August 19, 1988

Docket No. 50-346
Serial No. DB-88-008

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Mr. Donald C. Shelton
Vice President, Nuclear
Toledo Edison Company
Plaza - Stop 712
300 Madison Avenue
Toledo, Ohio 43652

Dear Mr. Shelton:

SUBJECT: AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NO. NPF-3: REACTOR VESSEL PRESSURE-TEMPERATURE OPERATING LIMITS AND SURVEILLANCE REQUIREMENTS (TAC NO. 66699)

The Commission has issued Amendment No. 116 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1. This amendment consists of changes to the Appendix A Technical Specifications (TS's) in response to your application dated March 30, 1988 (No. 1490), as supplemented by letter dated May 4, 1988 (No. 1510).

This amendment revises License Condition 2.C(3)(d) and TS Sections 3.4.2, 3.4.9.1, 4.4.9.1.2, and Figures 3.4-2, 3.4-2a, 3.4-2b, 3.4-3, and 3.4-4. Table 4.4-5 is deleted. These revisions account for the projected changes in material properties for reactor vessel exposure to high energy neutrons for 10 effective full-power years.

A copy of the Safety Evaluation related to this amendment and a copy of the notice of issuance, which has been forwarded to the Office of the Federal Register for publication, are also enclosed.

Sincerely,

/S/

Albert W. De Agazio, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:
1. Amendment No.116 to License No. NPF-3
2. Safety Evaluation
3. Notice of Issuance

cc: See next page

*SEE PREVIOUS CONCURRENCE

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Date:	08/04/88	08/05/88	08/08/88	08/16/88

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Davis-Besse Nuclear Power Station
Unit No. 1

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

TOLEDO EDISON COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DOCKET NO. 50-346

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 116
License No. NPF-3

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

   A. The application for amendment by the Toledo Edison Company and The Cleveland Electric Illuminating Company (the licensees) dated March 30, 1988, supplemented May 4, 1988 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 by amending paragraphs 2.C.(2) and 2.C.(3)(d) of Facility Operating License No. NPF-3 to read as follows:

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2.C.(2) Technical Specifications

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

2.C.(3)(d) Prior to operation beyond 10 Effective Full Power Years, the Toledo Edison Company shall provide to the NRC a reanalysis and proposed modifications, as necessary, to ensure continued means of protection against low temperature reactor coolant system overpressure events.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to entering Mode 2 following the fifth refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Kenneth E. Perkins, Director
Project Directorate III-3
Division of Reactor Projects - III, IV, V, & Special Projects

Attachment: Changes to the Technical Specifications

Date of Issuance: August 19, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 116

FACILITY OPERATING LICENSE NO. NPF-3

DOCKET NO. 50-346

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove	Insert
3/4 4-3	3/4 4-3
3/4 4-4a	3/4 4-4a
3/4 4-4b	3/4 4-4b
3/4 4-24	3/4 4-24
3/4 4-25	3/4 4-25
3/4 4-26	3/4 4-26
3/4 4-27	3/4 4-27
3/4 4-28	3/4 4-28
B 3/4 4-7	B 3/4 4-7
B 3/4 4-8	B 3/4 4-8
B 3/4 4-9	B 3/4 4-9
B 3/4 4-10	B 3/4 4-10
B 3/4 4-11	B 3/4 4-11
B 3/4 4-12	B 3/4 4-12

SAFETY VALVES AND ELECTROMATIC RELIEF VALVE - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of ≤ 2525 PSIG.* When not isolated, the pressurizer electromatic relief valve shall have a trip setpoint of ≥ 2390 PSIG and an allowable value of ≥ 2390 PSIG.**

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 For the pressurizer code safety valves, there are no additional Surveillance Requirements other than those required by Specification 4.0.5. For the pressurizer electromatic relief valve a channel calibration check shall be performed every 18 months.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

** Allowable value for channel calibration check.

LIMITING CONDITION FOR OPERATION

3.4.2 Decay Heat Removal System relief valve DH-4849 shall be OPERABLE with a lift setting of ≤ 330 PSIG* and isolation valves DH-11 and DH-12 open and control power to their valve operators removed.

APPLICABILITY: MODES 4 and 5.

ACTION:

A. With DH-4849 not OPERABLE:

1. Make the valve OPERABLE within eight hours; or

2. a. Within next one hour, disable the capability of both high pressure injection (HPI) pumps to inject water into the reactor coolant system; and

b. Within next eight hours:

1. Disable the automatic transfer of makeup pump suction to the borated water storage tank on low makeup tank level; and

2. Reduce makeup tank level to ≤ 73 inches and reduce reactor coolant system pressure and pressurizer level within the acceptable region on Figures 3.4-2a (in MODE 4) and 3.4-2b (in MODE 5).

B. With DH-11 or DH-12 closed, open DH-21 and DH-23 within one hour.

C. With the control power not removed from DH-11 and DH-12, remove the power to the valve operators at the Motor Control Centers within one hour.

SURVEILLANCE REQUIREMENTS

4.4.2 Decay Heat Removal System relief valve DH-4849 shall be determined OPERABLE:

a. per the surveillance requirements of Specification 4.0.5.

b. at least once per 24 hours by verifying either:

1. isolation valves DH-11 and DH-12 open with control power removed from their valve operators; or

2. valves DH-21 and DH-23 open.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

Figure 3.4-2b

Reactor Coolant System Pressure - Pressurizer Level Limits for inoperable Decay Heat Removal System Relief Value in MODE 5

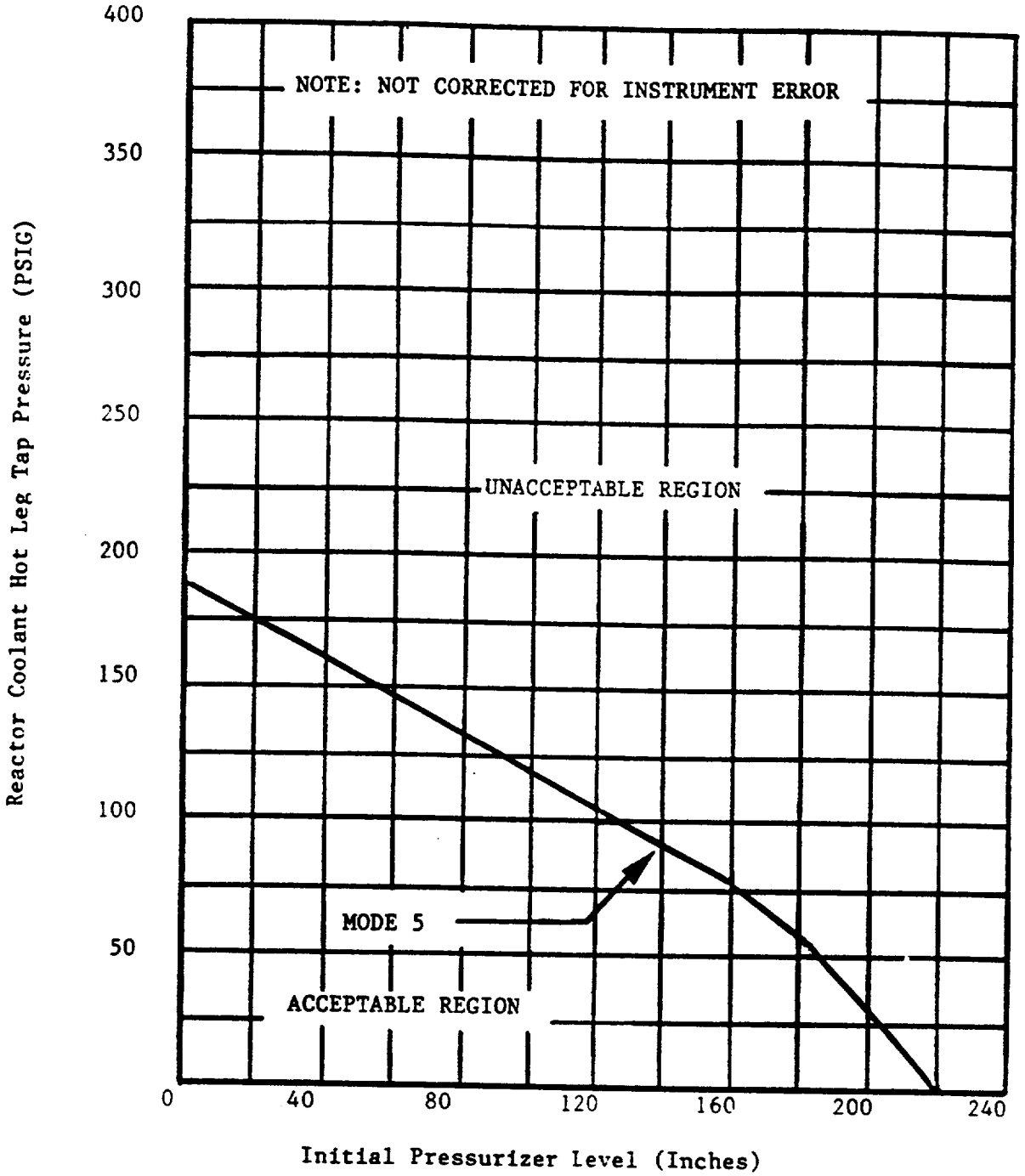
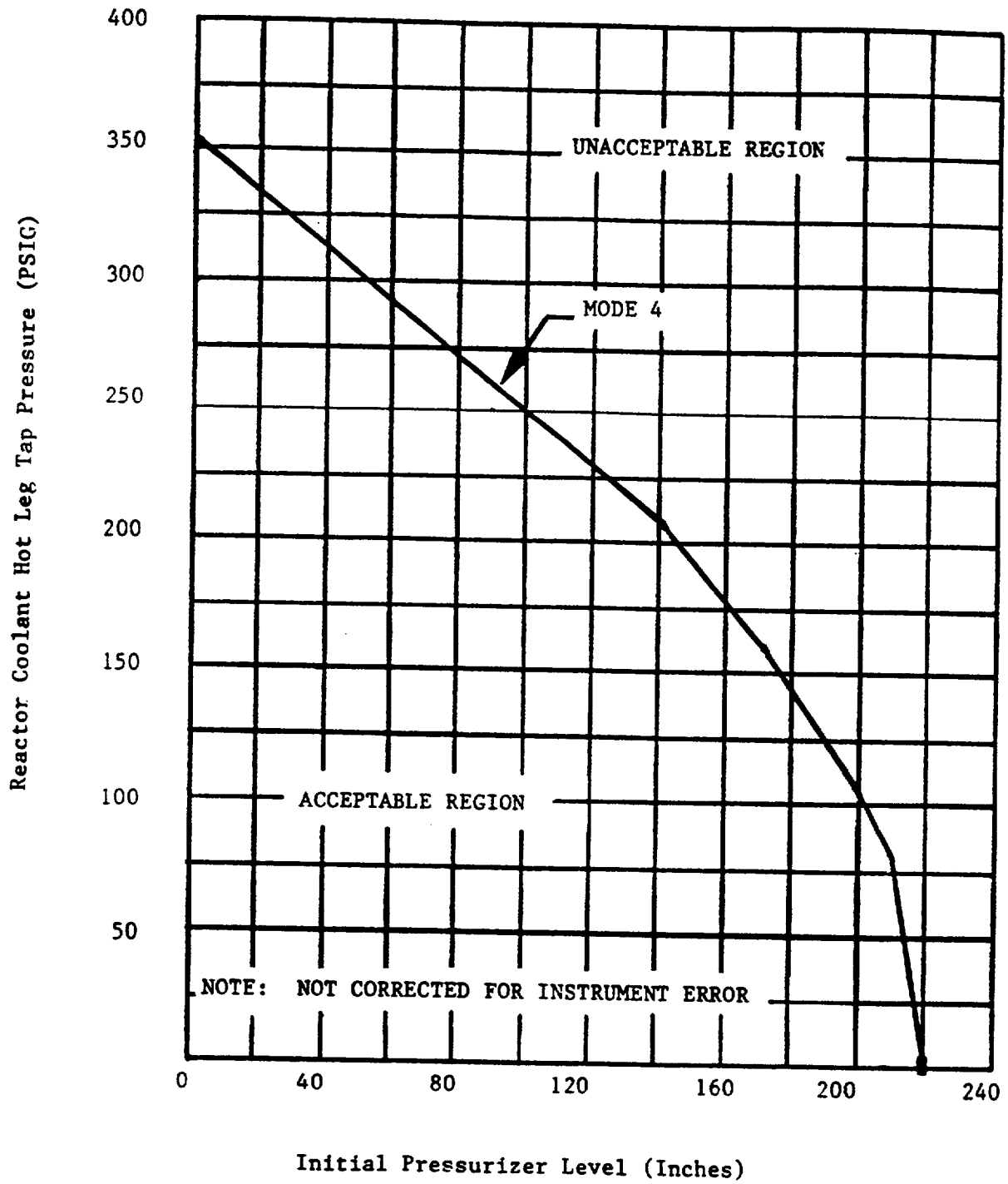


Figure 3.4-2a

Reactor Coolant System Pressure - Pressurizer Level Limits for inoperable Decay Heat Removal System Relief Valve in MODE 4



3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant system (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2, 3.4-3 and 3.4-4 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 50°F in any one hour period, and

b. A maximum cooldown of 100°F in any one hour period with cold leg temperature $\geq$ 270°F and a maximum cooldown of 50°F in any one hour period with cold leg temperature <270°F.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens representative of the vessel materials shall be removed and examined, to determine changes in material properties, at the intervals defined in BAW 1543A. The results of these examinations shall be used to update Figures 3.4-2, 3.4-3 and 3.4-4.

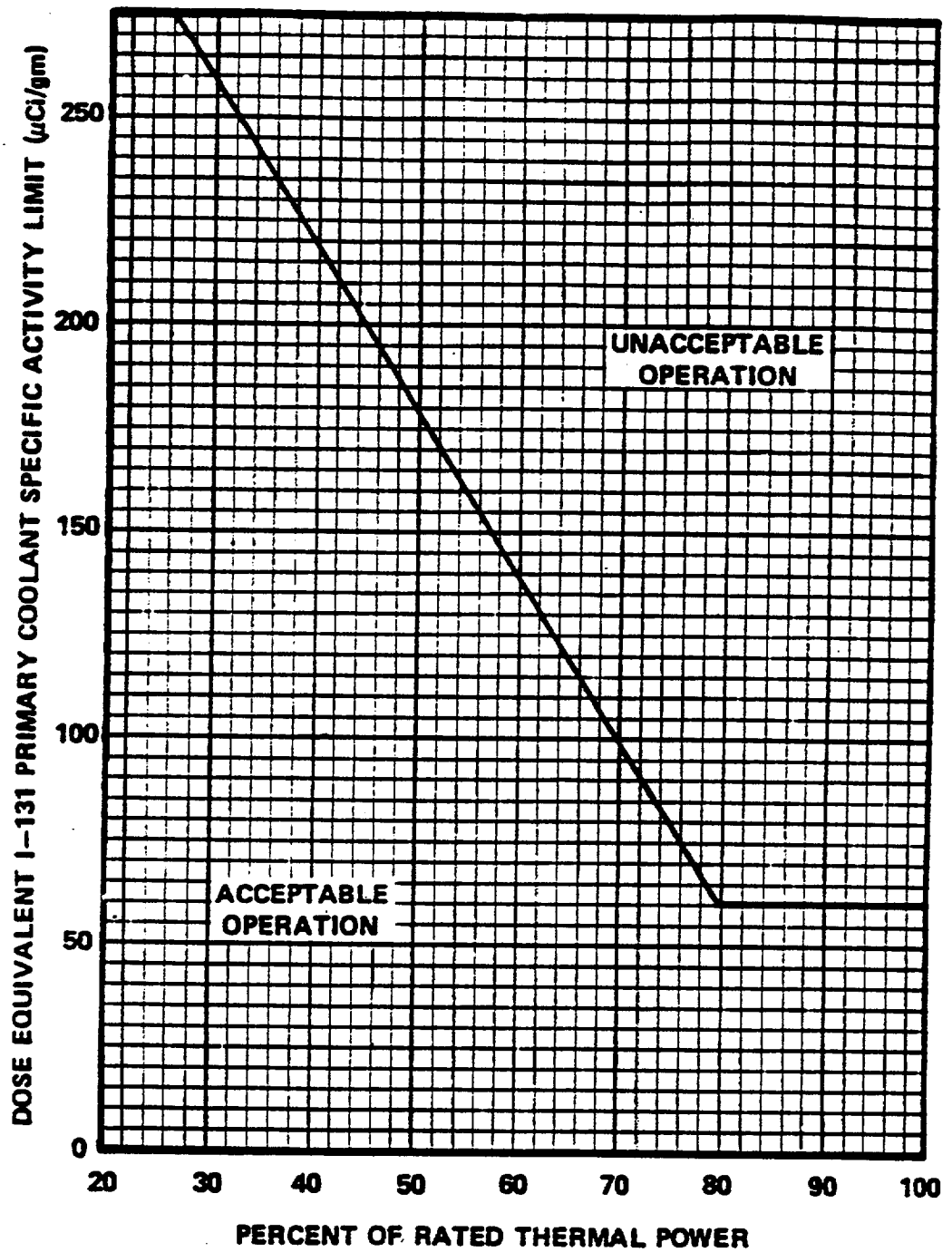


FIGURE 3.4-1

DOSE EQUIVALENT I–131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0 μCi/gram Dose Equivalent I-131

Figure 3.4-3

Reactor Coolant System Pressure - Temperature Limits for Cooldown for the First 10 EFPY

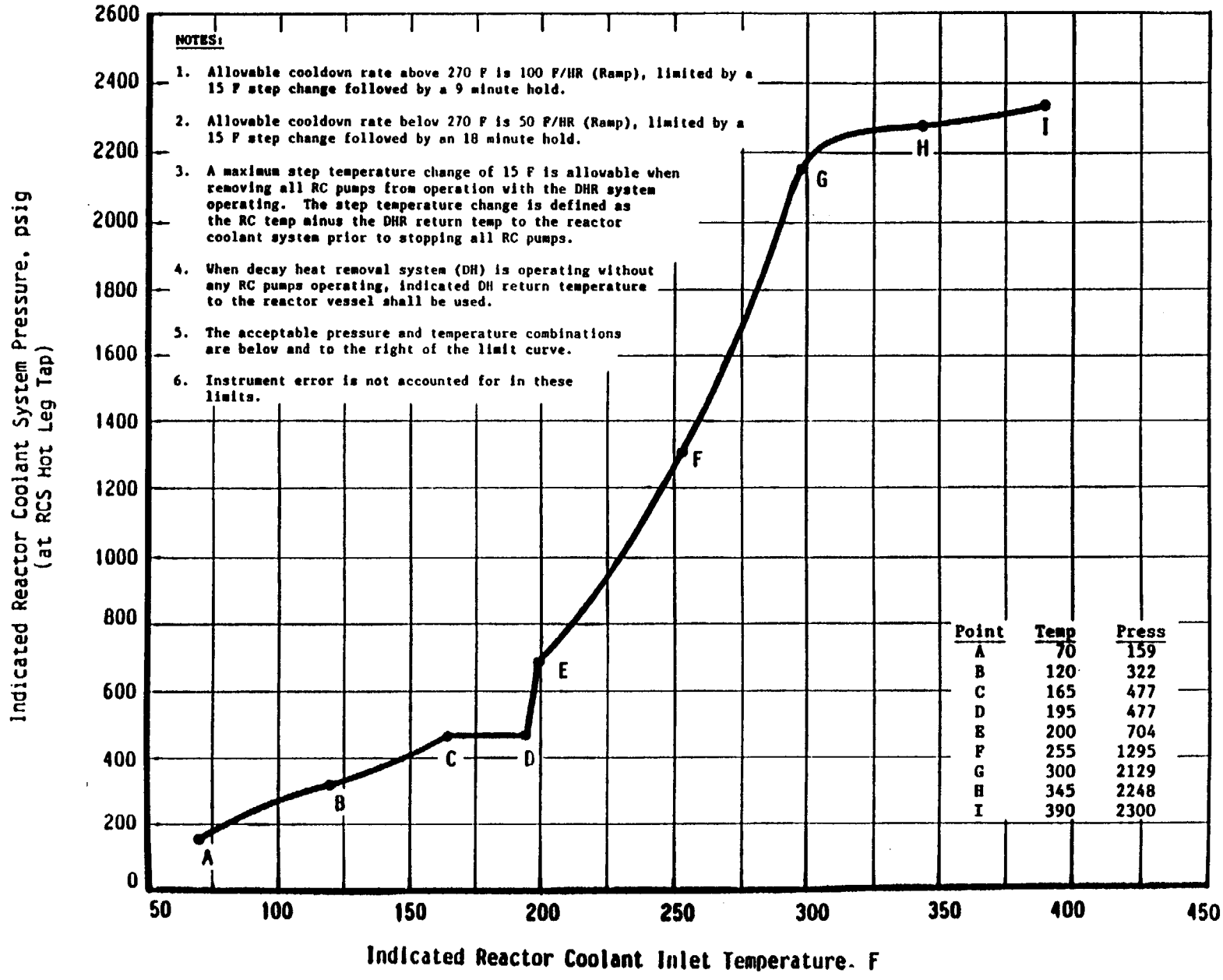


Figure 3.4-2

Reactor Coolant System Pressure - Temperature Limits
for Heatup and Core Criticality for the First 10 EFPY

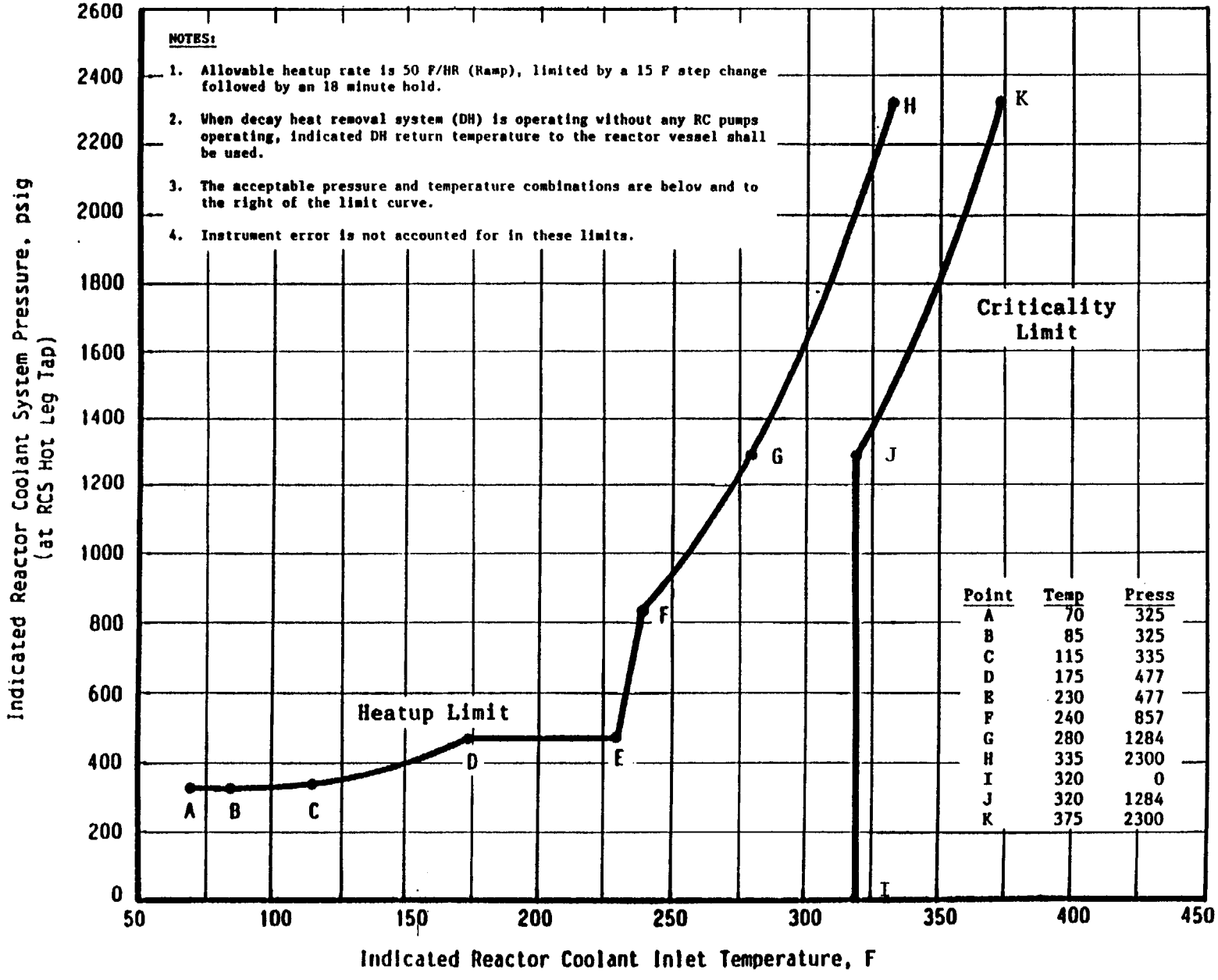


Table 4.4-5

Reactor Vessel Material Irradiation
Surveillance Schedule

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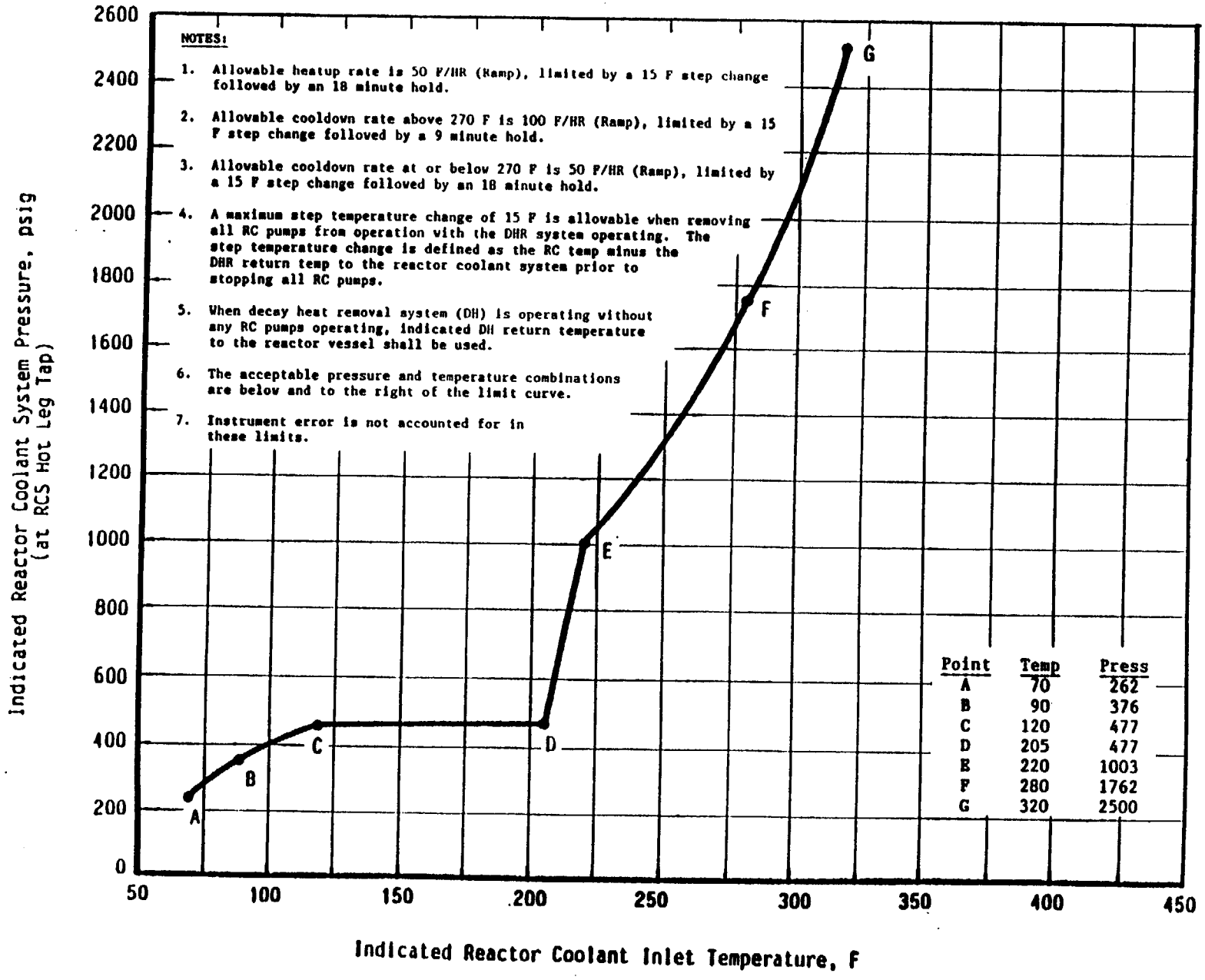
Figure 3.4-4

Reactor Coolant System Pressure - Temperature Heatup and Cooldown Limits for Inservice Leak and Hydrostatic Tests for the First 10 EFPY



Bases Figure 4-2

Effect of Fluence and Copper Content on Shift of RT_{NDT} for Reactor Vessel Steels Exposed to 550°F Temperature

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Bases Figure 4-1

Fast Neutron Fluence (E>1 Mev) as a Function of Full Power Service Life

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The closure head region is significantly stressed at relatively low temperatures (due to mechanical loads resulting from bolt pre-load). This region largely controls the pressure-temperature limitations of the first several service periods. The outlet nozzles of the reactor vessel also affect the pressure-temperature limit curves of the first several service periods. This is due to the high local stresses at the inside corner of the nozzle which can be two to three times the membrane stresses of the shell. After the first several years of neutron radiation exposure, the RT_{NDT} temperature of the beltline region materials will be high enough so that the beltline region of the reactor vessel will start to control the pressure-temperature limitations of the reactor coolant pressure boundary. For the service period for which the limit curves are established, the maximum allowable pressure as a function of fluid temperature is obtained through a point-by-point comparison of the limits imposed by the closure head region, outlet nozzles, and beltline region. The maximum allowable pressure is taken to be the lower pressure of the three calculated pressures. The pressure limit is adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all operating reactor coolant pump combinations. The limit curves were prepared based upon the most limiting adjusted reference temperature of all the beltline region materials at the end of the tenth effective full power year.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating, in accordance with Appendix H to 10 CFR 50, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside the radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The limit curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Bases Table 4-1

Reactor Vessel Toughness

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The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in BAW 1543A. The withdrawal schedule is based on four considerations: (a) uncover possible technical anomalies as early in life as they can be detected (end of first fuel cycle), (b) define the material properties needed to perform the analysis required by Appendix G to 10 CFR 50, (c) reserve two capsules for evaluation of the effectiveness of thermal annealing in the event inplace annealing becomes necessary, (d) provide material property data corresponding to the reactor vessel beltline conditions at the end of service. This withdrawal schedule is specified to assure compliance with the requirements of Appendix H to 10 CFR 50. Appendix H references the requirements of ASTM E185 for surveillance program criteria.

The unirradiated transverse impact properties of the beltline region materials, required by Appendices G and H to 10 CFR 50, were determined for those materials for which sufficient amounts of material were available. The adjusted reference temperatures are calculated by adding the predicted radiation-induced ΔRT_{NDT} and the unirradiated RT_{NDT}. The procedures described in Regulatory Guide 1.99, Rev. 2, were used for predicting the radiation induced ΔRT_{NDT} as a function of the material's copper and nickel content and neutron fluence.

Figure 3.4-2 presents the pressure-temperature limit curve for normal heatup. This figure also presents the core criticality limits as required by Appendix G to 10 CFR 50. Figure 3.4-3 presents the pressure-temperature limit curve for normal cooldown. Figure 3.4-4 presents the pressure-temperature limit curves for heatup and cooldown for inservice leak and hydrostatic testing.

All pressure-temperature limit curves are applicable up to the tenth effective full power year. The protection against non-ductile failure is assured by maintaining the coolant pressure below the upper limits of Figures 3.4-2, 3.4-3 and 3.4-4.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NO. NPF-3

TOLEDO EDISON COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

1.0 INTRODUCTION

By letter dated March 30, 1988, Toledo Edison Company requested approval to revise section 3/4.4.9 and bases section 3/4.4.9 of the Davis-Besse Technical Specifications; both relate to the pressure-temperature limits of the Reactor Coolant System and surveillance capsule withdrawal schedule.

The primary purpose of this request is to revise the existing pressure temperature limits and extend the operation period of the limits up to 10 effective full power years (EFPY). The revised pressure-temperature limits are based on the Babcock and Wilcox analyses (BAW-1882 and BAW-2011) of irradiation data taken from removed surveillance capsule TE1-A. The limits provide permissible pressure and temperature for three operating conditions - heatup and core criticality (Figure 3.4-2), cooldown (Figure 3.4-3), and inservice leak and hydrostatic tests (Figure 3.4-4). The licensee also requested approval to revise the existing surveillance capsule withdrawal schedule.

2.0 DISCUSSION

Part of the NRC's effort to ensure integrity of the reactor vessel is to periodically evaluate the reduction in fracture toughness of the vessel material caused by neutron irradiation embrittlement. The effort consists of three steps.

First, the licensee is required to establish a surveillance program in accordance with Appendix H of 10 CFR Part 50, which requires periodic withdrawal of surveillance capsules from the reactor vessel. The capsules are installed in the vessel prior to startup and they should contain test specimens that were made from the plate, weld, and heat-affected-zone materials of the reactor beltline.

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Secondly, the licensee is required to perform Charpy impact tests, tensile tests, and neutron fluence measurements on the specimens. These tests define the condition of vessel embrittlement at the time of capsule withdrawal in terms of the increase of the reference temperature, RT_{NDT}, and upper shelf energy. The licensee must also predict the future vessel embrittlement by calculating the adjusted RT_{NDT} and upper shelf energy at a specific EFPY. The licensee may use either Revision 1 or draft Revision 2 of Regulatory Guide 1.99 to calculate the adjusted RT_{NDT}. The upper shelf energy is the average energy value for all specimens whose test temperature is above the upper end of the transition temperature region. The licensee is required by 10 CFR Part 50, Appendix G, to provide assurance that the adjusted RT_{NDT} will not exceed 200°F and that the upper shelf energy will not be below 50 ft-lb at the end of plant life.

Thirdly, the licensee is required to develop a set of pressure-temperature curves based on the adjusted RT_{NDT} of the limiting vessel materials. The curves must satisfy the recommended methods and requirements described in 10 CFR Part 50, Appendix G, and Standard Review Plan 5.3.2.

3.0 EVALUATION

Surveillance Program

The surveillance program in Davis-Besse Unit 1 is part of the Integrated Reactor Vessel Material Surveillance Program sponsored by Babcock and Wilcox (Report No. BAW-1543A), which the staff approved in 1985. B&W initiated the Integrated Program as a result of flow-induced vibration that caused cracking of surveillance capsule holder tubes in some B&W reactor vessels. The capsules were removed from these nuclear plants, redesigned, and installed in other B&W reactors. Davis-Besse, Unit 1 was selected as a host reactor.

The surveillance program in Davis-Besse consists of six surveillance capsules. Capsules TE1-F, and TE1-B, and TE1-A have been removed at the end of fuel cycles 1, 3, and 4, respectively. The proposed pressure-temperature limits were determined based on the data taken from Capsule TE1-A, which was positioned inside the reactor vessel between the thermal shield and the vessel wall.

This capsule contained Charpy V-notch test specimens fabricated from two base metals, a weld metal, two heat-affected-zone metals, and a correlation metal. All test specimens were machined from the 1/4-thickness of the forging material and can be traced to the vessel material.

The licensee proposed to delete the existing capsule withdrawal schedule in section 3/4.4.9 of the Technical Specifications and replace it with the withdrawal schedule recommended for Davis-Besse in the Integrated Program in Bases section 3/4.4.9. The remaining three capsules in Davis-Besse will be removed at the end of the sixth, eighth and eleventh fuel cycles. The staff concludes that this withdrawal schedule for the remaining capsules satisfies the requirements of Appendix H to 10 CFR Part 50 and ASTM E 185-82.

UPPER SHELF ENERGY

The licensee used methods in Regulatory Guide 1.99, Revision 2, and BAW-1803 (a B&W Owner Group method) to evaluate the reactor vessel end-of-life upper shelf energy (USE). The weld material in the Davis-Besse reactor used a submerged arc weld process with Linde 80 flux. This weld material has high copper content and low unirradiated Charpy USE; therefore, it is susceptible to neutron irradiation damage. The method given in Regulatory Guide 1.99, Rev. 2 estimated a USE of 47 ft-lbs whereas the BAW-1803 method estimated a 67 ft-lbs USE at end-of-life for weld WF-182-1. The BAW-1803 method uses a statistical analysis of all the existing surveillance Linde 80 weld metal; therefore, it estimates a higher USE than Reg. Guide 1.99. The staff considers that the statistical analysis in BAW-1803 is an acceptable alternative to the R.G. 1.99 Rev. 2 method and that 67 ft-lbs is acceptable. The licensee showed that none of the vessel material will have a USE below 50 ft-lbs at end-of-life at the T/4 wall location. The staff concludes that the USE's of Davis-Besse reactor materials satisfy Appendix G of 10 CFR Part 50.

REFERENCE TEMPERATURE

The observed increase in reference temperature (RT_{NDT}) in the surveillance specimens was determined based on the difference between the RT_{NDT} at the time of withdrawal and the initial RT_{NDT}. The RT_{NDT} is the test temperature that corresponds to the 30 ft-lb impact absorption energy in the Charpy V-notch impact tests. From the Charpy test, the licensee determined that the maximum observed increase in the RT_{NDT} was 175°F and occurred in the middle circumference seam weld, WF-182-1. Based on the maximum RT_{NDT} shift, this weld metal is selected to be the limiting material. Capsule TE1-A received a neutron fluence of 1.29E19 n/cm^2 and the inner wall fluence at 10 EFPY was calculated to be 5.63E18 n/cm^2. The ratio of the neutron flux density at the capsule location to the maximum calculated neutron flux density at the inside surface of the vessel wall (lead factor) was about 2.0.

Since the pressure-temperature limits are usually needed for the future EFPY's, the licensee needs to predict the adjusted RT_{NDT} at a future EFPY so that a valid RT_{NDT} could be used in determining the pressure-temperature limits. The staff used the Revision 2 method to evaluate the licensee's predictions of the adjusted RT_{NDT}. The predicted RT_{NDT} depends on the neutron fluence at the specific vessel wall location, specific EFPY, and copper content and nickel content of the vessel material. The neutron fluence at T/4 vessel wall location (T = vessel thickness) and 10 EFPY was extrapolated from the observed neutron fluence of capsule TE1-A and was calculated to be 3.46E18 n/cm^2. The maximum predicted adjusted RT_{NDT} was determined to be 146°F for the limiting weld material.

The observed and predicted RT_{NDT} for base metal and heat-affected-zone metal are compared in Table 1. The predicted RT_{NDT} for the base metal exceeds the observed RT_{NDT} while the predicted RT_{NDT} for the weld metal is less than the observed RT_{NDT}. The licensee used the predicted RT_{NDT} of 147°F to determine the pressure-temperature limits instead of the observed value of 175°F. The requirements in subsection 2.1 of Reg. Guide 1.99, Rev. 2 allow the use of the lower of the two RT_{NDT} when the licensee uses previous known surveillance data in the prediction. The staff finds that the use of 147°F for the RT_{NDT} in the proposed pressure-temperature limits is more reasonable than 175°F because the RT_{NDT} of 175°F corresponds to a neutron fluence at 22 EFPY.

The RT_{NDT} at end-of-life and T/4 location was calculated to be 222°F. Davis-Besse was a part of the pressurized thermal shock study that has an allowable RT_{NDT} of 300°F, as described in section 50.61 of 10 CFR Part 50. The staff, therefore, finds the end-of-life RT_{NDT} acceptable.

PRESSURE-TEMPERATURE LIMITS

The staff used the methods described in Standard Review Plan 5.3.2 to evaluate three proposed pressure-temperature limits for heatup and core criticality, cooldown, and hydrostatic leak test. The SRP 5.3.2 method uses the principle of linear elastic fracture mechanics to calculate the pressure-temperature limits. The basic parameter is the stress intensity factor, which is a function of pressure and thermal stresses and flaws in the vessel material. The stress intensity factor is related to the reactor coolant system temperature and RT_{NDT} of the vessel material by an exponential curve. For a given operating pressure, a reference stress intensity factor is calculated and via the exponential curve, the system operating temperature could be determined. Aside from the pressure and thermal stresses, the pressure-temperature limits should also consider the bolt preload stress at reactor closure flange regions as described in Appendix G to 10 CFR Part 50. The staff finds that the proposed pressure-temperature limits in Figures 3.4-2, -3, and -4 are in accord with SRP 5.3.2 and meet the requirements of Appendix G to 10 CFR Part 50.

4.0 FINDINGS

The staff finds that the proposed surveillance capsule withdrawal schedule is acceptable based on the requirements of Appendix H to 10 CFR Part 50 and ASTM E 185-82.

The staff has concluded that the proposed pressure-temperature limits on the reactor coolant system for heatup, core criticality, cooldown, and hydrostatic test operating conditions are in conformance with requirements of Appendix G to 10 CFR Part 50. The proposed limits are acceptable up to 10 effective full power years.

The staff agrees that the proposed surveillance withdrawal schedule and pressure-temperature limits may be incorporated into the Davis-Besse Technical Specifications.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register (July 18, 1988, 53 FR 27095). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

6.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: J. Tsao

Dated: August 19, 1988

TABLE 1

Specimen (1)	Copper Content %	Nickel Content %	Neutron Fluence @ Inner Surface n/cm^2	Observed RT_{NDT} °F	Adjusted RT_{NDT} @ 10 EFPY and T/4 wall location: Staff Prediction °F	Adjusted RT_{NDT} @ 10 EFPY and T/4 wall location: Licensee Prediction °F
Base Metal BCC-241	0.02	0.81	5.63E18	28	78	78
Base Metal AKJ-233	0.04	0.77	5.63E18	2	56	56
Heat Affected Zone BCC-241	0.02	0.81	5.63E18	34	78	Not Available
Weld Metal WF-182-1	0.24	0.63	5.63E18	175	146(2)	147(2)

Notes

(1) All surveillance specimens can be traced to the vessel material.

(2) Use capsule fluences and observed RT_{NDT} shifts from capsules TE1-A, TE1-B, and TE1-F to calculate the chemistry factor per subsection 2.1 of Reg. Guide 1.99 Rev. 2.

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U. S. NUCLEAR REGULATORY COMMISSION

TOLEDO EDISON COMPANY, ET AL.

DOCKET NO. 50-346

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 116 to Facility Operating License No. NPF-3, issued to The Toledo Edison Company and The Cleveland Electric Illuminating Company (the licensee), which changed the license and revised the Technical Specifications for operation of the Davis-Besse Nuclear Power Station, Unit No. 1 (the facility) located in Ottawa County, Ohio. The amendment was effective as of the date of its issuance.

The amendment changed a license condition and the Technical Specifications to revise the existing pressure temperature limits and extend the operation period of the limits up to 10 effective full power years; the amendment also approved revision of the existing surveillance capsule removal schedule.

The application for the amendment 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 amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER

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on May 4, 1988 (53 FR 15932). No request for hearing or petition for leave to intervene was filed following this notice.

For further details with respect to this action see (1) the application for amendment dated March 30, 1988 as supplemented May 4, 1988, (2) Amendment No. 116 to License No. NPF-3, (3) the Commission's related Safety Evaluation dated August 19, 1988 and (4) the Environmental Assessment dated July 7, 1988 (53 FR 27095). 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 University of Toledo Library, Documents Department, 2801 Bancroft Avenue, Toledo, Ohio 43606.

A copy of items (2), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Projects - III, IV, V and Special Projects.

Dated at Rockville, Maryland this 19th day of August 1988.

FOR THE NUCLEAR REGULATORY COMMISSION

Albert W. De Agazio

Albert W. De Agazio, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects - III,
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