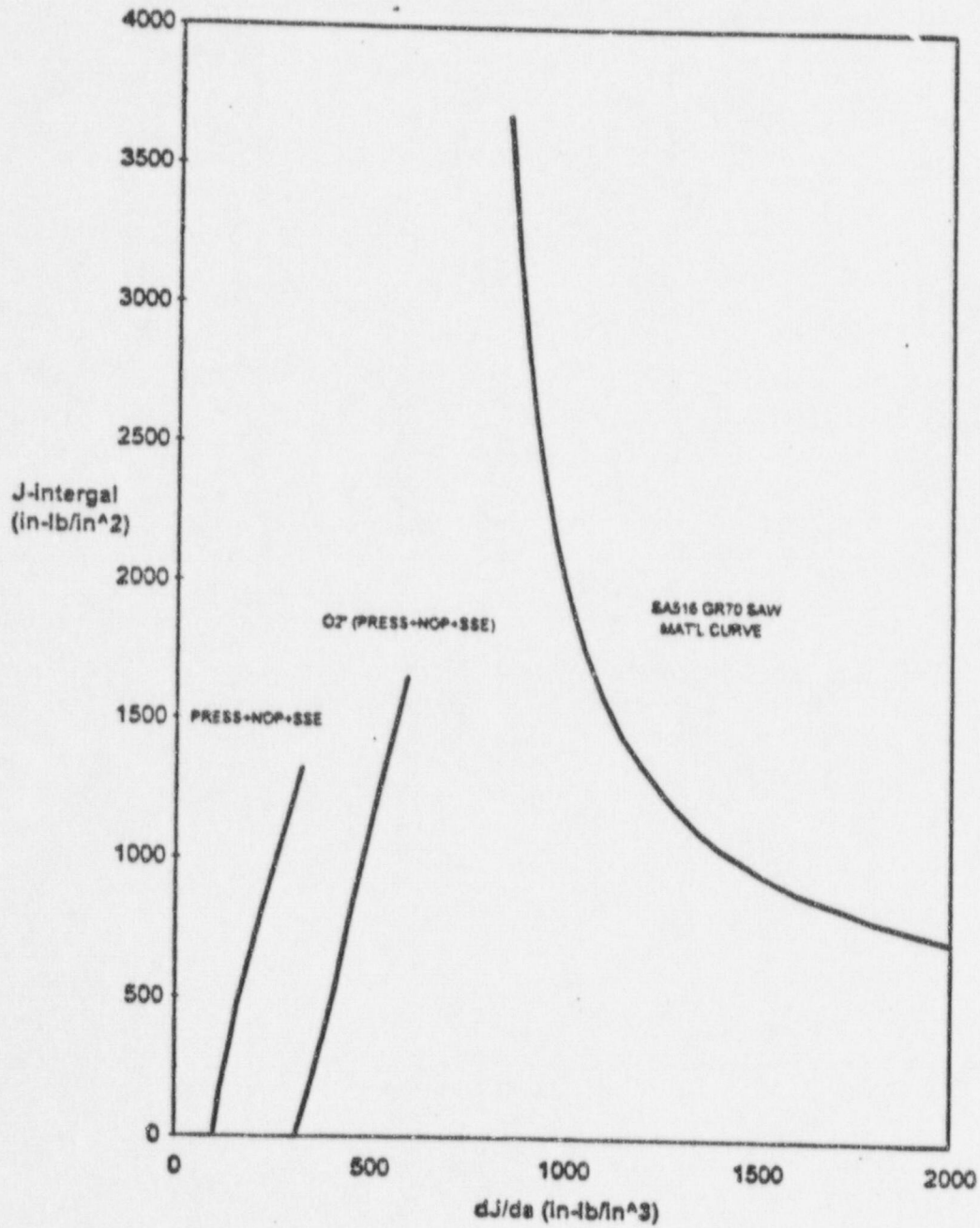
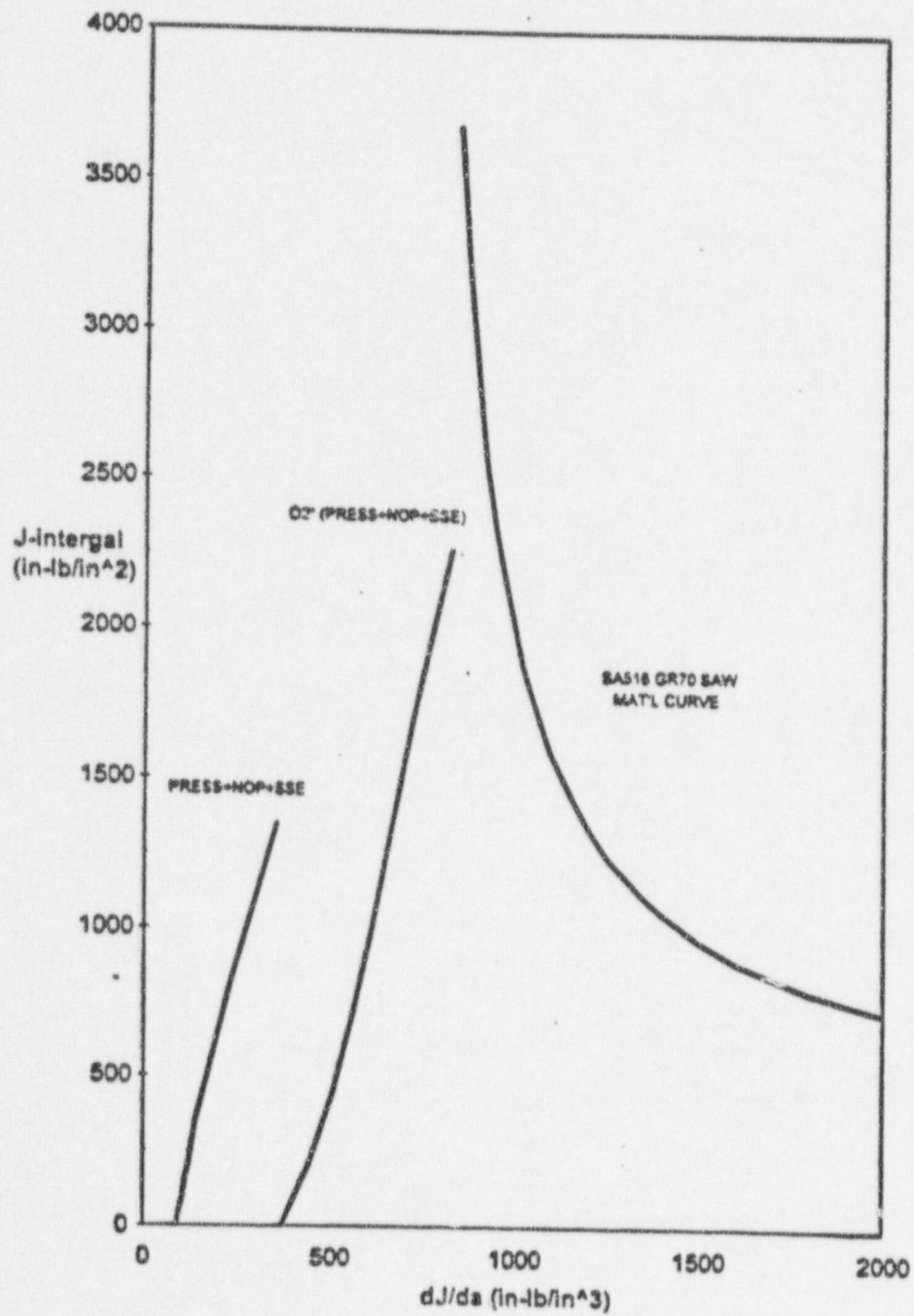


Figure 3: Licensee's Stability Evaluation for Flaws at Location 6
Using SA-376 Type 304 Tensile Properties



Millstone Nuclear Power Station
Unit 2

cc:

Lillian M. Cuoco, Esquire
Senior Nuclear Counsel
Northeast Utilities Service Company
P. O. Box 270
Hartford, CT 06141-0270

Mr. John Buckingham
Department of Public Utility Control
Electric Unit
10 Liberty Square
New Britain, CT 06051

Edward L. Wilds, Jr., Ph.D.
Director, Division of Radiation
Department of Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

First Selectmen
Town of Waterford
15 Rope Ferry Road
Waterford, CT 06385

Mr. Wayne D. Lanning, Director
Millstone Inspections
Office of the Regional Administrator
475 Allendale Road
King of Prussia, PA 19406-1415

Charles Brinkman, Manager
Washington Nuclear Operations
ABB Combustion Engineering
12300 Twinbrook Pkwy, Suite 330
Rockville, MD 20852

Senior Resident Inspector
Millstone Nuclear Power Station
c/o U.S. Nuclear Regulatory Commission
P.O. Box 513
Niantic, CT 06357

Mr. F. C. Rothen
Vice President - Work Services
Northeast Utilities Service Company
P. O. Box 128
Waterford, CT 06385

Ernest C. Hadley, Esquire
1040 B Main Street
P.O. Box 549
West Wareham, MA 02576

Mr. John F. Streeter
Recovery Officer - Nuclear Oversight
Northeast Utilities Service Company
P. O. Box 128
Waterford, CT 06385

Mr. John Carlin
Vice President - Human Services
Northeast Utilities Service Company
P. O. Box 128
Waterford, CT 06385

Mr. Allan Johanson, Assistant Director
Office of Policy and Management
Policy Development and Planning
Division
450 Capitol Avenue - MS# 52ERN
P. O. Box 341441
Hartford, CT 06134-1441

Mr. M. H. Brothers
Vice President - Operations
Northeast Nuclear Energy Company
P.O. Box 128
Waterford, CT 06385

Mr. J. A. Price
Director - Unit 2
Northeast Nuclear Energy Company
P.O. Box 128
Waterford, CT 06385

Millstone Nuclear Power Station
Unit 2

cc:

Mr. Leon J. Olivier
Chief Nuclear Officer - Millstone
Northeast Nuclear Energy Company
P.O. Box 128
Waterford, CT 06385

Citizens Regulatory Commission
ATTN: Ms. Susan Perry Luxton
180 Great Neck Road
Waterford, CT 06385

Deborah Katz, President
Citizens Awareness Network
P. O. Box 83
Shelburne Falls, MA 03170

The Honorable Terry Concannon
Co-Chair
Nuclear Energy Advisory Council
Room 4035
Legislative Office Building
Capitol Avenue
Hartford, CT 06106

Mr. Evan W. Woollacott
Co-Chair
Nuclear Energy Advisory Council
128 Terry's Plain Road
Simsbury, CT 06070

Little Harbor Consultants, Inc.
Millstone - ITPOP Project Office
P. O. Box 0630
Niantic, CT 06357-0630

Mr. Daniel L. Curry
Project Director
Parsons Power Group Inc.
2675 Morgantown Road
Reading, PA 19607

Attorney Nicholas J. Scobbo, Jr.
Ferriter, Scobbo, Caruso, Rodophele, PC
1 Beacon Street, 11th Floor
Boston, MA 02108

Mr. J. P. McElwain
Recovery Officer - Millstone Unit 2
Northeast Nuclear Energy Company
P. O. Box 128
Waterford, Connecticut 06385