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. 81
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 August 27, 1984, 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 2.C.(2) of Facility Operating License No. NPF-3 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 81 , are hereby incorporated in the license. The Toledo Edison Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

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: December 17, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 81

FACILITY OPERATING LICENSE NO. NPF-3

DOCKET NO. 50-346

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

Page

3/4 4-24
3/4 4-28
B 3/4 4-12

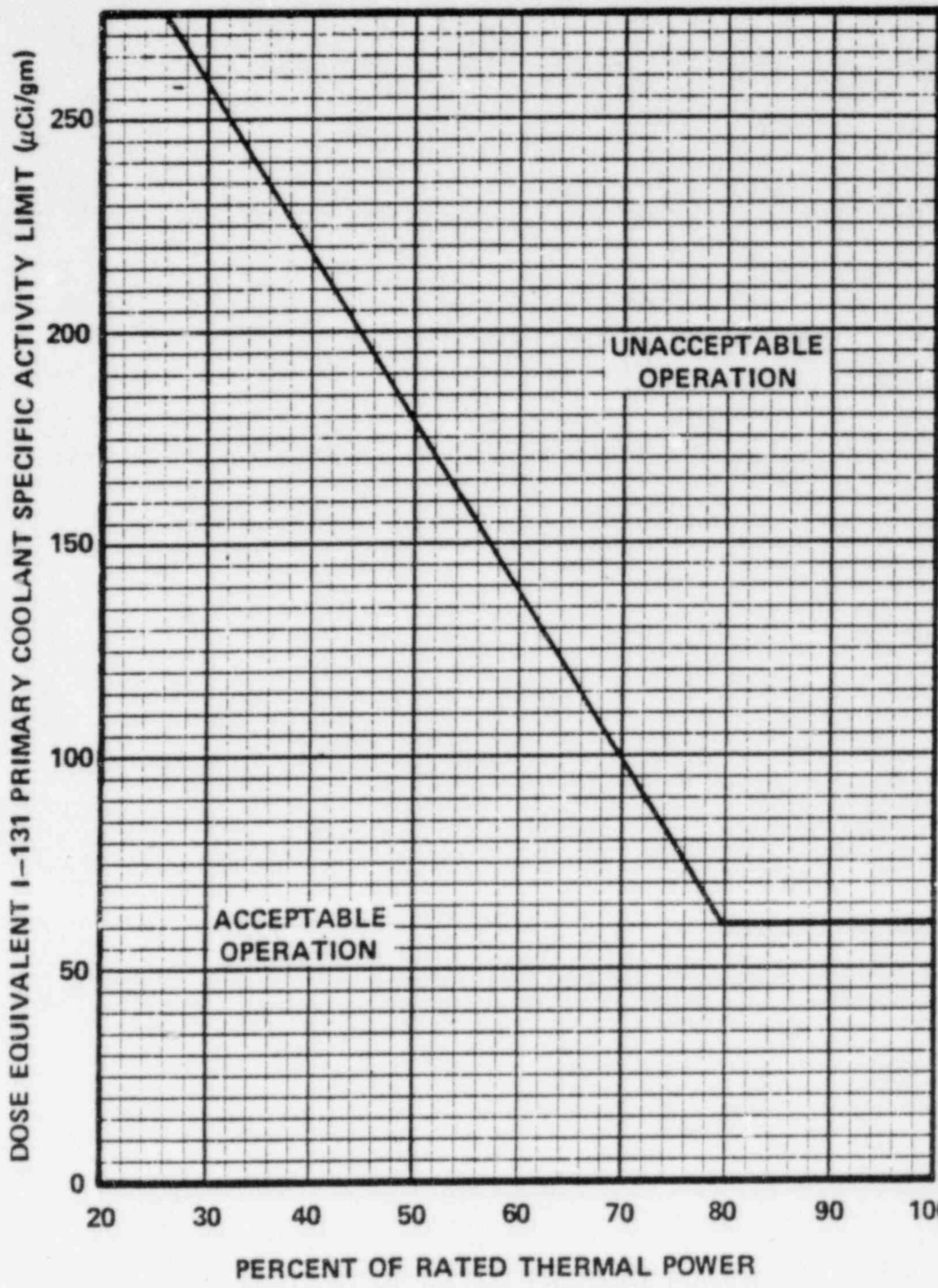


FIGURE 3.4-1

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

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 100°F in any one hour period, and

b. A maximum cooldown of 100°F in any one hour period.

APPLICABILITY: At all times.

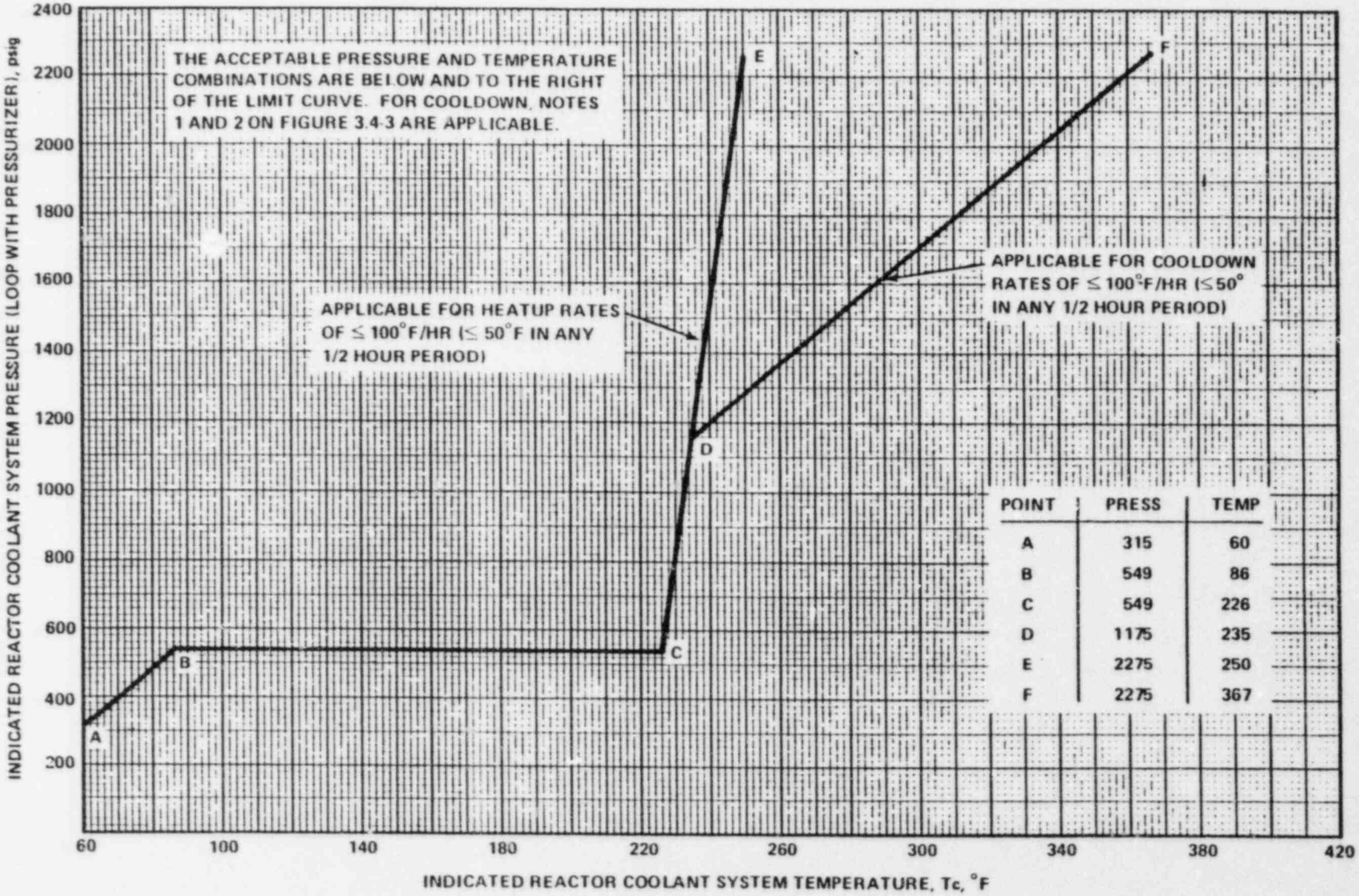
ACTION:

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 fracture toughness properties 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 reduce RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, 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 shown in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2, 3.4-3 and 3.4-4.



POINT	PRESS	TEMP
A	315	60
B	549	86
C	549	226
D	1175	235
E	2275	250
F	2275	367

Figure 3.4-1 Reactor Coolant System Pressure–Temperature Heatup and Cooldown Limits for Inservice Leak and Hydrostatic Tests for the First 5 EFPY

TABLE 4.4-5

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

Sequence	Time of Withdrawal
First	Earliest of: 1.5 EFPY; capsule fluence $> 5 \times 10^{18}$ n/cm^2; highest RT_{NDT} of an encapsulated material equals 50F.
Second	Earliest of: 3 EFPY; capsule fluence midway between that of the first and third capsules.
Third	Earliest of: 6 EFPY; capsule fluence corresponds to that of the EOL fluence of the reactor vessel 1/4T location.
Fourth	Schedule to be submitted for NRC approval prior to Cycle 6 operation.
Fifth	Schedule to be submitted for NRC approval prior to Cycle 6 operation.
Sixth	Schedule to be submitted for NRC approval prior to Cycle 6 operation.

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 unirradiated impact properties and residual elements of the beltline region materials are listed in Bases Table 4-1. The adjusted reference temperatures are calculated by adding the predicted radiation-induced ΔRT_{NDT} and the unirradiated RT_{NDT}. The predicted ΔRT_{NDT} are calculated using the respective neutron fluence and copper and phosphorus contents. Bases Figure 4-1 illustrates the calculated peak neutron fluence, at several locations through the reactor vessel beltline region wall and at the center of the surveillance capsules as a function of exposure time.

Bases Figure 4-2 illustrates the design curves for predicting the radiation-induced ΔRT_{NDT} as a function of the material's copper and phosphorus content and neutron fluence. The adjusted RT_{NDT}'s of the beltline region materials at the end of the fifth full power year are listed in Bases Table 4-1. The adjusted RT_{NDT}'s are given for the 1/4T and 3/4T (T is wall thickness) vessel wall locations. The assumed RT_{NDT} of the closure head region is 40°F and the outlet nozzle steel forgings is 60°F.

During cooldown at the higher temperatures, the limits are imposed by thermal and loading cycles on the steam generator tubes. These limits are segments D-E and D-F of the limit lines on Figures 3.4-2 and 3.4-4, respectively. These limits will not require adjustments due to the neutron fluences.

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 fifth 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.

BASES

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5. 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 the inplace annealing becomes necessary, (d) provide material property data corresponding to the reactor vessel line surface conditions at the end of service. The withdrawal schedule of Table 4.4-5 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. Table 4.4-5 is designed to meet the requirements of ASTM E185-82.