UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
DUKE POWER COMPANY, ET AL.	)	Docket Nos. 50-413
	)	50-414
(Catawba Nuclear Station,	)	
Units 1 and 2)	)	

AFFIDAVIT OF BARRY J. ELLIOT
IN SUPPORT OF SUMMARY DISPOSITION
OF CESG CONTENTION 18 (PALMETTO 44)

I, Barry J. Elliot, state under oath that:

1. I am a Materials Engineer in the Materials Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. I am responsible for reviewing the pressure-temperature operating limit curves and beltline material surveillance programs for all reactor vessels in nuclear power plants. I have personal knowledge of the matters set forth herein and believe them to be true and correct. A statement of my professional qualifications is attached.

2. As originally proffered, CESG Contention 18 (Palmetto 44) stated:

> The license should not issue because reactor degradation in the form of a much more rapid increase in reference temperature than had been anticipated has occurred at a number of PWR's including the Applicant's Oconee Unit 1. Until and unless the NRC and the industry can avoid reactor embrittlement, Catawba should not be permitted to operate.

This contention, though initially rejected, was "admitted as clarified" in an Order of the Licensing Board, dated July 8, 1982. The "clarified contention" states:

> The Board appears to have misread this contention. Reactor materials did indeed comply with 10 CFR Part 50; App. G requirements when tested. Intervenors' concern is with the unanticipated "rapid increase in reference temperature" which has been found in essentially all reactor vessels examined of which Oconee and Robinson are merely two nearby examples. All of these reactor vessels were required to conform to essentially the same ASME codes. Perhaps it should be stated that the "reference temperature" is the nil ductility reference temperature below which the application of sufficient stress produces a glass-like brittle fracture rather than a chewing-gum-like ductile stretching. Taffy provides a common example of ductile stretching. Above the nil ductility temperature it "pulls" to an extraordinary degree without breaking. Below the nil ductility temperature it breaks like peanut brittle in a fracture mode.
>
> Premature reactor embrittlement increases hazard because ECCS water during a LOCA can chill a reactor vessel below an elevated nil ductility temperature under conditions of appreciable stress. It is only prudent in light of what is clearly a common problem to anticipate and avoid these consequences at Catawba where no evidence exists that these reactors will behave any differently than other ASME Section III, Subsection NA Components. The Board should note the language in ASME Code, Subsection NA 1130 p. 3, (1971 ed.):
>
>> The Rules.....do not cover deterioration which may occur in service as a result of radiation effects, corrosion, erosion or instability of the materials.
>
> It is this now somewhat illuminated blindspot which Intervenors seek to address.

The Staff has interpreted this contention as a claim, principally, that the NRC's projection of the amount of increase in reference temperature RT_{NDT}, which results from neutron irradiation damage, is nonconservative, that the amount of reactor material degradation for the Catawba reactor vessels cannot be accurately measured, and, as a result, that a safety hazard therefore exists.

3. The Staff's initial projection (as stated in its Safety Evaluation for the Catawba OL) of the amount of increase in RT_{NDT} resulting from neutron irradiation damage for the Catawba reactor vessel

is based on the method presented in Regulatory Guide 1.99, Rev. 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." This projection method was considered conservative because the trend curves in the Regulatory Guide form an upper bound for the data which was available at the time of its issuance.

4. The conservative nature of the Regulatory Guide is demonstrated by comparing its projection method with the test results from the Oconee Reactor Vessel Surveillance Program and the projection method in Commission Report SECY-82-465, "Pressurized Thermal Shock." The Staff in its safety evaluation in a letter from E. L. Conner to H. B. Tucker dated March 11, 1983 (attached) compared the change in RT_{NDT} for the materials in the Oconee Reactor Vessel Surveillance Program to the projection method of Regulatory Guide 1.99. This report indicates that the actual increase in RT_{NDT} for the Oconee Capsule OC III-B and OC II-A weld metals is 89°F and 104°F, respectively (Table 1 of Staff's Safety Evaluation in E. L. Conner letter). For these capsule materials the Regulatory Guide 1.99 method would project an increase in RT_{NDT} of 170°F and 226°F, respectively. Contrary to Intervenors' concern that there has been "unanticipated 'rapid increase in reference temperature' . . . " at Oconee, the increase has been well below that which was predicted by the methods used for this purpose. With respect to Oconee, therefore, the increase in RT_{NDT} projected by the Regulatory Guide is conservative, inasmuch as it is approximately twice the increase in RT_{NDT} of the irradiated Oconee Reactor Vessel Surveillance material.

5. In Commission Report Secy-82-465, the Staff statistically evaluated the increase in RT_{NDT} resulting from irradiation damage from

all PWR reactor vessel surveillance materials. The source of this data is 25 operating PWR plant surveillance programs. The range of neutron fluence represented by the data is 8×10^{17} n/cm^2 to 8×10^{19} n/cm^2 (E 1 MeV). Since the end-of-life fluence for the Catawba reactor vessels will be within this range, the conclusions reached in the study are valid for Catawba for the life of the plant.

6. The Commission evaluation resulted in the "Guthrie Formula" (Page E-6, Appendix E, SECY-82-465) which had a standard deviation of 24°F. For the limiting Catawba Units 1 and 2 reactor vessel beltline materials, the "Guthrie Formula" predicts at the end of life of the plant (40 years of operation) that the inside surface mean RT_{NDT} would increase by 62°F and 61.5°F respectively. The Regulatory Guide 1.99 method predicts for the Catawba Units 1 and 2 materials that the increase in RT_{NDT} would be 58°F and 94°F respectively. For Catawba Unit 1, the RT_{NDT} by the Regulatory Guide prediction method is within 4°F of the mean RT_{NDT} of the "Guthrie Formula" and for Catawba Unit 2, the RT_{NDT} by the Regulatory Guide prediction method is within the 95% confidence interval (two standard deviations) of the "Guthrie Formula." This shows that the Regulatory Guide and "Guthrie Formula" prediction methods are consistent and are conservative with respect to the Oconee surveillance test data. Thus the staff considers these projection methods conservative for predicting the increase in RT_{NDT}, which results from neutron irradiation damage.

7. In addition to the prediction methods previously discussed, the Commission requires that all commercially operated reactor vessels comply with the requirements of Appendix H, 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements." This program

requires that samples from the limiting reactor vessel materials be placed inside reactor vessel surveillance capsules which are irradiated within the reactor vessel. According to the withdrawal schedule referenced in this Appendix, the capsules must be withdrawn and the materials must be tested to determine the amount of reactor vessel material embrittlement resulting from neutron irradiation damage. For the Catawba reactor vessels, for which it has been conservatively predicted that the end of service adjusted reference temperature will not exceed 200°F, the capsule withdrawal schedule is set forth in ASTM E 185, Section 7.6.2. The results of these tests will be used to determine the actual increase in RT_{NDT} for the Catawba reactor vessels. Thus the Staff considers that the combination of reference temperature increase prediction methods previously discussed and the Applicants' reactor vessel surveillance program will accurately determine the amount of reactor material degradation for the Catawba reactor vessel materials.

8. The Staff ensures safe operation of the reactor vessel during normal, anticipated upset and test conditions by requiring the vessel to be operated within the limits of Appendix G, 10 CFR Part 50, "Fracture Toughness Requirements." According to this Appendix, the RT_{NDT} for the limiting reactor vessel material is the basis for the reactor vessel operating limits. The Staff will compare the results of the surveillance program with the Staff projection methods (i.e., "Guthrie," or Reg. Guide 1.99) and will use the higher RT_{NDT} for calculating operating limit curves. Since the Catawba reactor vessel materials will have their RT_{NDT} accurately determined throughout the life of the plant, the Staff considers that the reactor vessels can be safely operated

during normal, anticipated upset and test conditions. In addition, the operating curves for normal and anticipated upset conditions are calculated using a safety factor of two on the pressure which will account for variance in physical parameters, such as weld chemistry.

9. The Staff ensures safe operation of the reactor vessel during faulted and emergency conditions by requiring the vessel RT_{NDT} to comply with the screening criteria of Commission Report SECY-82-465, "Pressurized Thermal Shock." This report on page 6 states that "the risk from PTS events for reactor vessels with RT_{NDT} values less than the proposed screening criterion (270°F for axial welds and 300°F for circumferential welds) is acceptable."

10. The projected end of life RT_{NDT} for the Catawba Units 1 and 2 reactor vessels are identified in Table I. The Staff considers that the 95% confidence RT_{NDT} represents the upper bound RT_{NDT} for the Catawba Units 1 and 2 reactor vessels, but that the RT_{NDT} projected by the Regulatory Guide 1.99 method and the mean RT_{NDT} projected by the "Guthrie Formula" represent the amount of embrittlement at end-of-life expected for the Catawba reactor vessels. The upper bound 95% confidence RT_{NDT} for the Catawba Units 1 and 2 reactor vessels are 162°F and 124.5°F, respectively. These values are well below the PTS screening criteria and indicate that the risk to the vessel during faulted and emergency conditions is acceptable.

11. The Staff therefore believes that the amount of reactor material degradation for the Catawba reactor vessels can be accurately measured and that the methods used to predict such degradation are conservative. Since Appendix G requires the vessel operating limits to

be based upon the values so determined for the limiting vessel material, there is reasonable assurance that the Catawba reactor vessels can and will be operated well within acceptable safety margins for material degradation.

Barry J. Elliot

Barry J. Elliot

Subscribed and sworn to before me
this 1st day of July 1983

Patricia Ficchetti
Notary Public

My commission expires: 7/1/84

Table I

Projected End-of-Life Reference Temperature, RT_{NDT}

	RT_{NDT} by Reg. Guide 1.99 Method	Mean RT_{NDT} by "Guthrie Formula"	Upper Bound 95% Confidence RT_{NDT} by "Guthrie Formula"
Catawba Unit 1	110°F	114 °F	162 °F
Catawba Unit 2	109°F	76.5°F	124.5°F

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

March [illegible], 1983

Dockets Nos. 50-269, 50-270
and 50-287

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

The Commission has issued the enclosed Amendments Nos. 119 , 119, and 116 to Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments consist of changes to the Station's common Technical Specifications (TSs) in response to your request dated November 12, 1982, as supplemented on February 24, 1983.

These amendments revise the TSs concerning the heatup, cooldown and inservice test limitations for the reactor coolant systems of each Oconee unit.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Eben L. Conner

Eben L. Conner, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 119 to DPR-38
2. Amendment No. 119 to DPR-47
3. Amendment No. 116 to DPR-55
4. Safety Evaluation
5. Notice

cc w/enclosures:
See next page

cc w/enclosure(s):

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DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 119
License No. DPR-38

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Duke Power Company (the licensee) dated November 12, 1982, as supplemented February 24, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility Operating License No. DPR-38 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 119, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment becomes effective on March 14, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: MAR 11 1983

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 119
License No. DPR-47

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Duke Power Company (the licensee) dated November 12, 1982, as supplemented February 24, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility Operating License No. DPR-47 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 119, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment becomes effective on March 14, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: MAR 11 1983

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT NO.3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 116
License No. DPR-55

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Duke Power Company (the licensee) dated November 12, 1982, as supplemented February 24, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility Operating License No. DPR-55 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 116, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment becomes effective on March 14, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: MAR 11 1983

3.1.2 Pressurization, Heatup, and Cooldown Limitation

Specification

3.1.2.1 The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited to follows:

Heatup:

Heatup rates and allowable combinations of pressure and temperature shall be limited in accordance with Table 3.1-1 and Figure
3.1.2-1A Unit 1
3.1.2-1B Unit 2
3.1.2-1C Unit 3

Cooldown:

Cooldown rates and allowable combinations of pressure and temperature shall be limited in accordance with Table 3.1-2 and Figure
3.1.2-2A Unit 1
3.1.2-2B Unit 2
3.1.2-2C Unit 3

3.1.2.2 Leak tests required by Specification 4.3 and ASME Section XI shall be limited to the heatup and cooldown rates and allowable combinations of pressure and temperature provided in Tables 3.1-1, 3.1-2 and Figure 3.1.2-3A Unit 1
3.1.2-3B Unit 2
3.1.2-3C Unit 3

3.1.2.3 For thermal steady state system hydro tests required by ASME Section XI the system may be pressurized to the limits set forth in Specification 2.2 and 3.1.2.2.

3.1.2.4 The secondary side of the steam generator shall not be pressurized above 237 psig if the temperature of the vessel shell is below 110°F.

3.1.2.5 The pressurizer heatup and cooldown rates shall not exceed 100°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 410°F.

3.1.2.6 Prior to exceeding fifteen (Unit 1)
fifteen (Unit 2)
fifteen (Unit 3)

effective full power years of operation.

Figures 3.1.2-1A (Unit 1), 3.1.2-2A (Unit 1)
3.1.2-1B (Unit 2), 3.1.2-2B (Unit 2)
3.1.2-1C (Unit 3), 3.1.2-2C (Unit 3)

and 3.1.2-3A (Unit 1)
3.1.2-3B (Unit 2)
3.1.2-3C (Unit 3)

and Technical Specification 3.1.2.1, 3.1.2.2 and 3.1.2.3 shall be updated for the next service period in accordance with 10 CFR 50, Appendix G, Section V.B and V.E.

3.1.2.7 The updated proposed technical specification referred to in 3.1.2.6 shall be submitted for NRC review at least 90 days prior to the end of the service period for Units 1, 2 and 3.

limitations of 110°F and 227 psig are based on the highest estimated RT_{NDT} of +40°F and the preoperational system hydrostatic test pressure of [illegible] psig. The average metal temperature is assumed to be equal to or greater than the coolant temperature. The limitations include margins of 25 psi and 10°F for possible instrument error.

The spray temperature difference is imposed to maintain the thermal stresses at the pressurized spary line nozzle below the design limit.

REFERENCES

(1) Analysis of Capsule OCII-A from Duke Power Company Oconee Unit 2 Reactor Vessel Materials Surveillance Program, BAW-1699, December 1981.

(2) Analysis of Capsule OCIII-B from Duke Power Company Oconee Unit 3 Reactor Vessel Materials Surveillance Program, BAW-1697, October 1981.

(3) Analysis of Capsule OCI-E from Duke Power Company Oconee Unit 1 Reactor Vessel Materials Surveillance Program, BAW-1436, September, 1977.

TABLE 3.1-1

OPERATIONAL GUIDANCE FOR PLANT HEATUP

I. RC Temperature Constraints

RC Temperature	Maximum Heatup Rate
T < 280°F	50°F/HR
T > 280°F	100°F/HR

II. RC Pump Constraints

None

TABLE 3.1-2

OPERATIONAL GUIDANCE FOR PLANT COOLDOWN

I. RC Temperature Constraints

RC Temperature(1)	Maximum Cooldown Rate(2)
T > 280°F	≤ 50°F in any ½ hour period
150°F < T < 280°F	≤ 25°F in any ½ hour period
T < 150°F	≤ 10°F in any 1 hour period
RCS depressurized(3)	≤ 50°F in any 1 hour period

(1) RC temperature is cold leg temperature if one or more RC pumps are in operation or if on natural circulation cooldown; otherwise it is the LPI cooler outlet temperature.

(2) These rate limits must be applied to the change in temperature indication from cold leg temperature to LPI cooler outlet temperature per Note (1).

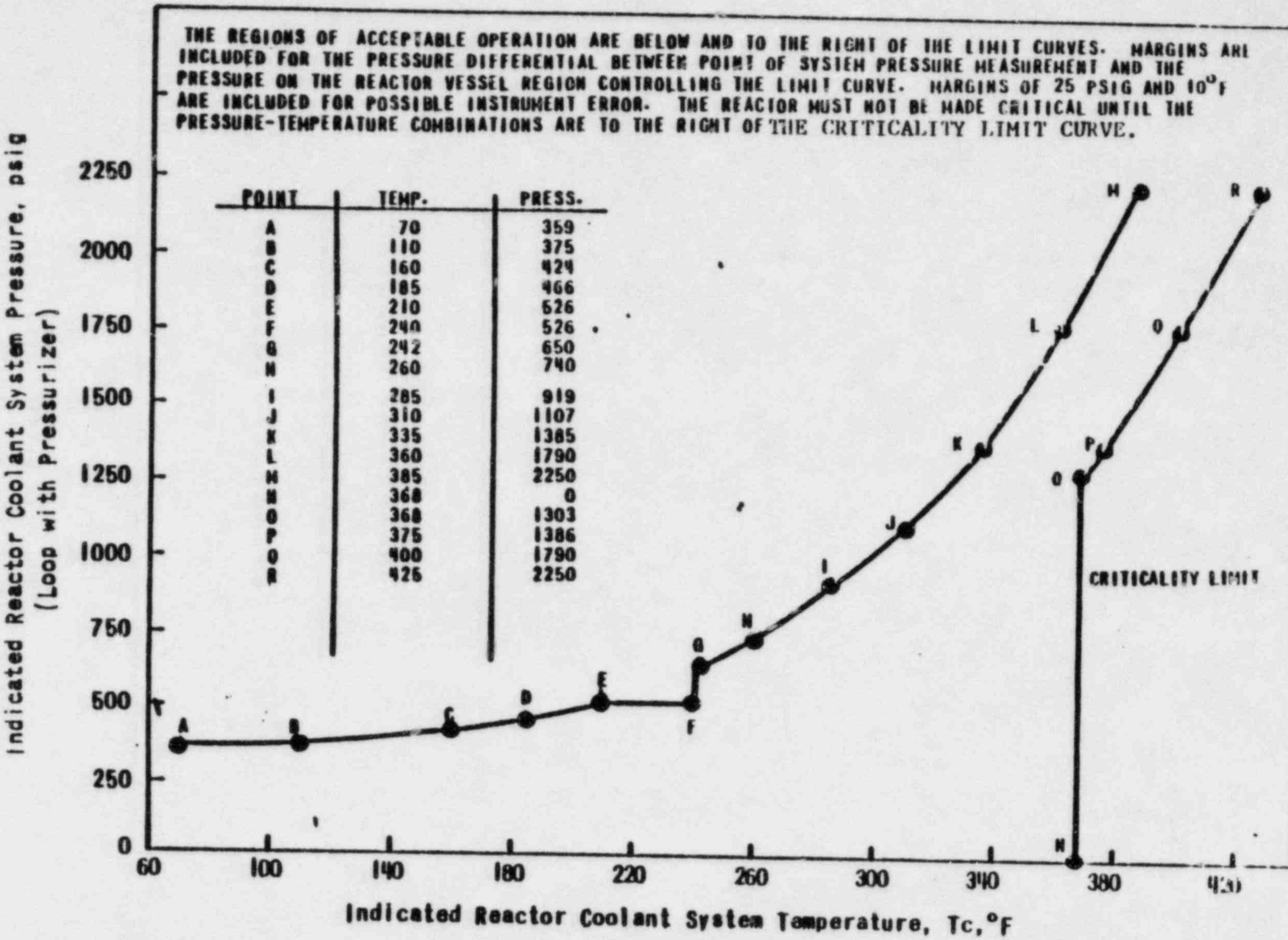
(3) When the RCS is depressurized such that all three of the following conditions exist:

a) RCS temperature < 200°F,
b) RCS pressure < 50 psig,
c) All RC Pumps off,

the maximum cooldown rate shall be relaxed to ≤ 50°F in any 1 hour period.

II. RC Pump Constraints For Validity of Guidance

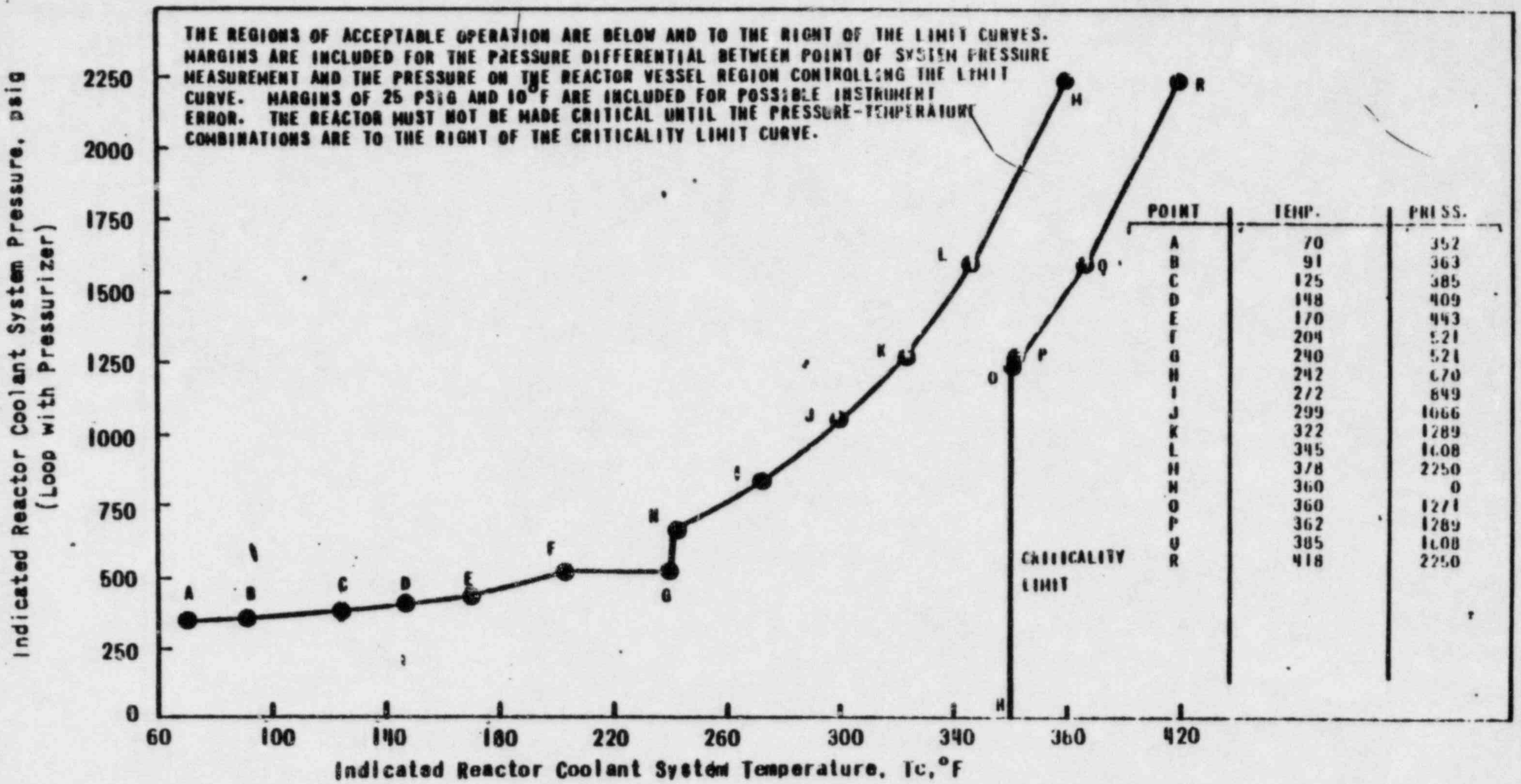
RC Temperature	Allowed Pump Combinations
> 270°F	Any
270-200°F	No more than 1 pump per loop
< 200°F	No more than 1 pump



POINT	TEMP.	PRESS.
A	70	359
B	110	375
C	160	424
D	185	466
E	210	526
F	240	526
G	242	650
H	260	740
I	285	919
J	310	1107
K	335	1385
L	360	1790
M	385	2250
N	368	0
O	368	1303
P	375	1386
Q	400	1790
R	426	2250

UNIT 1 OCONEE NUCLEAR STATION
REACTOR COOLANT SYSTEM NORMAL OPERATION-HEATUP
LIMITATIONS APPLICABLE FOR THE FIRST 15 EFPY

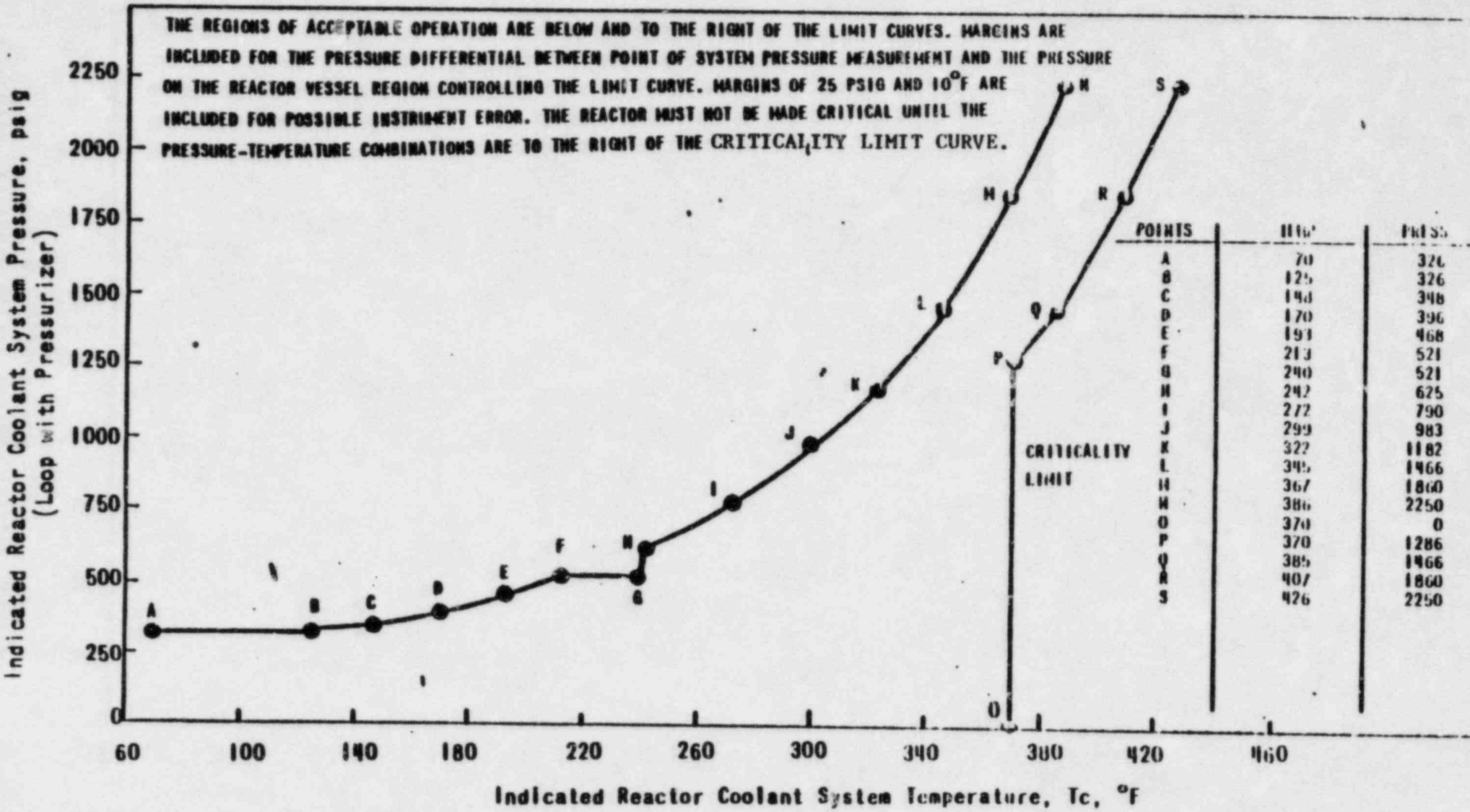
Figure 3.1.2-1A



POINT	TEMP.	PRESS.
A	70	352
B	91	363
C	125	385
D	148	409
E	170	443
F	204	521
G	240	521
H	242	670
I	272	849
J	299	1066
K	322	1289
L	345	1608
M	378	2250
N	360	0
O	360	1271
P	362	1289
Q	385	1608
R	418	2250

UNIT 2 OCONEE NUCLEAR STATION
REACTOR COOLANT SYSTEM NORMAL OPERATION-
HEATUP LIMITATIONS APPLICABLE FOR FIRST 15.6 EFPY

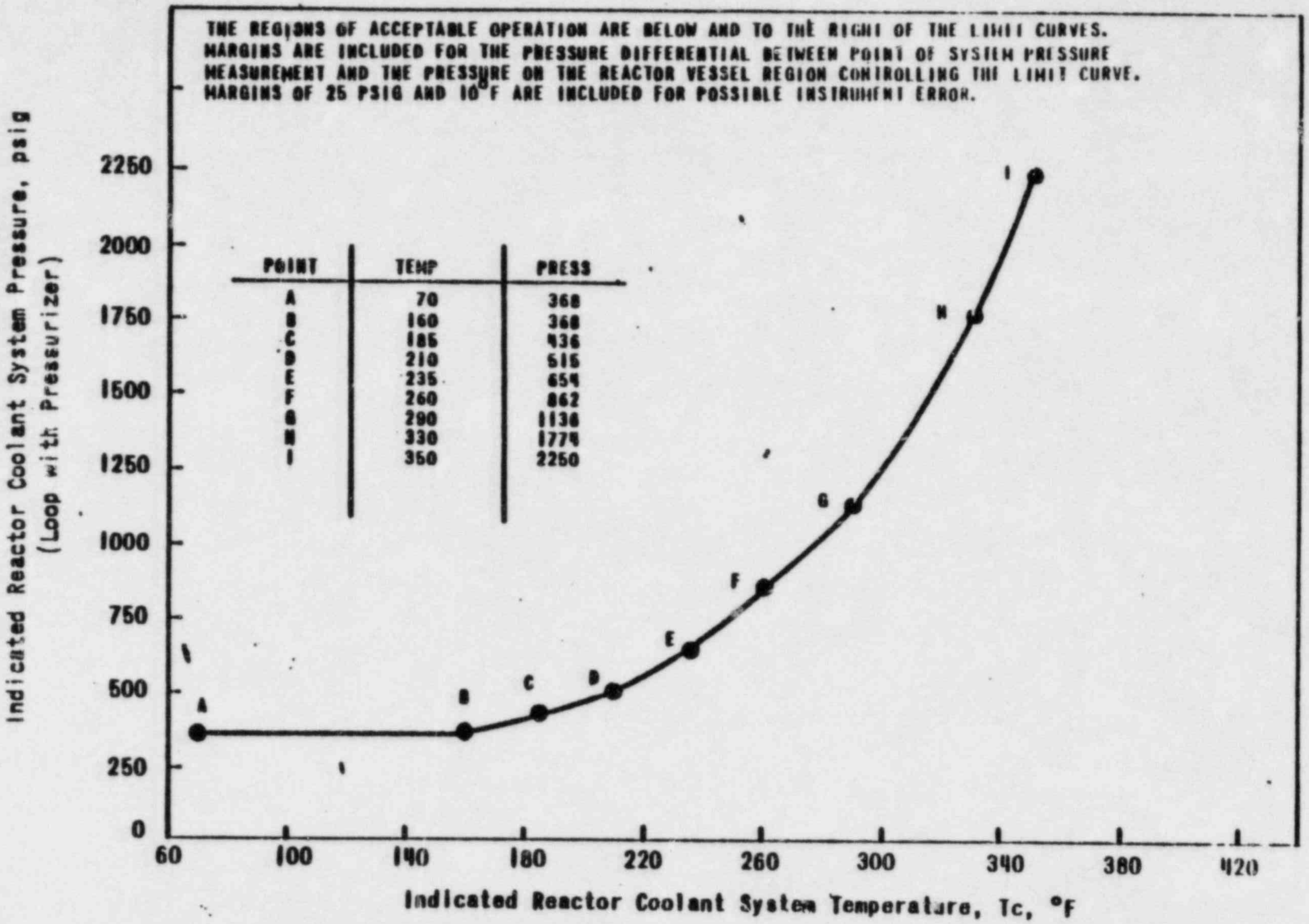
Figure 3.1.2-1B



POINTS	[illegible]	PRESS
A	70	326
B	125	326
C	148	348
D	170	396
E	193	468
F	213	521
G	240	521
H	242	625
I	272	790
J	299	983
K	322	1182
L	345	1466
M	367	1860
N	386	2250
O	370	0
P	370	1286
Q	385	1466
R	407	1860
S	426	2250

UNIT 3 OCONEE NUCLEAR STATION
REACTOR COOLANT SYSTEM NORMAL OPERATION-HEATUP
LIMITATIONS APPLICABLE FOR FIRST 15.0 EFPY

Figure 3.1.2-1C

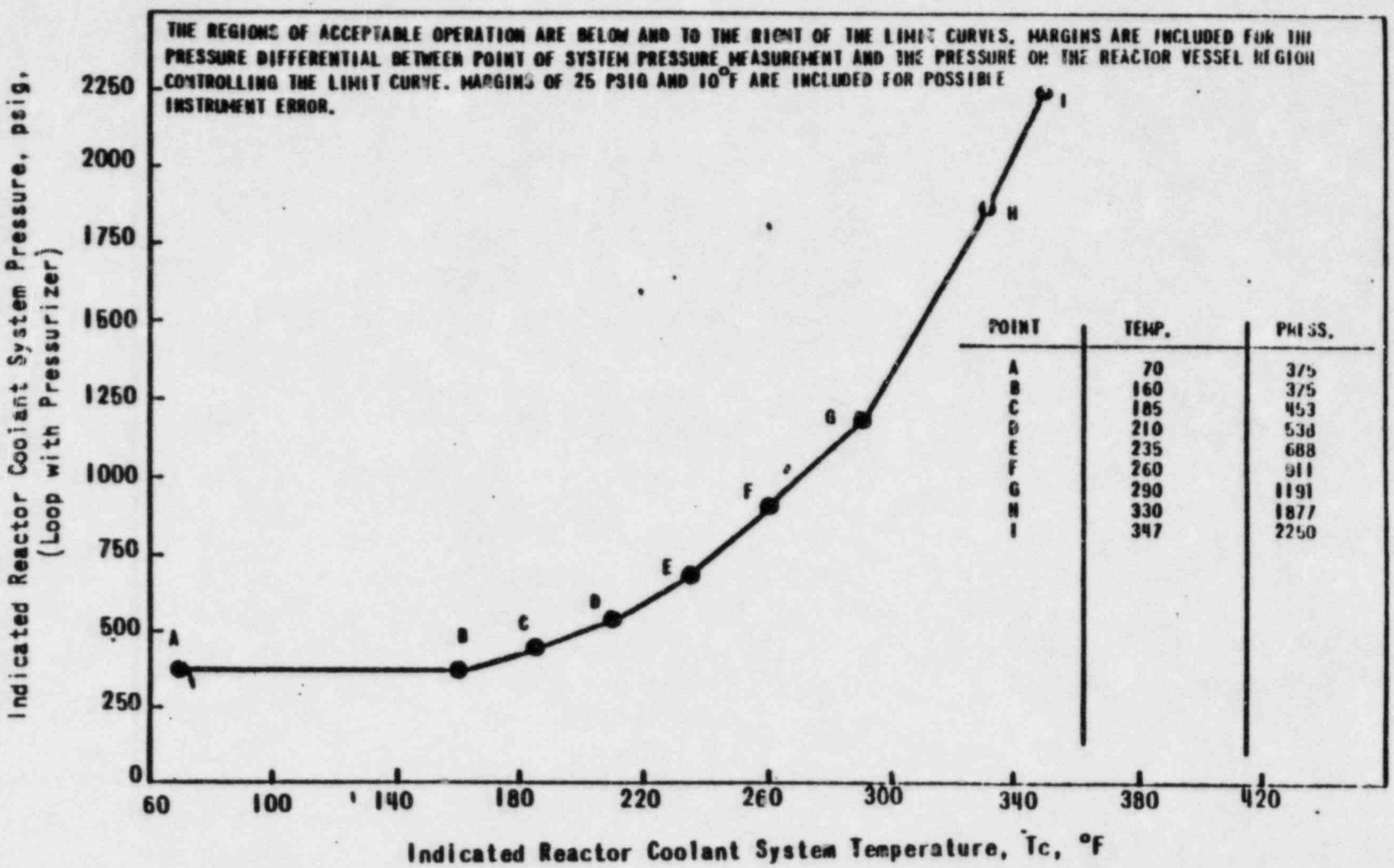


POINT	TEMP	PRESS
A	70	368
B	160	368
C	185	436
D	210	516
E	235	654
F	260	862
G	290	1136
H	330	1779
I	350	2250

Figure 3.1.2-2A

UNIT 1 OCONEE NUCLEAR STATION
REACTOR COOLANT SYSTEM NORMAL OPERATION-
COOLDOWN LIMITATIONS APPLICABLE FOR

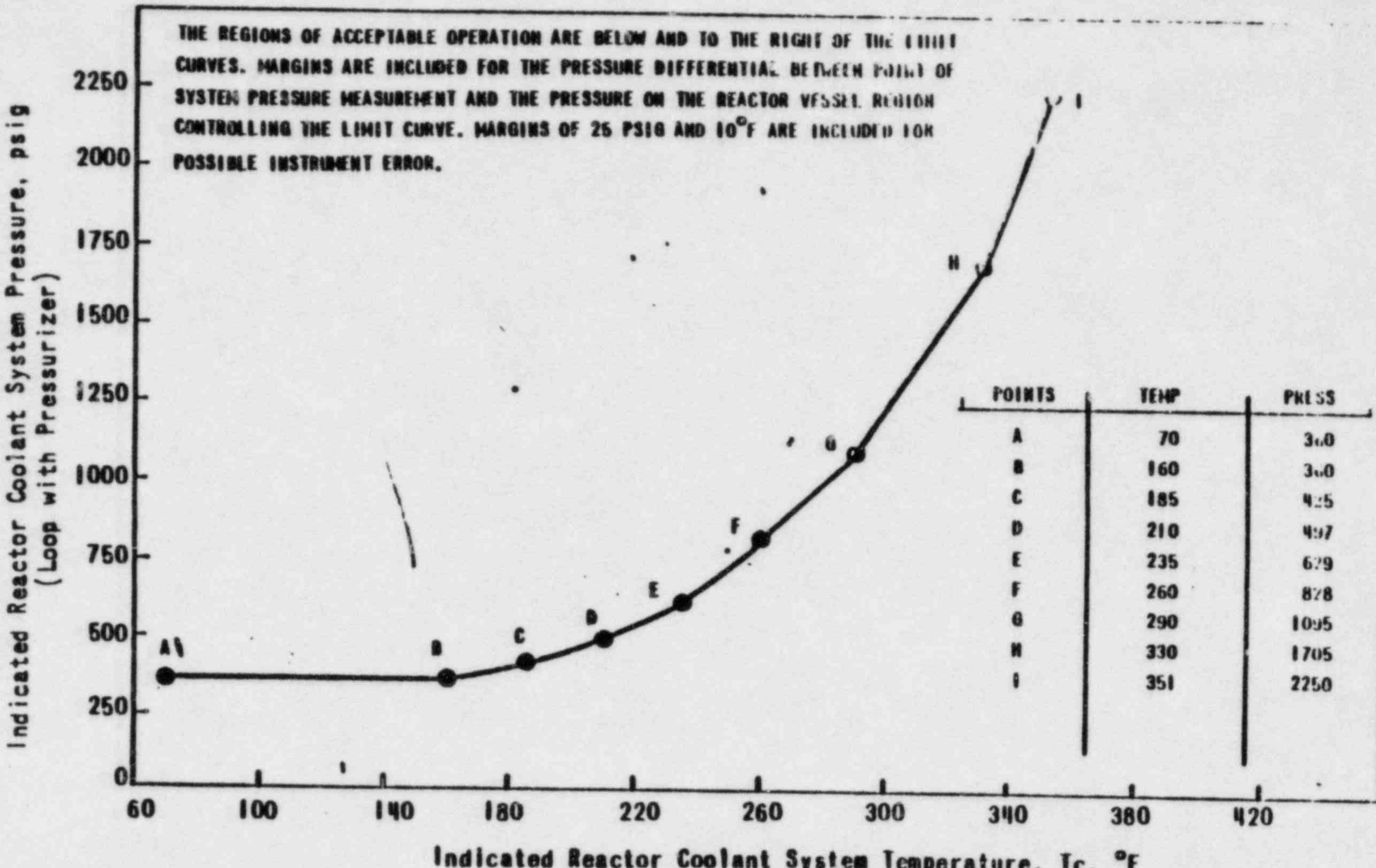
Amendments Nos. 119, 119, & 116



POINT	TEMP.	PRESS.
A	70	375
B	160	375
C	185	453
D	210	538
E	235	688
F	260	911
G	290	1191
H	330	1877
I	347	2250

UNIT 2 OCONEE NUCLEAR STATION
REACTOR COOLANT SYSTEM NORMAL OPERATION-COOLDOWN
LIMITATIONS APPLICABLE FOR FIRST 15.0 EFPY

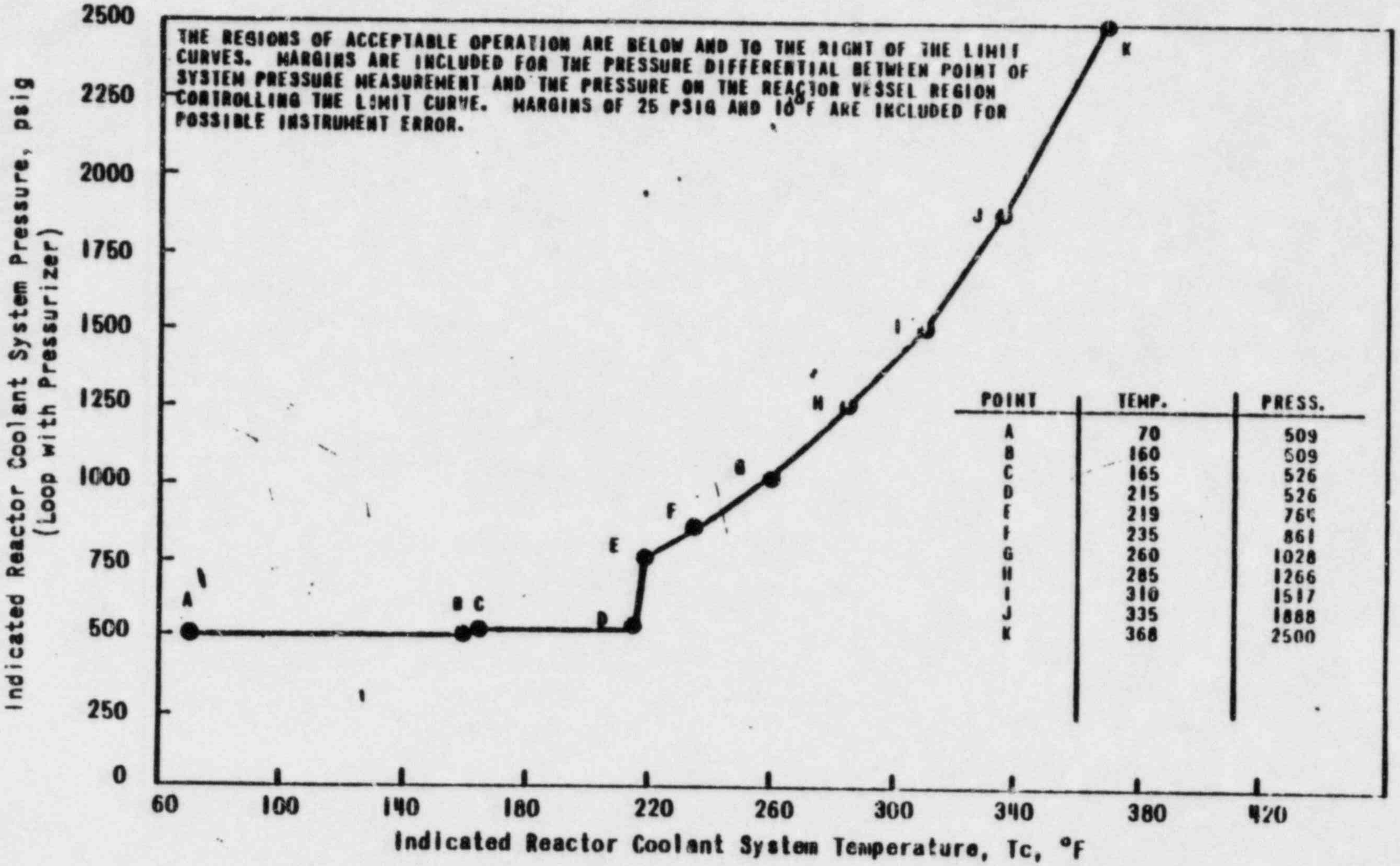
Figure 3.1.2-2B



POINTS	TEMP	PRESS
A	70	360
B	160	360
C	185	425
D	210	497
E	235	629
F	260	828
G	290	1095
H	330	1705
I	351	2250

UNIT 3 OCONEE NUCLEAR STATION
REACTOR COOLANT SYSTEM NORMAL OPERATION-COOLDOWN
LIMITATIONS APPLICABLE FOR FIRST 15.0 EFPY

Figure 3.1.2-2C

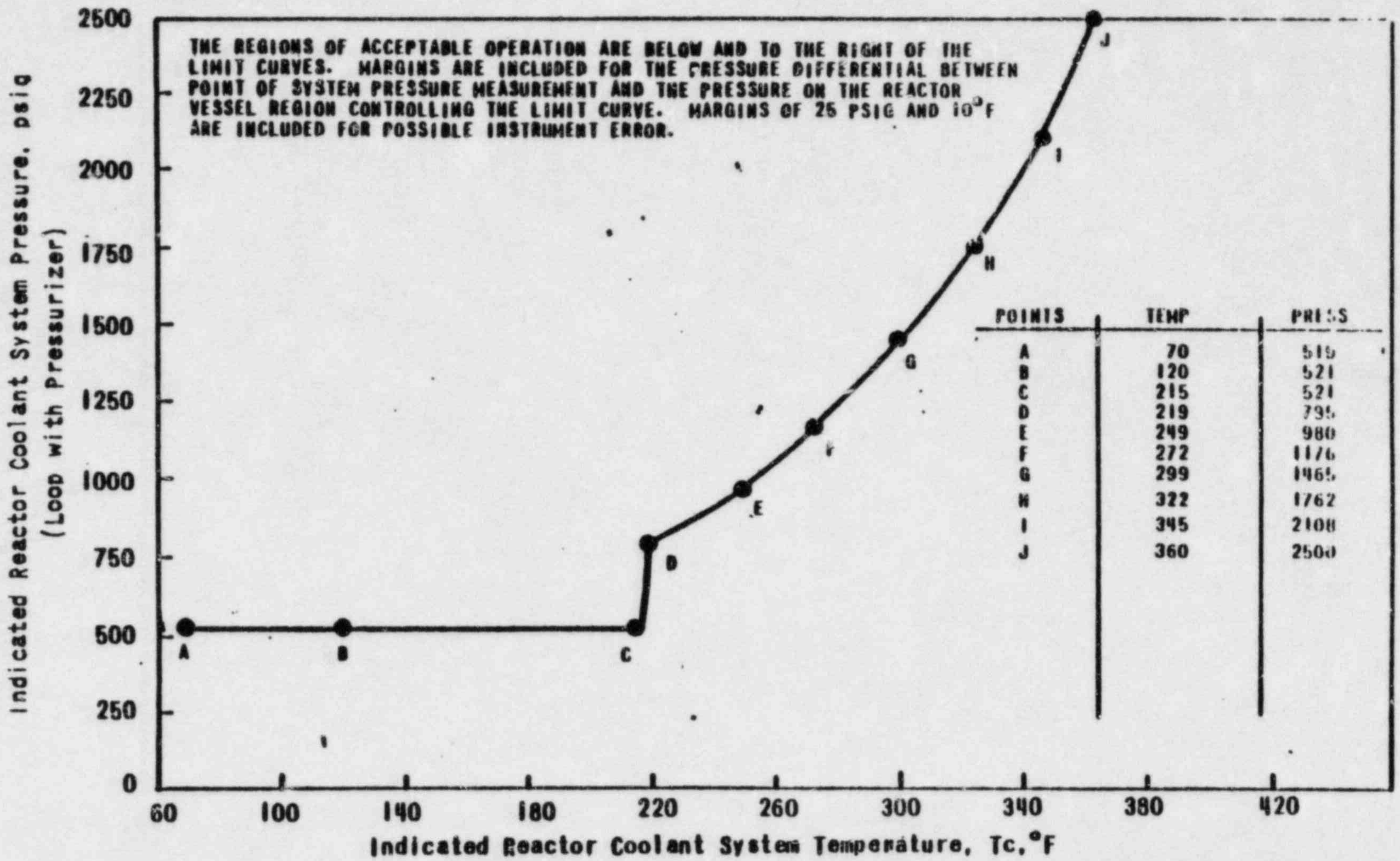


POINT	TEMP.	PRESS.
A	70	509
B	160	509
C	165	526
D	215	526
E	219	769
F	235	861
G	260	1028
H	285	1266
I	310	1517
J	335	1888
K	368	2500

UNIT ! OCONEE NUCLEAR STATION

REACTOR COOLANT SYSTEM INSERVICE LEAK AND HYDROSTATIC TEST HEATUP AND COOLDOWN LIMITATION

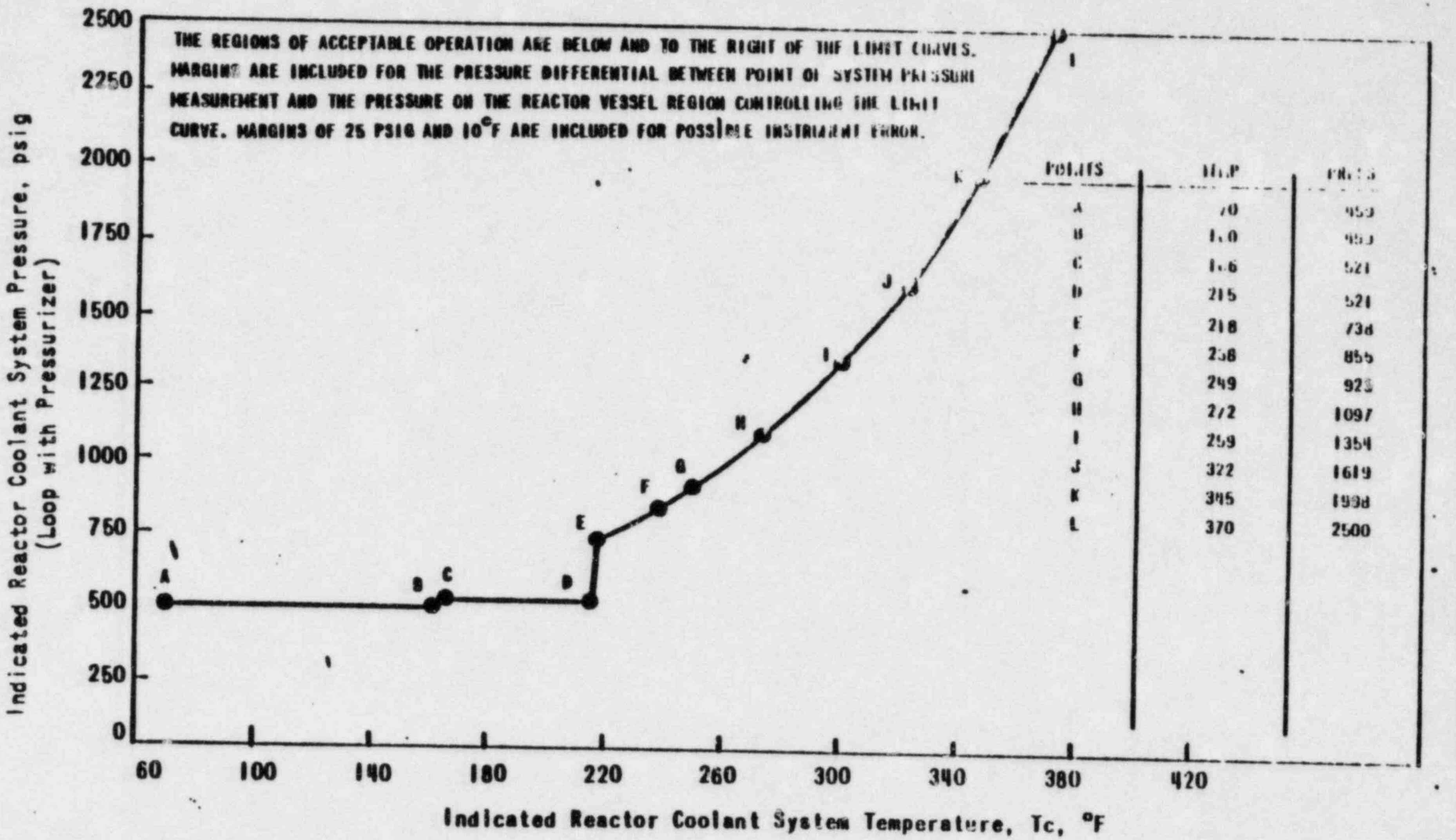
Figure 3.1.2-3A



POINTS	TEMP	PRESS
A	70	519
B	120	521
C	215	521
D	219	795
E	249	980
F	272	1176
G	299	1465
H	322	1762
I	345	2108
J	360	2500

UNIT 2 OCONEE NUCLEAR STATION
REACTOR COOLANT SYSTEM INSERVICE LEAK AND
HYDROSTATIC TEST HEATUP AND COOLDOWN
LIMITATIONS APPLICABLE FOR 15.0 EFPY

Figure 3.1.2-3B



POINTS	TEMP	PRESS
A	70	[illegible]
B	[illegible]	[illegible]
C	[illegible]	521
D	215	521
E	218	738
F	238	855
G	249	923
H	272	1097
I	299	1354
J	322	1619
K	345	1998
L	370	2500

UNIT 3 OCONEE NUCLEAR STATION
REACTOR COOLANT SYSTEM INSERVICE LEAK AND
HYDROSTATIC TEST HEATUP & COOLDOWN LIMITATIONS FOR FIRST
15.0 EFPY

Figure 3.1.2-3C

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 119 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 119 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NO. DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS NOS. 1, 2 AND 3

DOCKETS NOS. 50-269, 50-270 AND 50-287

Introduction

By letter dated November 12, 1982, as revised on February 24, 1983, Duke Power Company (DPC or the licensee) proposed a change to the Oconee Nuclear Station, Units 1, 2 and 3 Technical Specifications (TSs). This change is a revision to the reactor vessel pressure-temperature limits.

Background

The licensee indicated that the bases for the proposed pressure-temperature limits were the material properties data in Babcock & Wilcox (B&W) Reports BAW-1697 and BAW-1699. The curves for each Oconee reactor vessel are to be valid for 15 effective full power years (EFPY).

The B&W Reports BAW-1697 and BAW-1699 contain the B&W analysis of reactor vessel material surveillance capsules OC III-B and OC II-A, respectively. These capsules are part of the B&W Owners Group Integrated Surveillance Program. As a result, the capsules were irradiated in both the Oconee and Crystal River 3 reactor vessels.

Evaluation

A comparison of the materials in the Oconee 1, 2 and 3 reactor vessels and the OC III-B and OC II-A capsules indicates that the limiting weld material in the Oconee 1, 2 and 3 reactor vessels is not contained in the OC III-B and OC II-A capsules. The limiting material in the Oconee 1, 2 and 3 reactor vessels is weld material SA 1430, WF 24, and WF 67, respectively. The weld materials in OC III-B and OC II-A are WF 209-1B and WF 209-1A, respectively. Although the weld materials in the vessel and the capsules are not identical, they were prepared by the same manufacturer, using the same type of wire and flux and heat treated to an equivalent metallurgical condition. As a result, the fracture toughness data from capsules OC III-B and OC II-A may be utilized for evaluating the proposed pressure-temperature limits.

The change in upper shelf energy (USE) and reference temperature resulting from neutron irradiation damage of the limiting materials in the OC III-B and OC II-A capsules are compared in Table 1 to the values predicted by Regulatory Guide 1.99, Rev. 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials", and the values predicted by B&W Report BAW-1511P dated October 1980. This comparison indicates that the Regulatory Guide 1.99 method for predicting change in RT_{NDT} resulting from neutron irradiation damage is conservative. In addition, the method in Figure 13 of B&W Report BAW-1511P for predicting the change in weld material USE properties resulting from neutron irradiation damage is more accurate than the method in Regulatory Guide 1.99. Hence, we utilized Regulatory Guide 1.99 methodology for estimating the change in vessel material RT_{NDT}, and Figure 3 in B&W Report BAW-1511P for estimating the change in reactor vessel material USE. We believe that Figure 3 in B&W Report BAW-1511P is more accurate than Regulatory Guide 1.99 for estimating the change in USE resulting from irradiation damage for Oconee vessel and surveillance weld materials because Figure 3 in B&W Report BAW-1511P was generated from reactor vessel surveillance weld materials similar to the Oconee vessel and surveillance weld materials, and the Regulatory Guide 1.99 curve was generated from reactor vessel weld materials utilized throughout the nuclear industry.

The USE for the Oconee beltline materials must exceed 50 ft-lbs at the 1/4 thickness location in order to meet the safety margins required by Paragraph IV.A.2 of Appendix G, 10 CFR Part 50. Using Figure 3 in B&W Report BAW-1511P, we estimate that the limiting materials in Oconee 1, 2 and 3 reactor vessel beltlines will have USE less than 50 ft-lbs at the 1/4 thickness location when their neutron fluence (E>1MeV) exceeds $5 \times 10^{18} n/cm^2$, $4.8 \times 10^{18} n/cm^2$ and $7.5 \times 10^{18} n/cm^2$, respectively. Based on the neutron fluence estimated by the licensee for each beltline material and the uncertainty in vessel dosimetry identified by B&W*, we conclude that the USE energy at the 1/4 thickness location for the Oconee beltline reactor vessel materials will exceed 50 ft-lbs for the period of time that the proposed pressure-temperature curves are applicable.

Using the method for predicting shift in RT_{NDT} in Regulatory Guide 1.99, Rev. 1, the neutron fluence estimates of the licensee, the unirradiated material properties in B&W Reports BAW-1511P, October 1980, and BAW-10046P, March 1976, and the method of calculating pressure-temperature limits identified in Standard Review Plan Section 5.3.2, the proposed pressure-temperature limit curves for Oconee Units 1, 2 and 3 meet the safety margins of Appendix G, 10 CFR Part 50, and are acceptable for 15 EFPY.

*C. Whitmarsh, Draft B&W Report to be Published.

Table [illegible]

Comparison of Change in Properties for OC III-B and OC II-A Capsule Weld Materials

	Change in RT_{NDT} (°F)			Change in USE [illegible]		
	Capsule	Reg. 1.99	BAW-1511P	Capsule	Reg. 1.99	BAW-1511P
WF 209-1B	89	170	N/A	24	34*	24.5
WF-209-1A	104	226	N/A	28	36	27

* Estimated per Figure 3, page C-10 of B&W Report BAW-1511P, October 1980.

Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated, do not create the possibility of an accident of a type different from any evaluated previously, and do not involve a significant reduction in a margin of safety, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: MAR 11 1983

The following NRC personnel have contributed to this Safety Evaluation: L. Lois, B. Elliot, E. Conner.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-269, 50-270 AND 50-287

DUKE POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 119,119 and 116to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, issued to Duke Power Company, which revised the Technical Specifications (TSs) for operation of the Oconee Nuclear Station, Units Nos. 1, 2 and 3, located in Oconee County, South Carolina. The amendments become effective on March 14, 1983.

These amendments revise the TSs concerning the heatup, cooldown and inservice test limitations for the reactor coolant systems of each Oconee unit.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated November 12, 1982, as supplemented February 24, 1983, (2) Amendments Nos. 119, 119, and 116 to Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Oconee County Library, 501 West Southbroad Street, Walhalla, South Carolina 29691. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 11th day of March 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

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ATTACHMENTS TO LICENSE AMENDMENTS

AMENDMENT NO. 119TO DPR-38

AMENDMENT NO. 119TO DPR-47

AMENDMENT NO. 116TO DPR-55

DOCKETS NOS. 50-269, 50-270 AND 50-287

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment numbers and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
vi	vi
3.1-3	3.1-3
3.1-3a	3.1-3a
3.1-5	3.1-5
--	3.1-5a
--	3.1-5b
3.1-6	3.1-6
3.1-6a	3.1-6a
3.1-6b	3.1-6b
3.1-7	3.1-7
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3.1-7b	3.1-7b
3.1-7c	3.1-7c
3.1-7d	3.1-7d
3.1-7e	3.1-7e

STATEMENT OF PROFESSIONAL QUALIFICATIONS

BARRY J. ELLIOT
U.S. NUCLEAR REGULATORY COMMISSION
MATERIALS ENGINEERING BRANCH, DIVISION OF
ENGINEERING, NUCLEAR REACTOR REGULATION

I am currently employed as a Materials Engineer in the Materials Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation. I am responsible for review and evaluation of safety analysis reports which are related to the material engineering aspects of components in nuclear power plant systems, and for providing technical assistance to Offices of NRR, I&E and RES in related reactor safety matters. I have been a member of the Materials Engineering Branch since March 31, 1980.

I was a full-time student at Rensselaer Polytechnic Institute, where in 1968, I received a Bachelors Degree in Materials Engineering. I attended evening classes at Fairleigh Dickenson University, where in 1971, I received a Masters Degree in Business Administration.

I was employed by Curtiss Wright Corporation from 1968, when I graduated from Rensselaer Polytechnic Institute, to 1980, when I was employed by the NRC. From 1968 to 1971 I worked in the Materials Development Laboratory of the Aeronautical Division where I performed failure analysis of reciprocating and gas turbine engines, and developed test apparatus to evaluate material reliablity. From 1971 to 1980 I worked in the Nuclear Division where I was responsible for developing and implementing non-destructive examination test procedures and fusion weld procedures for inspection and fabrication of Navy Nuclear Pressure Vessels.

I am a member of the American Society of Metals.