NRC FORM 195
(2-76)

U.S. NUCLEAR REGULATORY COMMISSION

NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL

DOCKET NUMBER	50-331
FILE NUMBER	
DATE OF DOCUMENT	5-21-76
DATE RECEIVED	5-26-76
NUMBER OF COPIES RECEIVED	3 signed 37 CC

TO: Mr. B.C. Rusche

FROM: Iowa Elec. Light & Power Co.
Cedar Rapids, Iowa
Lee Liu

☒ LETTER ☒ NOTORIZED
☒ ORIGINAL ☒ UNCLASSIFIED
☐ COPY

PROP

INPUT FORM

DESCRIPTION Ltr notarized 5-21-76 requests for amdt to Tech Specs & trans the following:

ENCLOSURE Proposed changes to DAEC Tech Specs..

(40 cys encl rec'd)

PLANT NAME: Duane Arnold Plant

Do Not Remove

ACKNOWLEDGED

FOR ACTION/INFORMATION

DHL 5-26-76

	ASSIGNED AD:			ASSIGNED AD:
✓	BRANCH CHIEF: (6)	LEAR		BRANCH CHIEF:
✓	PROJECT MANAGER:	PAULSON		PROJECT MANAGER:
✓	LIC. ASST.:	PARRISH		LIC. ASST.:

INTERNAL DISTRIBUTION

✓	REG FILE		SYSTEMS SAFETY		PLANT SYSTEMS		ENVIRO TECH
✓	NRC PDR		HEINEMAN		TEDESCO		ERNST
✓	I & E (2)		SCHROEDER		BENAROYA		BALLARD
✓	OELD				LAINAS		SPANGLER
✓	GOSSICK & STAFF		ENGINEERING		IPPOLITO		
	MIPC		MCCARY				SITE TECH
	CASE		KNIGHT		OPERATING REACTORS		GAMMILL
	HANAUER		SIHWEIL		STELLO		STEPP
	HARLESS		PAWLICKI				HULMAN
					OPERATING TECH		
	PROJECT MANAGEMENT		REACTOR SAFETY	✓	EISENHUT		SITE ANALYSIS
	BOYD		ROSS	✓	SHAO		VOLLMER
	P COLLINS		NOVAK	✓	BAER		BUNCH
	HOUSTON		ROSZTOCZY	✓	SCHWENCER	✓	J. COLLINS
	PETERSON		CHECK	✓	GRIMES		KREGER
	MELTZ						
	HELTEMES		AT & I		SITE SAFETY & ENVIRO		
	SKOVHOLT		SALTZMAN		ANALYSIS		
			RUTBERG		DENTON & MULLER		

EXTERNAL DISTRIBUTION

✓	LPDR: Cedar Rapids	NATL LAB	BROOKHAVEN NATL LAB	CONTROL NUMBER
✓	TIC	REG. V-IE	ULRIKSON (ORNL)	
✓	NSIC	LA PDR		5291
	ASLB	CONSULTANTS		
✓	ACRS 16 [redacted] /SENT TO L.A.			

IOWA ELECTRIC LIGHT AND POWER COMPANY

General Office

CEDAR RAPIDS, IOWA

LEE LIU
VICE PRESIDENT – ENGINEERING

May 21, 1976

50-331

Mr. B. C. Rusche, Director
Office of Nuclear Reactor Regulation
Nuclear Regulatory Commission
Washington, D.C. 20545

Dear Mr. Rusche:

Transmitted herewith, in accordance with the requirements of 10CFR50.59 and 50.90, is an application for amendment of DPR-49 to incorporate proposed changes in Technical Specifications (Appendix A to License) for the Duane Arnold Energy Center (DAEC), described in the enclosure hereto.

This proposed change has been reviewed and approved by the DAEC Operations Committee and the DAEC Safety Committee and does not involve a significant hazards consideration.

Three signed and notarized originals and 37 additional copies of this application are transmitted herewith. This application, consisting of the foregoing letter and enclosures hereto, is true and accurate to the best of my knowledge and belief.

Iowa Electric Light and Power Company

By ______________________
Lee Liu
Vice President, Engineering

LL:D
cc: D. Arnold
J. Newman
J. Shea
J. Keppler

Sworn and subscribed to before me on this 21st day of May, 1976.

Jean R. Smith
Notary Public in and for the State of Iowa.

Jean R. Smith
NOTARY PUBLIC
STATE OF IOWA
Commission Expires
September 30, 1978

5291

PROPOSED CHANGE RTS-66A TO DAEC TECHNICAL SPECIFICATIONS

I. Affected Technical Specifications

Appendix A of the Technical Specifications for the DAEC (DPR-49) provides as follows:

Specification 3.6.A provides pressure and temperature limits for the reactor coolant system.

II. Proposed Changes in Technical Specifications

The licensees of DPR-49 propose the following changes in the Technical Specifications set forth in I above:

Delete the present Specification 3.6.A and 4.6.A (Limiting Conditions for Operations, Surveillance Requirements and Bases for Thermal and Pressurization Limitations) including Table 3.6-1 and Figure 3.6-1.

Replace with the attached Specification 3.6.A and 4.6.A, including Figures 3.6-1, 3.6-2A, 3.6-2B and 3.6-2C.

III. Justification for Proposed Change

This proposed change is being submitted in response to a request from the NRC (Letter; Mr. G. Lear, Chief, Operating Reactors Branch #3, Division of Operating Reactors, United States Nuclear Regulatory Commission to Mr. D. Arnold, President, Iowa Electric Light and Power Company; dated February 17, 1976).

The subject letter requested that Iowa Electric Light and Power Company review the reactor coolant system pressure limits contained in the Technical Specifications to determine if they are in full compliance with 10 CFR 50, Appendix G. The following comments apply to the DAEC Reactor Pressure Vessel.

The ferritic pressure boundary material of the Duane Arnold RPV was qualified by impact testing in accordance with the 1965 Edition of the ASME Boiler and Pressure Vessel Code, Section III, with Addenda to and including the Summer 1967 Addenda. From an operational standpoint, this Code requires that for any significant pressurization (taken to be more than 20 percent of Code hydrostatic test pressure, i.e., 312 psig) the minimum metal temperature of all vessel shell and head material must be at least 100°F (nil ductility transition temperature (NDTT) + 60°F).

A major condition necessary for full compliance with Appendix G is satisfaction of the requirements of the Summer 1972 Addenda to the ASME Boiler and Pressure Vessel Code, Section III. This is not possible with components which were purchased to earlier Code requirements in that appropriate material specimens are not available for the additional testing required. The last paragraph on Page 19013 of the July 17, 1973 Federal Register stated:

> "Although the requirements of Appendices G and H become effective on August 16, 1973, the Commission recognizes that there may be an interim period when, for plants now under construction, the method of compliance with certain provisions may be determined on a case-by-case basis. For example, if the test data needed to establish certain fracture control requirements are not available because they were not required at the time material sampling was done, estimated values that are appropriately conservative may be acceptable."

On this basis, the following is proposed for the DAEC:

The intent of the special method of compliance with Section III of the ASME Boiler and Pressure Vessel Code, Appendix G, for the RPV is to provide operating limitations on pressure and temperature based on fracture toughness. These operating limits should ensure that a margin of safety against a nonductile failure of the RPV will be very nearly the same as a vessel built to the Summer 1972 Addenda. Also, it should be noted that this vessel must also be operated within the requirements of the original applicable Code. The specific temperature limits for operation when the core is critical stated in 10 CFR 50, Appendix G, Paragraph IV.A.2.C, will be adhered to based on an analysis of the vessel heads and shell areas remote from discontinuities.

Operating limits on RPV pressure and temperature during normal heatup and cooldown, and during inservice hydrostatic testing, were established using Appendix G to the ASME Boiler and Pressure Vessel Code, Section III, 1971 (Appendix G was first added in the Summer 1972 Addenda) as a guide (see Figures 3.6-2A, B and C of the proposed Technical Specifications). These operating limits will ensure that a large postulated surface flaw, having a depth of one-quarter of the material thickness, can be safely accommodated in regions of the RPV shell remote from discontinuities.

For the purpose of setting these operating limits, the reference temperature, RT_{NDT}, was determined from the impact test data taken in accordance with requirements of the Code to which this RPV was designed and manufactured. There are two shell courses in the beltline region. Most of the active fuel zone is within the #2 shell; this shell course extends from about 40 inches above the top of the fuel to 37 inches above the bottom of the fuel. The #1 shell extends downward from 37 inches above the bottom of the active fuel zone. From dropweight test results, the base metal NDT temperature of the #1 shell (consisting of two plates) is +30°F or less while the actual base metal NDT temperature of the #2 ring was determined by dropweight tests to be -30°F. The NDTT used for the weld metal of shell #1 and #2 is +10°F which is the maximum specified value. One set of beltline operating limit curves were made with an assumed RT_{NDT} of +30°F representing the initial properties of the #1 shell and another set of curves were made with an assumed RT_{NDT} of +10°F to represent the #2 shell. While the initial curves representing the #1 shell will be more limiting than those representing the #2 shell, the #1 shell will see a lower neutron induced NDTT shift. From dosimetry data opposite the center of the core, the basic relationship between thermal power and neutron fluence can be established. This maximum fluence measured near the inside of the shell should be extrapolated to a distance of 1/4 of the way through the #2 shell (1.12 inches) and used to periodically adjust the curves which represent the #2 shell. A conservative ratio should be developed to account for the lesser fluence at the upper end of the #1 shell; then, this lesser fluence extrapolated to 1/4 of the #1 shell's thickness (1.26 inches) then this value used to periodically adjust the operating limits representing the #1 shell.

General Electric recommendations for making this periodic adjustment include the application of the shift curve, Figure 2, from NEDO-10115 (Figure 3.6-1 of the proposed Technical Specifications), Upper Limit for 550°F GE BWR Operating Experience. Until the estimated neutron fluence at the 1/4 thickness points discussed above reaches 3×10^{17}, there should be no need to make any adjustments to the beltline operating limit curves.

Another set of operating limits were calculated for the areas remote from the beltline region. These curves are based on an assumed RT_{NDT} of +40°F as that is the maximum allowed NDT temperature for any of the ferritic pressure boundary material in the vessel.

For a given pressure and a particular operating condition category (pressure testing, non-nuclear heatup or cooldown, or core operation) the most limiting of the two beltline curve temperatures adjusted for shift of the temperature from the curve representing areas remote from the beltline should be used as the minimal metal temperature. The curves for non-nuclear heatup or cooldown following core shutdown were calculated with an assumed constant through-wall temperature difference equal to the maximum value created by continuous heating at 100°F per hour to the normal operating temperature. The curves for pressure testing and non-nuclear heatup include vertical lines at the appropriate temperatures representing the NDTT +60°F limit for significant pressurization required by the applicable Code for the vessel manufacture. The curves for core operation were made from the non-nuclear heatup or cooldown curves with an additional 40°F margin as required by 10 CFR 50, Appendix G. Also, in accordance with Appendix G, the curves for core operation show by vertical lines that no core operation is permitted below the minimum temperature for an 1100 psig ISI hydrostatic test. The enclosed proposed Technical Specifications illustrate the use of the operating limits.

In addition to the proposed Technical Specifications to conform to Appendix G, we have modified the withdrawal requirements of the test specimens to conform to Appendix H.

IV. Review Procedures

This proposed change has been reviewed by the DAEC Operations Committee and Safety Committee which have found that this proposed change does not involve a significant hazards consideration.

TECHNICAL SPECIFICATIONS

LIST OF FIGURES

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

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 shell temperatures during inservice hydrostatic or leak testing shall be at or above the highest of the temperatures shown on the curves of Figure 3.6.2A where the RPV shell beltline region curves are increased by the expected shift in RT_{NDT} from Figure 3.6-1.

During heatup by non-nuclear means, cooldown following nuclear shutdown or low level

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 CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
physics tests the reactor vessel shell and fluid temperatures of 4.6.A.1 shall be at or above the highest of the temperatures of Figure 3.6-2B where the RPV beltline curves are increased by the expected shift in RT_{NDT} from Figure 3.6-1. During all operation with a critical core, other than for low level physics tests, the reactor vessel shell and fluid temperatures of 4.6.A.1 shall be at or above the highest of the temperatures of Figure 3.6-2C where the RPV beltline curves are increased by the expected shift in RT_{NDT} from Figure 3.6-1.	Test specimens of the reactor vessel base, weld and heat affected zone metal subjected to the highest fluence of greater than 1 MeV neutrons shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The specimens and sample program shall conform to ASTM E 185-66 to the degree discussed in the FSAR. Samples shall be withdrawn at one-fourth and three-fourths service life in accordance with 10CFR50, Appendix H. Neutron flux wires shall be installed in the reactor vessel adjacent to the reactor vessel wall at the core midplane level. The wires shall be 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 from Figure 3.6-1.
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 100°F.	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 flange shall be permanently recorded.
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.	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. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.	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 & 4.6.A BASES:

Reactor Coolant Heatup and Cooldown

The vessel has been analyzed for stresses caused by thermal and pressure transients. Heating and cooling transients throughout plant life at uniform rates of 100°F per hour were considered in the temperature range of 100 to 546°F and were shown to be within the requirements for stress intensity and fatigue limits of Section III of the ASME Boiler and Pressure Vessel Code (1971 Edition including Summer 1972 Addenda).

Reactor Vessel Temperature and Pressure

Operating limits on the reactor vessel pressure and temperature during normal heatup and cooldown, and during inservice hydrostatic testing, were established using Appendix G of the Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition, as a guide. These operating limits assure that a large postulated surface flaw, having a depth of one-quarter of the material thickness, can be safely accommodated in regions of the vessel shell remote from discontinuities. For the purpose of setting these operating limits, the reference temperature, RT_{NDT}, of the vessel material was estimated from impact test data taken in accordance with requirements of the Code to which this vessel was designed and manufactured (1965 Edition including Summer 1967 Addenda). Where the dropweight NDT temperature was known, the reference temperature used was the NDT temperature. Where the dropweight NDT temperature was not known, the reference temperature used was the temperature at which 30 ft.lb. of energy was expected to occur on the basis of reported Charpy V notch test data. For areas of the vessel shell remote from the core beltline region, the highest NDTT permissible by the vessel purchase specification for any vessel pressure boundary material is +40°F and this value is used for the RT_{NDT} in lieu of certified test results.

The fracture toughness of all ferritic steels gradually and uniformly decreases with exposure to fast neutrons above a threshold value, and it is prudent and conservative to account for this in the operation of the RPV. Two types of information are needed in this analysis: a) A relationship between the change in fracture toughness of the RPV steel and the neutron fluence (integrated neutron flux), and b) a measure of the neutron fluence at the point of interest in the RPV wall.

A relationship between neutron fluence and change in Charpy V, 30 foot pound transition temperature has been developed for SA302B/SA533 steel based on at least 35 experimental data points as shown in Figure 3.6-1. In turn, this change in transition temperature can be related to a change in the temperature ordinate shown in Figure G 2110-1 in Appendix G of Section III of the Boiler Code.

The neutron fluence at any point in the pressure vessel wall can be computed from core physics data. The neutron fluence can also be measured experimentally on the ID of the vessel wall. At present valid experimental measurements can be made only over time periods of less than 5 years because of the limitations of the dosimeter materials. This causes no problem because of the exact relationship between thermal power produced and the number of neutrons produced from a given core geometry. A single experimental measurement in a time period of two years can be used to predict the fluence for the life of the plant in terms of thermal power output if no great changes in core geometry are made.

The vessel pressurization temperatures at any time period can be determined from the thermal power output of the plant and its relation to the neutron fluence as shown on Figure 3.6-1 and Figures 3.6-2A, B or C as appropriate. During the first two fuel cycles only calculated neutron fluence values can be used. At the second refueling, neutron dosimeter wires which are installed adjacent to the vessel wall will be removed to verify the calculated neutron fluence. As more experience is gained in calculating the fluence, the need to verify it experimentally will disappear. Because of the many experimental points used to derive Figure 3.6-1, there is no need to reverify it for technical reasons, but in case verification is required for other reasons, three sets of mechanical test specimens representing the base metal, weld metal and weld heat affected zone metal have been placed in the vessel.

There are two shell courses in the beltline region. Most of the active fuel is within the #2 shell; this shell course extends from about 40 inches above the top of the fuel to 37 inches above the bottom of the fuel. The #1 shell extends downward from 37 inches above the bottom of the active fuel zone.

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 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.

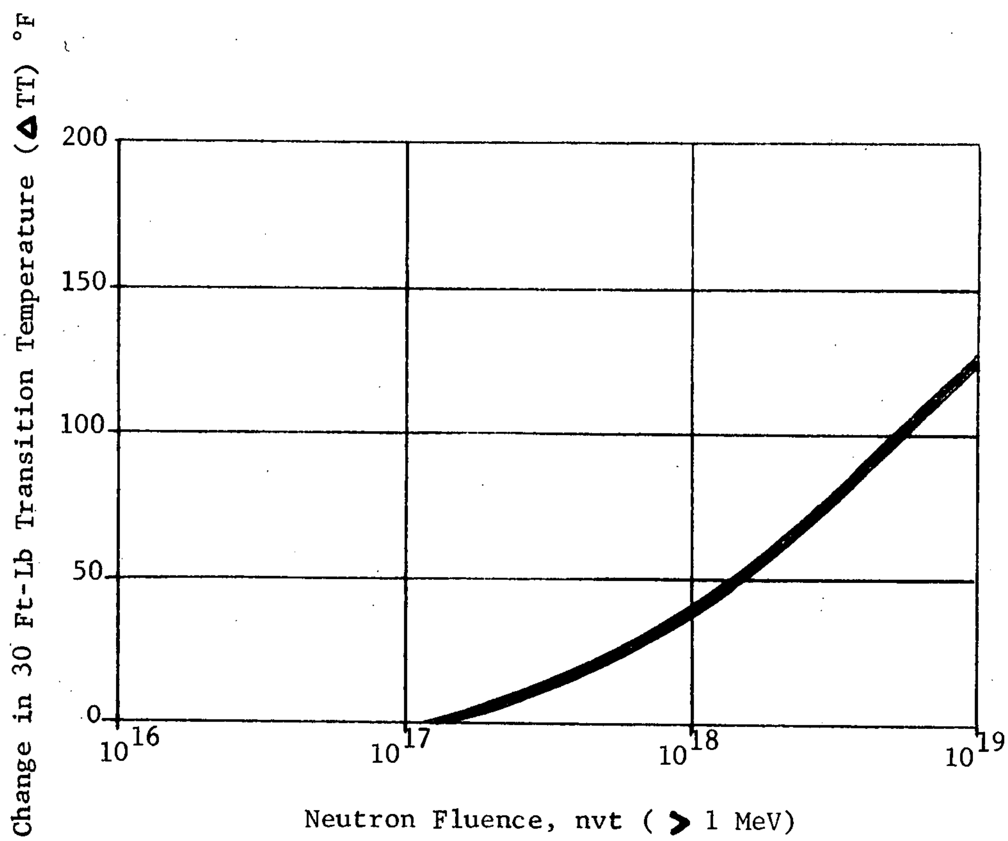

Figure 3.6-1 Change in Charpy V Transition Temperature Versus Neutron Exposure

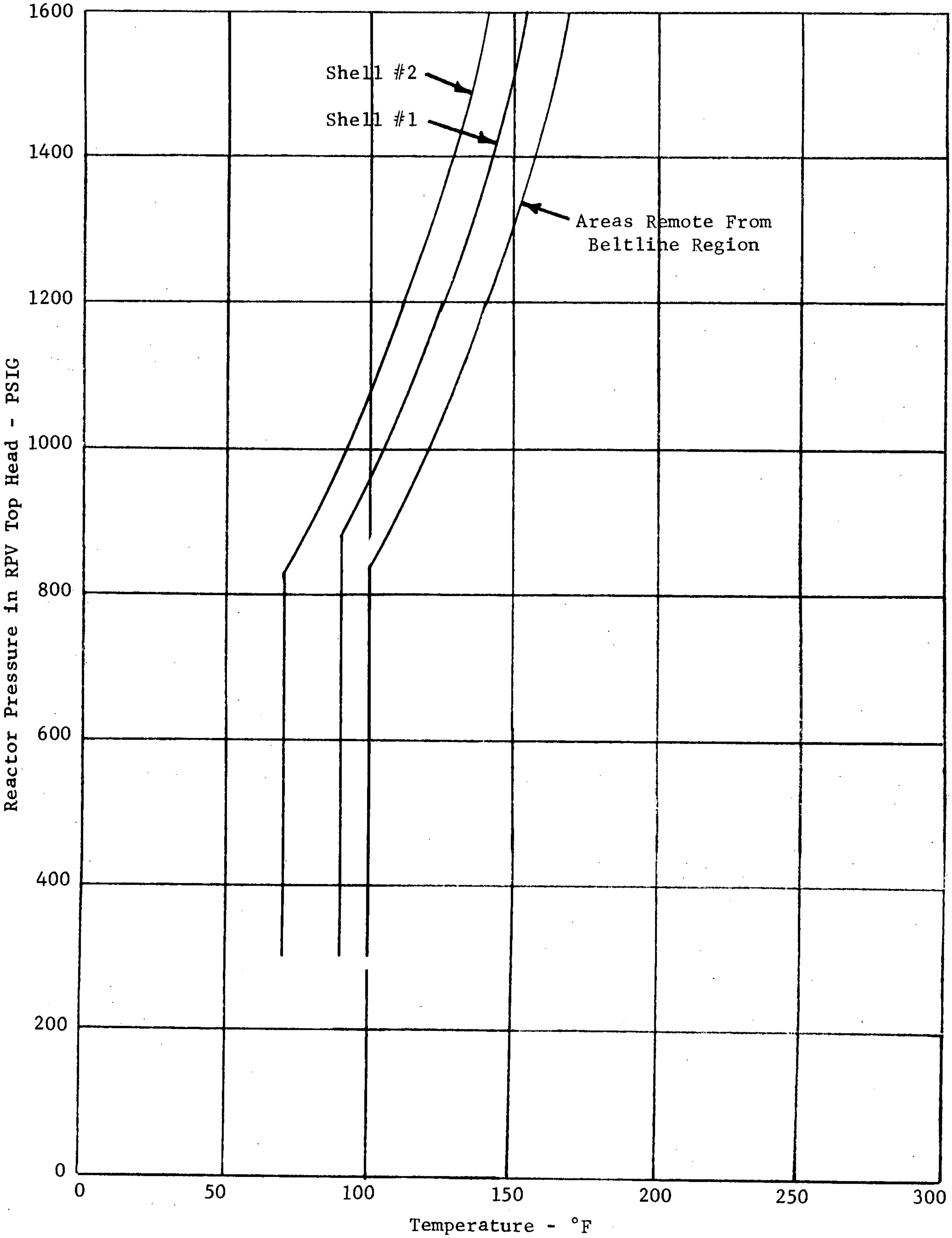

Figure 3.6-2A Minimum Temperature for Pressure Tests

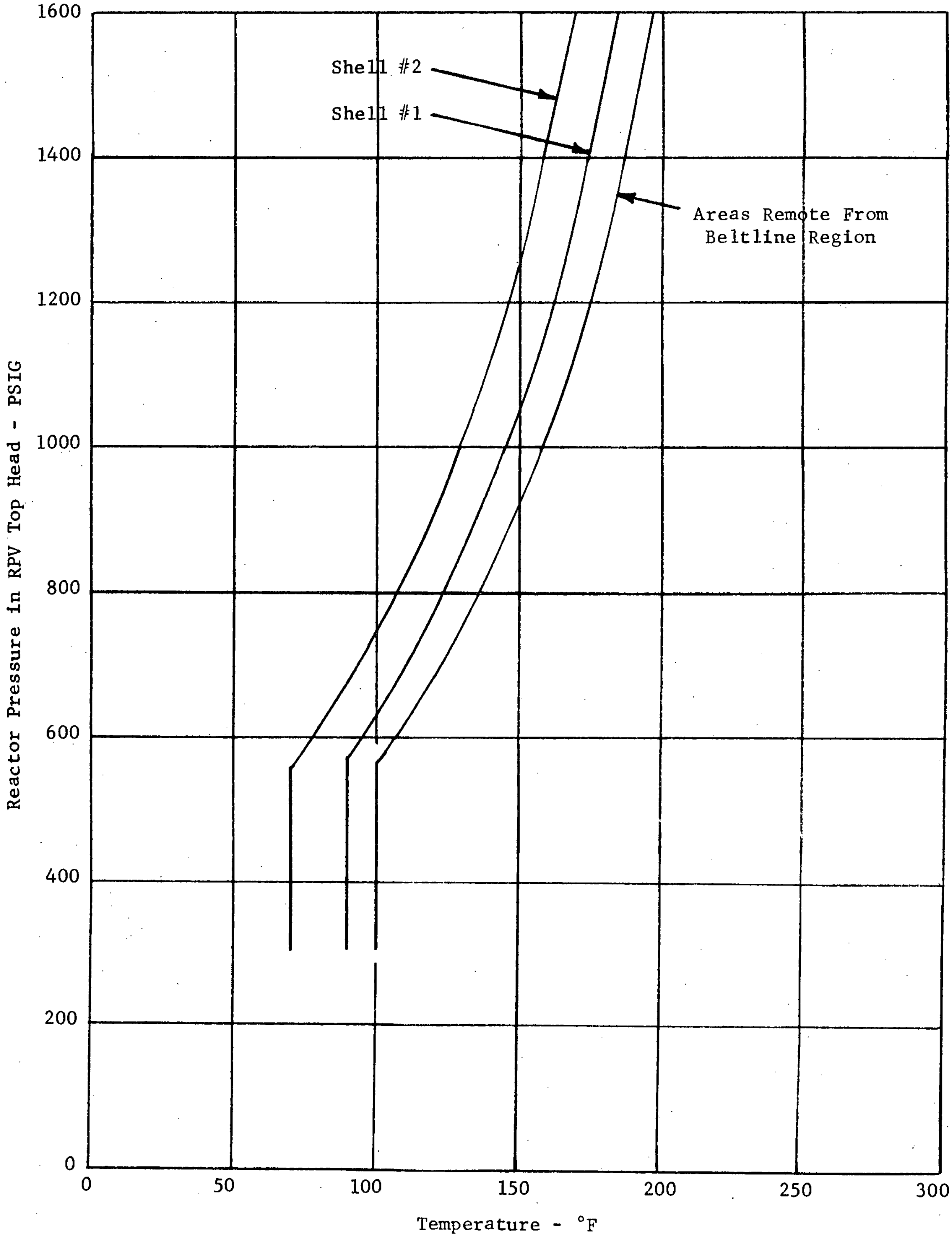

Figure 3.6-2B Minimum Temperature for Mechanical Heatup or Cooldown Following Nuclear Shutdown

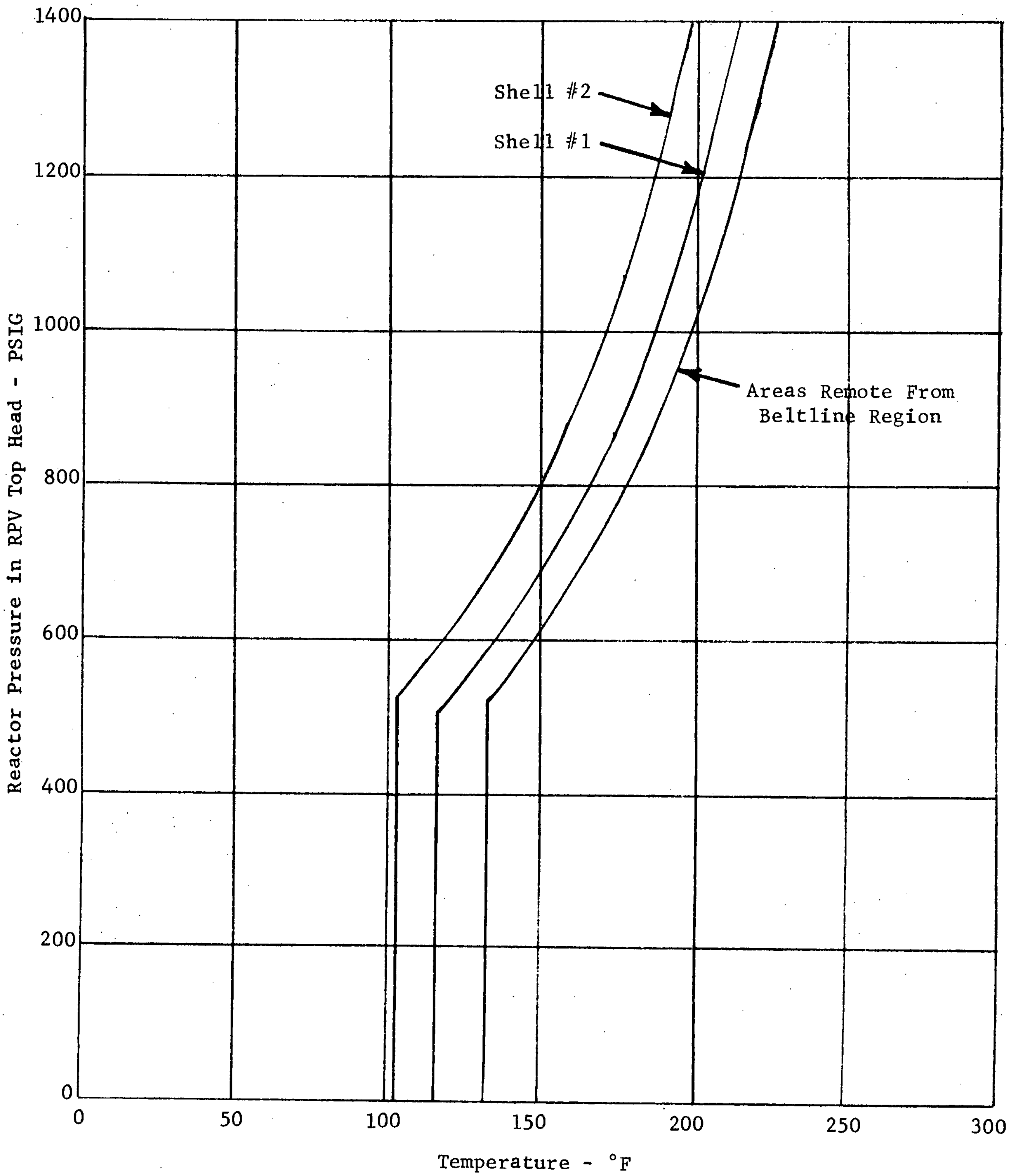

Figure 3.6-2C Minimum Temperature for Criticality