Dudley

Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

March 30, 1990

John Z
WP

#49

Dr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Dresden Nuclear Power Station Unit 2
Systematic Evaluation Program (SEP) Status
NRC Docket No. 50-237

References: (a) J.W. Craig letter to T.J. Kovach dated November 29, 1989 transmitting IPSAR Final Report (NUREG 0823 Supplement No. 1).

(b) D.R Muller letter to T.J. Kovach dated May 1, 1989 concerning open item on SEP Topic III-1.

Dr. Murley:

In the Reference (a) final NRC report on the closure status of Systematic Evaluation Program (SEP) issues for Dresden Unit 2, one item was noted as requiring an additional technical submittal by Commonwealth Edison Company (CECo). That item, which was also identified in Reference (b), is a residual concern from the Staff's review of SEP Topic III-1. It required additional justification of the fracture toughness of the LPCI Heat Exchangers shell side with respect to expected minimum service temperature.

Enclosed is an analysis (utilizing two and three dimensional computer models) which demonstrates adequate fracture toughness to shutdown temperatures considerably lower than the 77^{o}F value cited in the references. The analysis was performed by SMC O'Donnell Inc. and completes the CECo submittals in support of the SEP program.

With the assistance of Sargent and Lundy Engineers, CECo has also completed a review of the eighty-eight (88) Dresden 2 SEP topics and corresponding resolutions, as requested in 1989 by the NRR Project Manager for Dresden. This review confirmed that no additional exemptions from requirements of the Dresden Unit 2 licensing bases are required as a result of the SEP program. It was also concluded that the resolution of each SEP topic, when the residual CECo commitments are complete, results in Unit 2 being either:

a) in compliance or consistent with the intent of new plant licensing requirements in place at the time of the topic's resolution; or

b) acceptable in terms of providing adequate protection to the health and safety of the public when compared to current licensing requirements.

AE01
1/1

Please contact this office should further information be required.

Very truly yours,

J. A. Silady

J. A. Silady
Nuclear Licensing Administrator

cc: J.W. Craig - PD III-2 Director, NRR
A.B. Davis - Regional Administrator, RIII
S.G. DuPont - Senior Resident Inspector, Dresden
P. Eng - Project Manager, NRR

/lmw:0768T

ENCLOSURE

Dresden Nuclear Power Station Unit 2
SEP Topic III-1

Classification of Structures, Components, and Systems (Seismic and Quality)

Reference: Integrated Plant Safety Analysis Report
NUREG-0823, Section 4.2 (2/83)
Supplement No. 1, Section 2.1.2,
"Fracture Toughness" (10/89)

The above section of the IPSAR Supplement No.1 identified the following open items in regards to the fracture toughness of the shell side of the Low Pressure Coolant Injection (LPCI) heat exchangers:

The NRC staff requested that the licensee determine:

1) Whether the lowest service temperature (LST) exceeds 77°F for the shell side of the (LPCI) heat exchanger,

2) The operating conditions when it does not,

3) The LST during these operating conditions, and

4) The design changes necessary so the LST does not exceed 77°F.

Historically, the LST recorded for the LPCI heat exchanger at shutdown was 51°F in February, 1989. The LPCI heat exchanger is cycled between steady-state and shutdown. Assuming no flow and a uniform temperature of 51°F in the vessel, the fracture toughness is adequate.

A two and three-dimensional fracture toughness analysis was conducted to address the issues. A summary of the analyses which were performed by SMC O'Donnell, Inc. is attached. The results of the analyses have demonstrated that the fracture toughness was adequate when the LST dropped to 51°F, therefore no design changes are necessary.

It should be noted that the IPSAR Supplement No. 1 Section 2.1.2 discussion of this topic contains an apparent typographical error which impacts the meaning and therefore warrants clarification. A mark-up of the affected page is also attached showing this error.

SMC O'DONNELL INC.
ENGINEERING DESIGN & ANALYSIS SERVICES

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(412) 655-1200
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March 16, 1990 Rev. 1

Ms. Alina Haber-Kovach
Commonwealth Edison Company
1400 Opus Place, Suite 400
Downers Grove, Illinois 60515

Subject: Dresden Nuclear Power Station - Unit 2
P. O. No. 328111, NU-4

Dear Ms. Haber-Kovach:

The remaining open item for Commonwealth Edison Company's (CECo) review of the Systematic Evaluation Program (SEP), Topic III-1, Classification of Structures, Components, and Systems is to demonstrate that the shell sides of the Low Pressure Coolant Injection (LPCI) system heat exchangers (HX) have adequate fracture toughness. In NUREG-0577, Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports; Resolution of Generic Technical Activity A-12, the USNRC calculated an average nil ductility temperature (NDT) of 40°F for plain carbon or mild steels, a group that includes A-212 B, the LPCI-HX shell material. From this, the USNRC has concluded that material in this group will have adequate fracture toughness if its temperature is maintained above 77°F. A review of the LPCI system showed that the temperature of the LPCI-HX shell will be the same as the torus temperature. The torus temperature, as determined by a recorder chart from February 1989 dropped as low as 51°F. Since this is less than 77°F, CECo needed to determine whether the shell material has adequate fracture toughness at this temperature.

The task of resolving this issue was assigned to SMC O'Donnell Inc., Pittsburgh, Pennsylvania by CECo as a request to perform a fracture toughness analysis of the LPCI-HX's shell side. The conclusion drawn from the results of this analysis is that by cycling the LPCI-HX between steady-state operation and shutdown (assuming no flow and a uniform temperature of 51°F in the vessel), the shell side of the LPCI-HX has adequate fracture toughness. Since the fracture toughness is adequate, no design changes are necessary to assure that the lowest service temperature exceeds 77°F.

This conclusion resulted from an in-depth analysis, documented in our report, Fracture Toughness Evaluation of the Shell Side of the Low Pressure Coolant Injection System (LPCI) Heat Exchanger for Dresden Station-Unit 2, dated March 1990. The analysis utilizes a two-dimensional axisymmetric finite element computer model of the heat exchanger, and a detailed three-dimensional quarter symmetry finite element computer model of the shell side of the heat exchanger, and was performed in compliance with NUREG-0577 and with the ASME Boiler and Pressure Vessel Code, Section III, Class C.

The 2-D model includes the tubesheet, the head, the tubes, the closure studs and one-half of the shell barrel (see Figure 1). Analyses that were performed using this model include the structural (internal tube side and shell side pressures and bolt preload) and thermal load conditions that the HX is subjected to. The 3-D model includes the shell side outlet nozzle, the nozzle reinforcement pad, the vertical support bracket, the tubesheet rim and the shell barrel (see Figure 2). This model represents the most highly stressed region of the HX. Displacements calculated by the 2-D model were applied to the detailed 3-D model at nodes which represent common locations in the two models.

The same loadings applied to the 2-D model were also applied to the 3-D model. The purpose of the 3-D model was to determine stresses at the structural discontinuities in the HX that could not be modeled in the 2-D axisymmetric model. By determining the location and magnitude of the maximum stress and using the material toughness K_{IC} of 76 ksi $\sqrt{in}$ (evaluated at 51°F from temperature adjusted data), the critical crack size at which a failure would occur, A_{cr}, was calculated. The K_{IC} value was obtained from an approximate K_{IC} vs. Temperature plot using A-212 B data taken from Figure B.9 of NUREG-0577 with the assumption that the two data points for A-212 B define the ductile-brittle transition region (see Figure 3). To obtain temperature adjusted data, the two data points were shifted to the right by 17°F. This is the temperature difference between the datum point at 60°F and the lowest allowable service temperature, 77°F. The straight line connecting the two adjusted transition boundary points was used to determine this conservative estimate of K_{IC} at 51°F.

To relate this information to the average nil ductility temperature (NDT) of 40°F, K_{IC} was calculated per the procedure in Rolfe and Barsom, Fracture and Fatigue Controls in Structures, Sec. 6.4, herein referred to as Reference 1. The straight line between this value, 57 ksi $\sqrt{in}$, and the adjusted upper bound of the transition region, 98 ksi $\sqrt{in}$, could also be used to conservatively evaluate K_{IC} at 51°F as indicated on Figure 3. It was judged that this K_{IC} value of 69 ksi $\sqrt{in}$ is unnecessarily conservative, so the 76 ksi $\sqrt{in}$ was used in the fracture mechanics evaluation. However, it is apparent from the extremely long calculated fatigue life discussed below that the 69 ksi $\sqrt{in}$ material toughness would also have been adequate.

For the calculated maximum principal stress, 16.7 ksi, from the 3-D analysis, the critical crack size is 5.25 inches. Using the maximum stress range and the crack growth rate equation from p.234 of Reference 1, the number of cycles, N, required for various assumed initial crack sizes, A_i, to grow to the critical crack size, A_{cr}, is calculated. See Figures 4a and 4b for plots of A_i vs. N for the LPCI-HX.

It was concluded, based upon an initial crack size of .010 inch (chosen as the maximum crack size that could go undetected by inspection of the unit at the time of its construction) and a cold shutdown temperature of 51°F, that the number of cycles that would be required to propagate a crack of this size to the critical crack size is approximately 1.5×10^6. Based solely from this fracture mechanics analysis and a system startup and shutdown cycle occurring on an average of once a month, the acceptable number of cycles is far beyond any anticipated service life of the LPCI-HX as well as several orders of magnitude beyond the maximum conceivable plant lifetime including life extension. Therefore, the fracture toughness of the shell side of the LPCI-HX would be more than adequate.

Sincerely,

SMC O'DONNELL INC.

Paul J. McMullen

Paul J. McMullen
Project Engineer

James M. Watson

James M. Watson
Project Manager

PJM/JMW/jg

c: E. Hampton

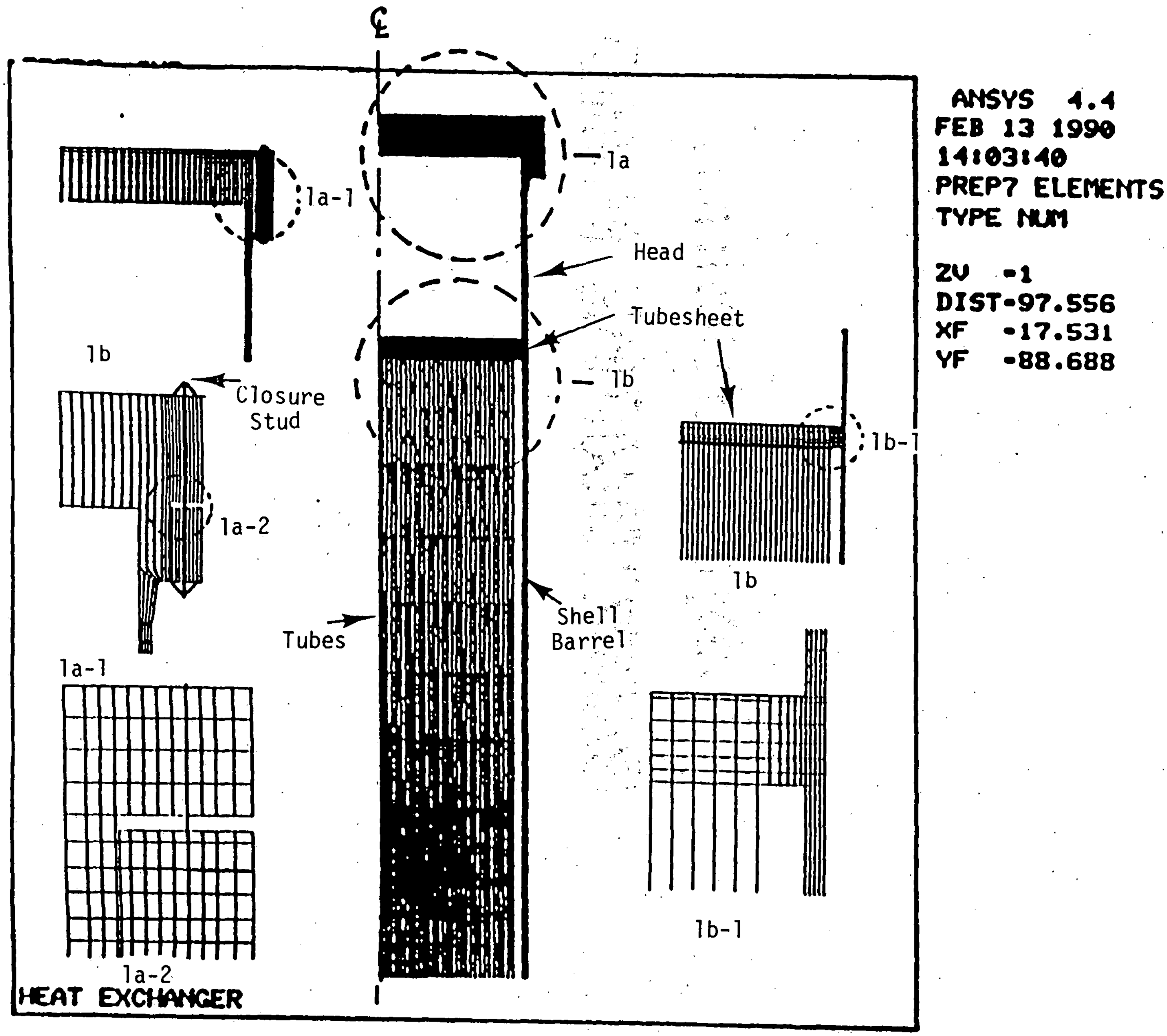


Figure 1 Two-Dimensional Axisymmetric Finite Element Model Geometry

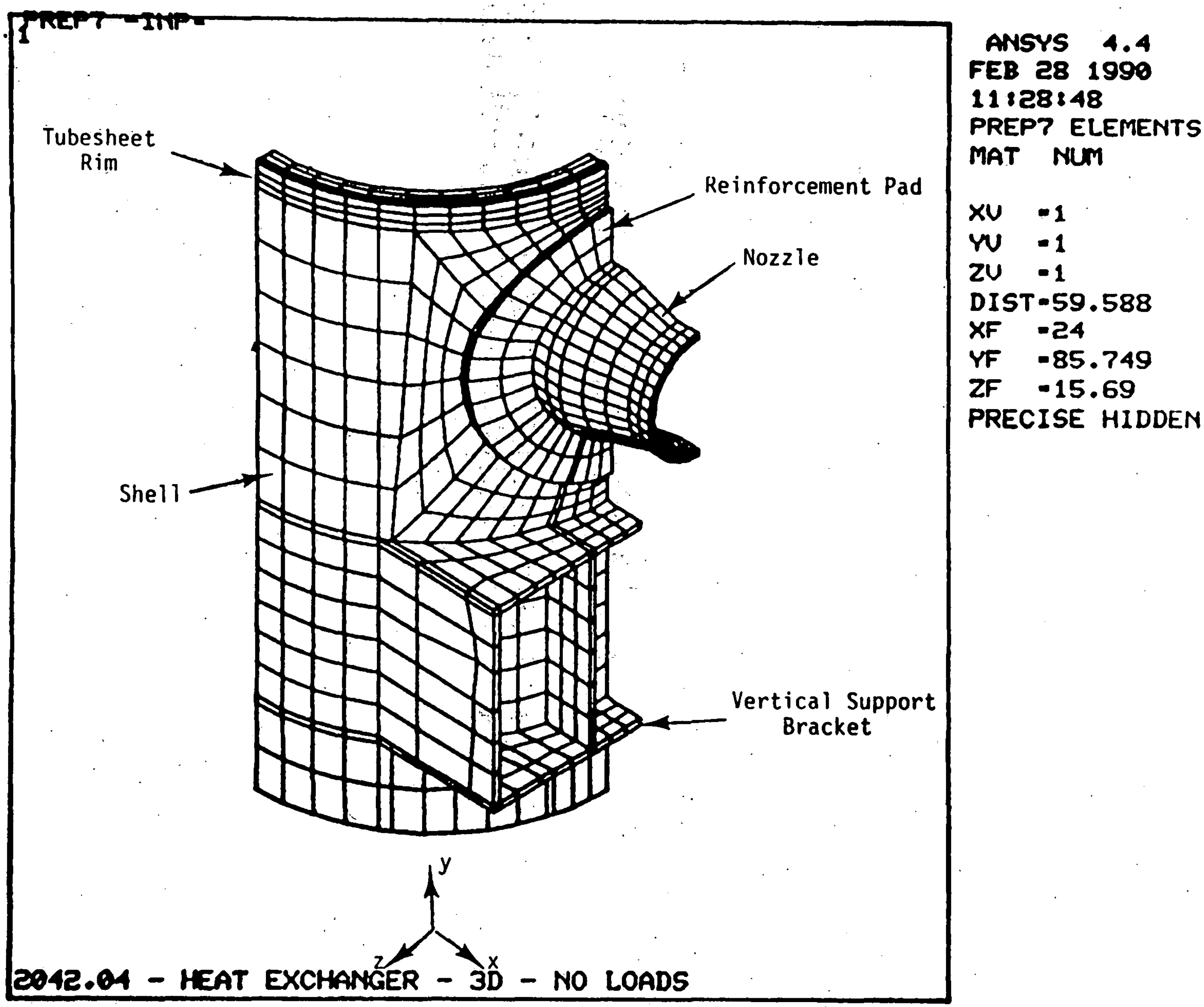
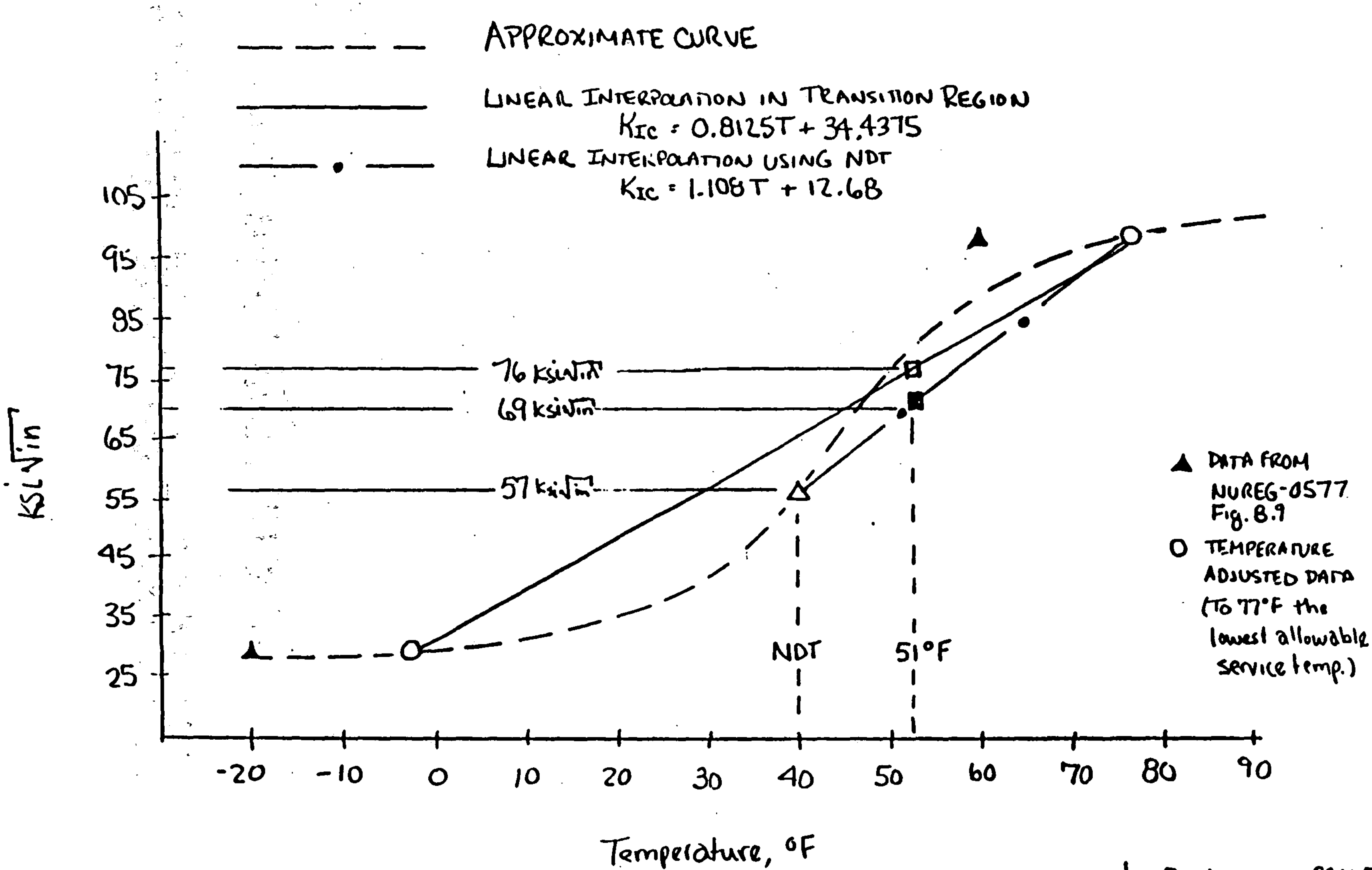


Figure 2 Three-Dimensional Finite Element Model Geometry

MATERIAL TOUGHNESS



NOTE: NOT TO SCALE

@NDT ⇒ $K_{IC} = \sqrt{EA(CVN)}$ *

$E = 28.8$ msi
$A = 5$
$CVN = 22.5$ ft·lb

$K_{IC} = 57$ ksi$\sqrt{in}$

* p. 180 of Reference 1 in Letter

Figure 3 Material Toughness

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CHKD. BY ____ DATE ____

SMC O'DONNELL INC.
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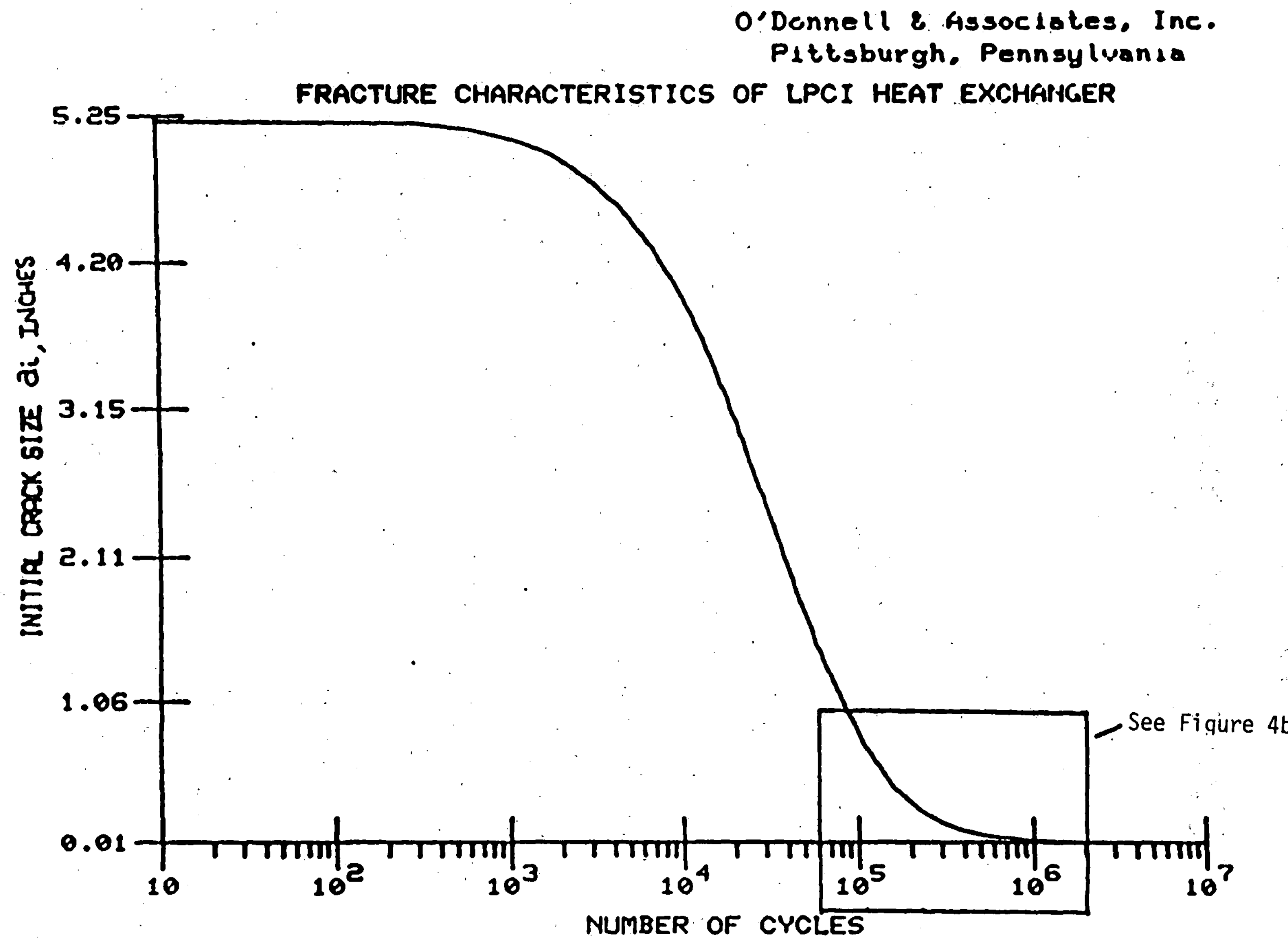

O'Donnell & Associates, Inc.
Pittsburgh, Pennsylvania
FRACTURE CHARACTERISTICS OF LPCI HEAT EXCHANGER
INITIAL CRACK SIZE a_i, INCHES
5.25
4.20
3.15
2.11
1.06
0.01
10
10^2
10^3
10^4
10^5
10^6
10^7
NUMBER OF CYCLES
See Figure 4b

Figure 4a

O'Donnell & Associates, Inc.
Pittsburgh, Pennsylvania

FRACTURE CHARACTERISTICS OF LPCI HEAT EXCHANGER

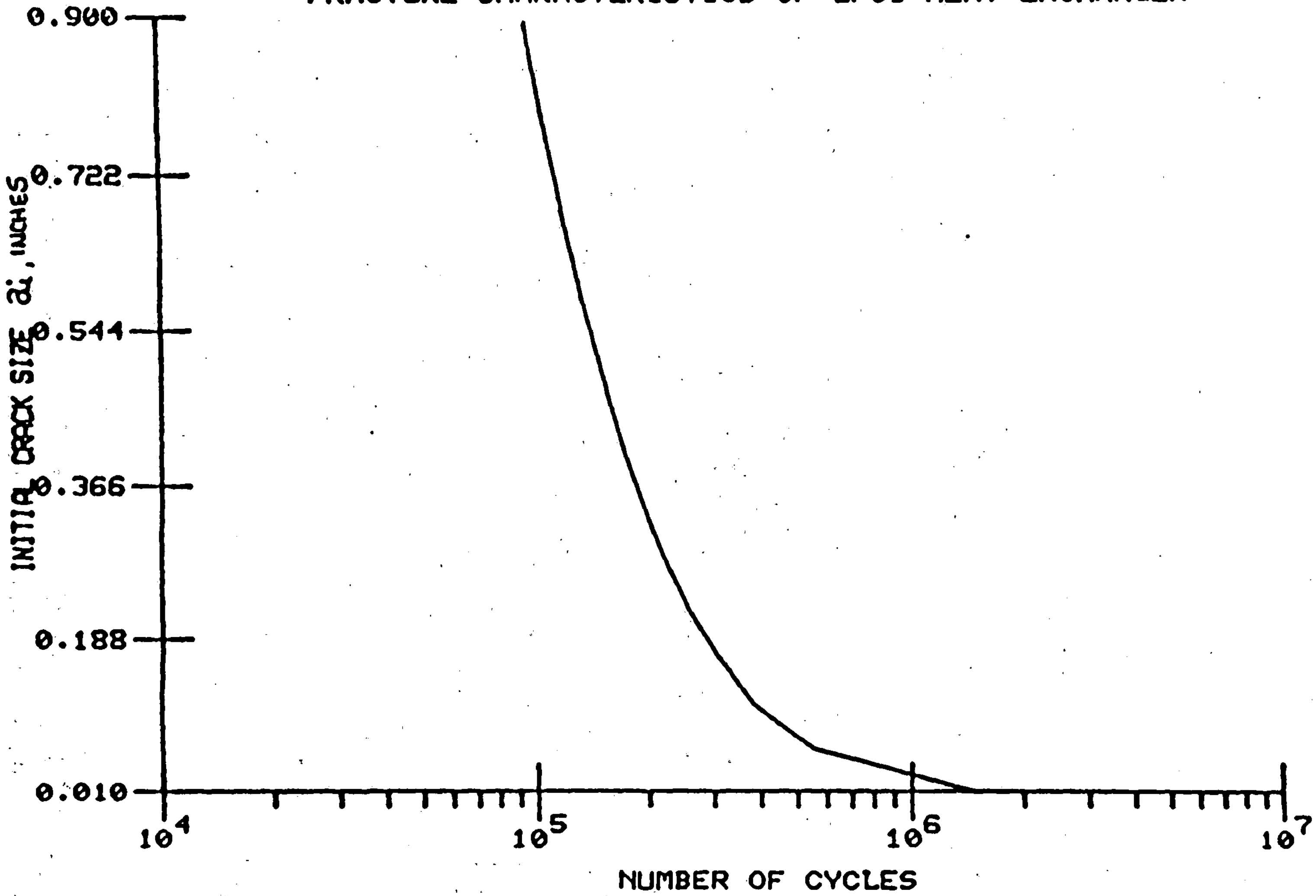

Figure 4b

- LPCI heat exchangers--shell side
- high-pressure coolant injection (HPCI) pump casings
- HPCI piping, fittings, and valves with nominal pipe diameter greater than 6 inches
- condensate/feedwater system piping from reactor vessel to outermost containment isolation valve
- main steam system piping, valves, and fittings

In Enclosure 5 to a January 19, 1983 letter, the staff indicated to the licensee that compliance with the fracture toughness requirements of later editions of the ASME Code could be demonstrated by one of the following:

(1) providing test results that meet the ASME Code requirements

(2) determining that the component's lowest service temperature (LST) is high enough to exempt the materials from testing

(3) determining that the component's failure will not result in unacceptable consequences

In letters dated April 20, 1987 and January 6, 1989, the licensee provided information to demonstrate that the components identified in its July 16, 1982 letter would meet the fracture toughness requirements of later editions of the ASME Code.

The core spray pump casings, the LPCI pump casings, and the HPCI pump casings were designated as Class B Quality Group and were constructed of carbon steel A-216, Grade WCB material. According to the ASME Code, these components would be exempt from testing, if the LST exceeds 60°F. The core spray and LPCI pumps were designed for a temperature range from 60°F to 165°F, and normally operate at approximately 95°F. The HPCI pumps were designed for a temperature range from 40°F to 140°F, and normally operate around 95°F. Although the licensee has not determined the LST, the design and normal operating temperatures for these components indicate that the materials will meet the intent of the material testing exemption requirements in the ASME Code.

The shell side of the LPCI heat exchangers were designated as Class C Quality Group and were constructed of carbon steel A-212, Grade B material. According to NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports," the 90% confidence upper bound NDT (nil ductility temperature) for this material would be 77°F. On the basis of later ASME Code requirements for this class of components, the materials on the shell side of the LPCI heat exchangers would be exempt from testing if the LST exceeds 77°F. The licensee indicates that the operating temperature of the shell side of the LPCI heat exchanger is about 40°F.

The HPCI piping, fittings, and valves were designated as Class B Quality Group and were constructed of carbon steel A-106, Grade B material. For this class component, the materials are exempt from testing if the LST exceeds 150°F. The licensee indicates that the LST for these components exceeds 150°F.

The feedwater system piping from the reactor vessel to outermost containment isolation valve and main steam system piping were designated as Class A Quality

"about" should be "above"