PROPOSED CHANGE RTS-181 TO THE
DUANE ARNOLD ENERGY CENTER
TECHNICAL SPECIFICATIONS

The holders of license DPR-49 for the Duane Arnold Energy Center propose to amend Appendix A (Technical Specifications) to said license by deleting the current pages and replacing them with the attached, new pages.

Summary

This Technical Specification change proposal updates the pressure-temperature operating limits for the Duane Arnold Energy Center reactor vessel and makes the limits valid through 12 effective full power years. The operating limits were adjusted to account for the minor estimated changes in fracture toughness due to 6 effective full power years of neutron fluence on the vessel. This update is supported by an analysis performed by General Electric which accompanies this proposal.

Also, the minimum temperature for which the reactor vessel head bolting studs can be in tension is lowered from 100°F to 74°F. Calculations have shown that the former limit of 100°F was overly conservative when established and was much higher than necessary or practical. The correction still includes the appropriate safety margin which is also supported by the G.E. analysis and is in accordance with the requirements of 10 CFR 50 Appendix G.

The following list of proposed changes is in the order that the changes appear in the Technical Specifications. The List of Affected Pages is presented following this list of changes.

1) On page 3.6-1, paragraph 3.6.A.2 is revised to make Figure 3.6-1 valid through 12 effective full power years instead of 6.

2) On page 3.6-2, the minimum temperature for which bolting studs can be in tension is lowered from 100°F to 74°F. Calculations have shown that the former limit of 100°F was too conservative when established and was much higher than is necessary or practical.

3) On page 3.6-2, the first Surveillance Requirement paragraph is revised to state that test specimens were installed at the start of operation.

4) On page 3.6-2, the second Surveillance Requirement paragraph is revised to state that samples shall be withdrawn at 6 and 15 effective full power years.

5) On page 3.6-2, the second Surveillance Requirement paragraph is revised to state that neutron flux wires were installed in the reactor vessel and then removed and tested during the second refueling outage.

6) On page 3.6-16, the first sentence of the bases is changed to state that 10 CFR 50, Appendix G was revised in May 1983.

7) On page 3.6-16, reference number (2) is added to describe the fracture toughness analysis performed by General Electric to support this evaluation. This reference will be added under the References section on page 3.6-41.

8) On page 3.6-16, the reference to the FSAR is revised to reference the Updated FSAR and the new appendix.

9) On pages 3.6-17 and 3.6-18, the paragraph which discusses the NDT temperature is replaced by two paragraphs which discuss the derivation of and the influences on the NDT curves and the minimum operating temperature of the closure flange region.

10) On page 3.6-49, the NDT curves of Figure 3.6-1 are revised to be valid through 12 effective full power years.

LIST OF AFFECTED PAGES

3.6-1
3.6-2
3.6-16
3.6-17
3.6-18
3.6-41
3.6-49

LIMITING CONDITION FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

Specification:

A. Thermal and Pressurization Limitations

1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a one-hour period.

2. The reactor vessel shall be vented and power operation shall not be conducted unless the reactor vessel temperature is equal to or greater than that shown in Curve C of Figure 3.6-1. Operation for hydrostatic or leakage tests, during heatup or cooldown, and with the core critical shall be conducted only when vessel temperature is equal to or above that shown in the appropriate curve of Figure 3.6-1. Figure 3.6-1 is effective through 12 effective full power years. At least six months prior to 12 effective full power years new curves will be submitted.

SURVEILLANCE REQUIREMENT

4.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the periodic examination and testing requirements for the reactor cooling system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Thermal and Pressurization Limitations

1. During heatups and cooldowns, the following temperatures shall be logged at least every 15 minutes until 3 consecutive readings at each given location are within 5°F.

   a. Reactor vessel shell adjacent to shell flange.

   b. Reactor vessel bottom drain.

   c. Recirculation loops A and B.

   d. Reactor vessel bottom head temperature.

2. Reactor vessel metal temperature at the outside surface of the bottom head in the vicinity of the control rod drive housing and reactor vessel shell adjacent to shell flange shall be recorded at least every 15 minutes during inservice hydrostatic or leak testing when the vessel pressure is >312 psig.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head is greater than 74°F.	Test specimens of the reactor vessel base, weld and heat affected zone metal subjected to the highest fluence of greater than 1 MeV neutrons were installed in the reactor vessel adjacent to the vessel wall at the core midplane level at the start of operation. The specimens and sample program shall conform to ASTM E 185-66 to the degree discussed in the FSAR. Samples shall be withdrawn at 6 and 15 effective full power years in accordance with 10 CFR 50, Appendix H. Neutron flux wires were installed in the reactor vessel adjacent to the reactor vessel wall at the core midplane level. The wires were removed and tested during the second refueling outage to experimentally verify the calculated values of neutron fluence at one-fourth of the beltline shell thickness that are used to determine the NDTT shift. Results of the flux wire test and the effects of copper and phosphorus on the beltline are reflected in Figure 3.6-1.
4. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.	3. When the reactor vessel head bolting studs are tensioned and the reactor is in a Cold Condition, the reactor vessel shell temperature immediately below the head flang shall be permanently recorded.
5. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.	4. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged. 5. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

3.6.A and 4.6.A BASES:

Thermal and Pressurization Limitations

The thermal limitations for the reactor vessel meet the requirements of 10 CFR 50, Appendix G, revised May 1983. (2)

The allowable rate of heatup and cooldown for the reactor vessel contained fluid is 100°F per hour averaged over a period of one hour. This rate has been chosen based on past experience with operating power plants. The associated time period for heatup and cooldown cycles when the 100°F per hour rate is limiting provides for efficient, but safe, plant operation.

Specific analyses were made based on a heating and cooling rate of 100°F/hour applied continuously over a temperature range of 100°F to 546°F. Calculated stresses were within ASME Boiler and Pressure Vessel Code Section III stress intensity and fatigue limits even at the flange area where maximum stress occurs.

Chicago Bridge and Iron Company performed detailed stress analysis as shown in the Updated FSAR Appendix 5A, "Site Assembly of the Reactor Vessel." This analysis includes more severe thermal conditions than those which would be encountered during normal heating and cooling operations.

The permissible flange to adjacent shell temperature differential of 145°F is the maximum calculated for 100°F hour heating and cooling rate applied continuously over a 100°F to

550°F range. The differential is due to the sluggish temperature response to the flange metal and its value decreases for any lower heating rate or the same rate applied over a narrower range.

The coolant in the bottom of the vessel is at a lower temperature than that in the upper regions of the vessel when there is no recirculation flow. This colder water is forced up when recirculation pumps are started. This will not result in stresses which exceed ASME Boiler and Pressure Vessel Code, Section III limits when the temperature differential is not greater than 145°F.

The reactor coolant system is a primary barrier against the release of fission products to the environs. In order to provide assurance that this barrier is maintained at a high degree of integrity, restrictions have been placed on the operating conditions to which it can be subjected.

The operating limits in Figure 3.6-1 are derived in accordance with 10 CFR 50 Appendix G, May 1983 and Appendix G of the ASME Code. Conditions in three regions influence the curves: the closure flange region, the non-beltline region which includes most nozzles and discontinuities, and the beltline region which is irradiated with fluence above 10^{17} n/cm^2 during the vessel operating life. Irradiation causes an increase in the nil-ductility temperature (RT_{NDT}) of the beltline materials, possibly to the point where the beltline region impacts the pressure-temperature limits for the vessel. However, for Figure 3.6-1, effective to 12 EFPY, the beltline which has an

RT_{NDT} of 40°F is less limiting than the non-beltline regions which generally experience higher stresses at nozzles and discontinuities. The limiting RT_{NDT} of 58°F for the Standby Liquid Control Nozzle (N10) is the highest RT_{NDT} of any component in the non-beltline region.

The closure flange region, with RT_{NDT} = 14°F, has a bolt preload and minimum operating temperature of 74°F. This exceeds original requirements of the ASME Code (Winter 1967 Addendum) and provides extra margin relative to current ASME Code requirements.

Neutron flux wires and samples of vessel material are installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The wires and samples will be removed and tested according to 10 CFR 50 Appendix H. Results of these analyses will be used to adjust Figure 3.6-1 as appropriate.

As described in paragraph 4.2.5 of the Safety Analysis report, detailed stress analyses have been made on the reactor vessel for both steady state and transient conditions with respect to material fatigue. The results of these transients are compared to allowable stress limits. Requiring the coolant temperature in an idle recirculation loop to be within 50°F of the operating loop temperature before a recirculation pump is started assures that the changes in coolant temperature at the reactor vessel nozzles and bottom head region are acceptable.

The records will be developed from engineering data available. If actual installation data is not available, the service life will be assumed to commence with the initial criticality of the plant. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3.6 and 4.6 References

1) General Electric Company, Low-Low Set Relief Logic System and Lower MSIV Water Level Trip for the Duane Arnold Energy Center, NEDE-30021-P, January, 1983.

2) General Electric Company, Duane Arnold Energy Center Reactor Pressure Vessel Fracture Toughness Analysis to 10 CFR 50 Appendix G, May 1983, NEDC-30839, December, 1984.

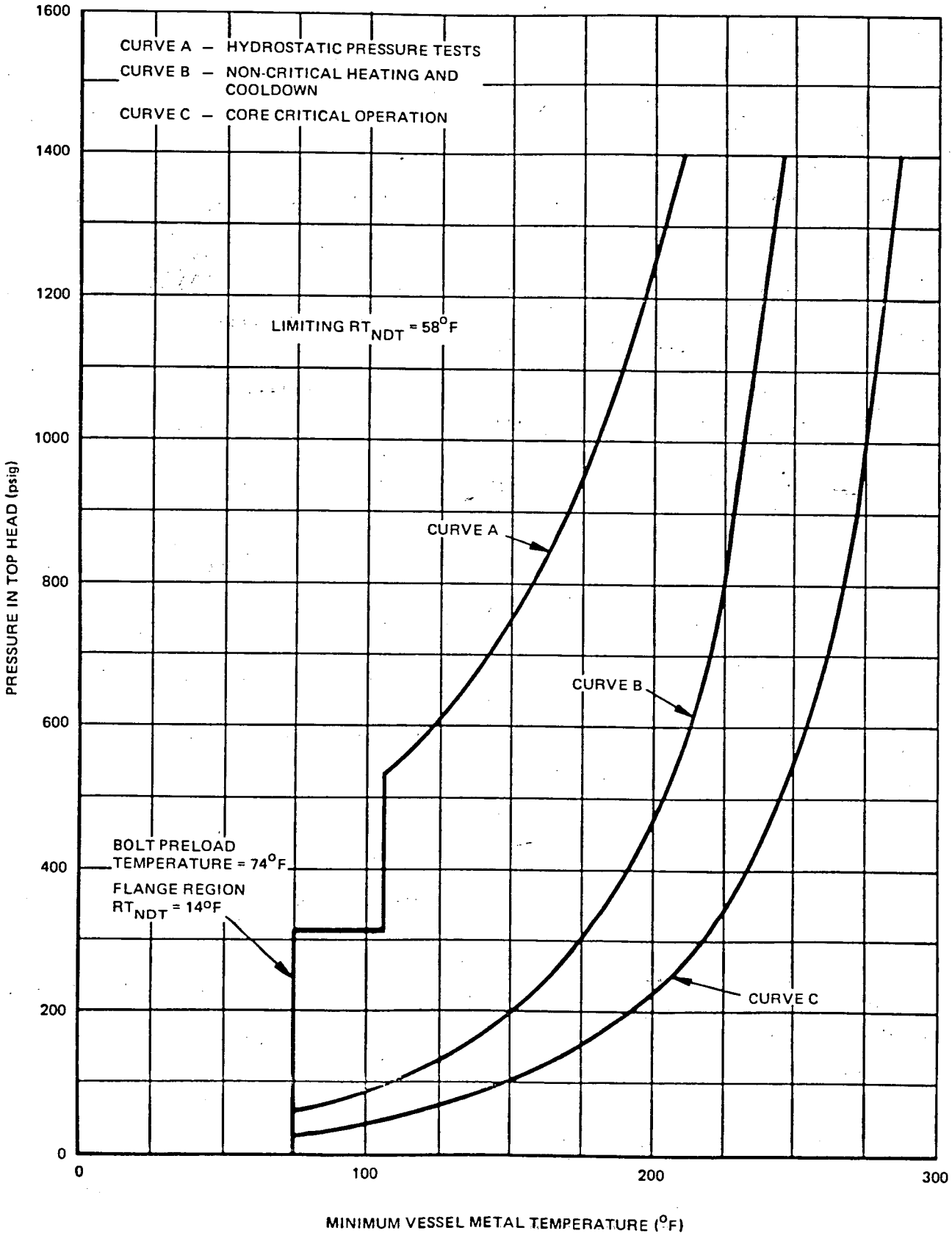

Figure 3.6-1. Pressure versus Minimum Temperature Valid to Twelve Full Power Years, per Appendix G of 10CFR50

Iowa Electric Light and Power Company
January 11, 1985
NG-85-0003

Mr. Harold Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
Technical Specification Changes (RTS-181)
Update of NDT Operating Limit Curves
File: A-117, J-40b

Dear Mr. Denton:

In accordance with the Code of Federal Regulations, Title 10, Parts-50.59 and 50.90, Iowa Electric Light and Power Company hereby requests revision to the Technical Specifications (TS) for the Duane Arnold Energy Center (DAEC).

This proposal adjusts the pressure-temperature operating limits for the Duane Arnold Energy Center reactor vessel and makes the limits valid through 12 effective full power years. This submittal is due 6 months prior to 6 effective full power years, which we presently anticipate to be July 15, 1985. This proposal also adjusts the vessel head bolting stud minimum temperature which was discovered to be higher than necessary and overly conservative. Both of these changes are supported by an analysis performed by General Electric, which is Attachment 3 to this proposal.

This application, proposed change RTS-181, has been reviewed by both our DAEC Operations Committee and DAEC Safety Committee. Per the fee schedule for license amendments (10 CFR 170), a check for $150 is enclosed. The balance of the application fee will be paid upon billing.

Pursuant to the requirements of 10 CFR 50.91, a copy of this submittal and analysis of no significant hazards considerations is being forwarded to our appointed state official.

This application, which consists of three signed originals and 37 copies with their enclosures, is true and accurate to the best of my knowledge and belief.

IOWA ELECTRIC LIGHT AND POWER COMPANY

BY Richard W. McGaughey
Richard W. McGaughey
Manager, Nuclear Division

Subscribed and sworn to Before Me on this 11th day of January 1985.

Kathleen M. Furman
Notary Public in and for the State of Iowa

RWM/MJM/ta*

Attachments:
1) Evaluation of Change Pursuant to 10 CFR 50.92
2) Proposed Change RTS-181 including List of Affected Pages
3) General Electric Analysis, NEDC-30839

cc: M. Murphy
L. Liu
S. Tuthill
M. Thadani
NRC Resident Office
T. Houvenagle (ICC)

EVALUATION OF CHANGE WITH RESPECT TO 10 CFR 50.92

Summary

This Technical Specification change proposal adjusts the pressure-temperature operating limits for the Duane Arnold Energy Center reactor vessel and makes the limits valid through 12 effective full power years. The operating limits were adjusted to account for the minor estimated changes in fracture toughness due to 6 effective full power years of neutron fluence on the vessel. This update is supported by an analysis performed by General Electric which accompanies this proposal.

Also, the minimum temperature for which the reactor vessel head bolting studs can be in tension is lowered from 100°F to 74°F. Calculations have shown that the former limit of 100°F was overly conservative when established and was much higher than necessary or practical. The correction still includes the appropriate safety margin which is also supported by the G.E. analysis and is in accordance with the requirements of 10 CFR 50 Appendix G.

In accordance with the requirements of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based upon the following information:

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response:

Neither the probability nor the consequences of an accident are increased since the operating limits are adjusted to incorporate the original fracture toughness conservatism present, over and above the safety margin, when the reactor vessel was new.

The bolting stud minimum temperature change is a correction of an overly conservative limit and does not increase the probability of occurrence nor the magnitude of the consequences of an accident.

(2) Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response:

The possibility of a new kind of accident is not created since the operating limits are merely being updated according to 10 CFR 50 Appendix G and no physical changes are being made.

The correction of the bolting stud minimum temperature is an adjustment to the limit and does not create a different type of accident.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response:

The margin of safety for the reactor pressure vessel is not being affected. The design margin over and above the margin of safety is actually being restored to a level similar to when the reactor vessel was new and the fracture toughness was slightly greater.

The correction of the bolting stud minimum temperature still includes an adequate margin of safety since the former limit was established too high and was overly conservative.

In the April 6, 1983 Federal Register, the NRC published a list of examples of amendments that are not likely to involve a significant hazards concern. Example number six of that list applies to the changing of the bolting stud minimum temperature and states:

"A change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan...: for example, a change resulting from the application of a small refinement of a previously used calculational model or design method."

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Specification:

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LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
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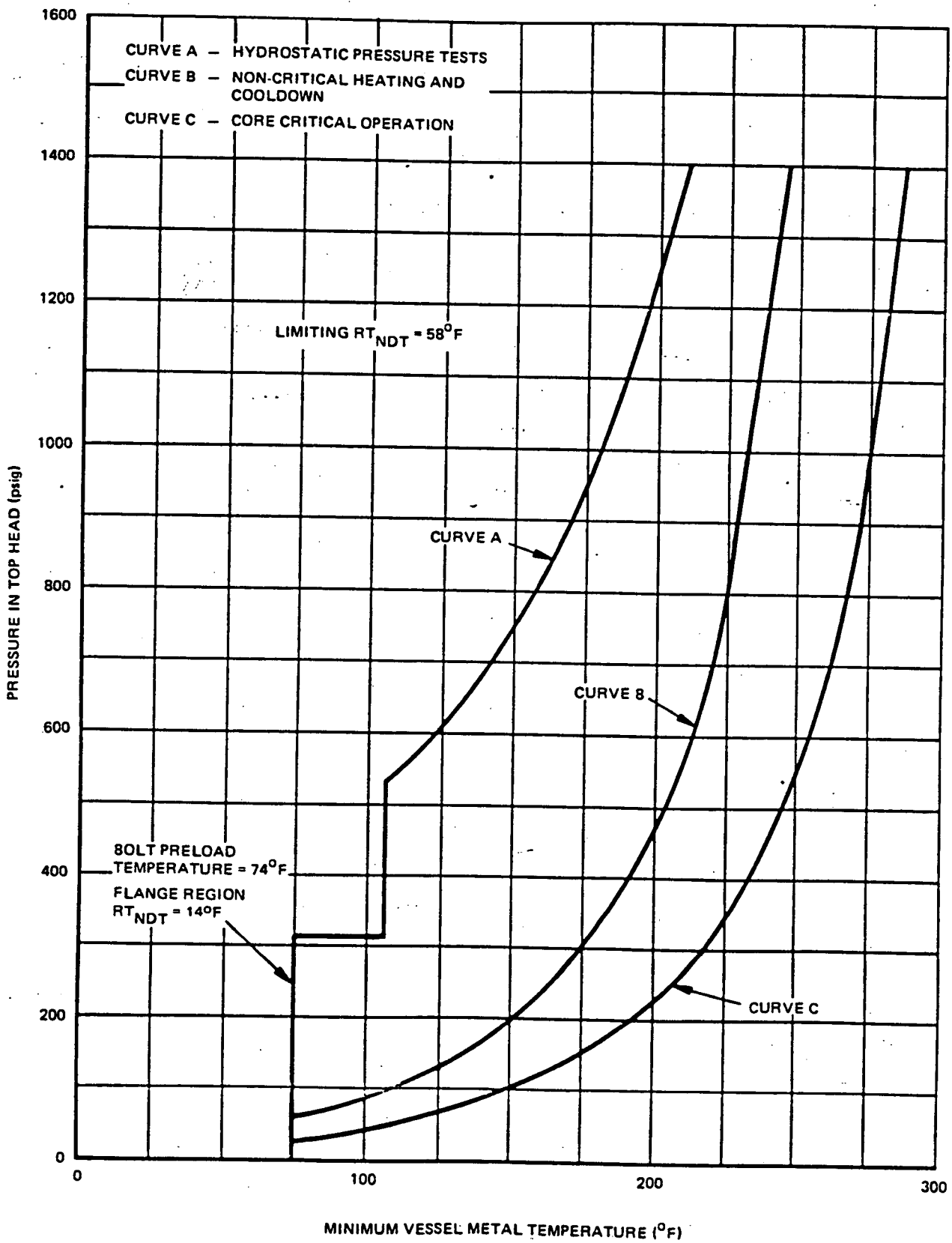

Figure 3.6-1. Pressure versus Minimum Temperature Valid to Twelve Full Power Years, per Appendix G of 10CFR50