December 17, 1984

DMB 016

Docket No. 50-346

Mr. Richard P. Crouse
Vice President, Nuclear
Toledo Edison Company
Edison Plaza - Stop 712
300 Madison Avenue
Toledo, Ohio 43652

Dear Mr. Crouse:

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SUBJECT: AMENDMENT NO. 81 TO FACILITY OPERATING LICENSE NO. NPF-3; REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

The Commission has issued Amendment No. 81 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 in response to a portion of Item 1 of your application dated August 27, 1984 (No. 1074). Item 2 of your letter will be the subject of a separate action.

This amendment changes Surveillance Requirement 4.4.9.1.2 and associated Table 4.4-5, which relate to the reactor vessel material surveillance program, to reflect changes in operating cycle length and fuel loading scheme. However, we have approved the schedule changes (Table 4.4-5) for the first three capsules only. We are withholding approval of the schedules for the remaining capsules pending additional information. The additional information must be submitted not later than 180 days prior to the end of the refueling outage prior to Cycle 6 operation. We have discussed this with your staff. The amendment also includes changes to the Basis to delete redundant information and clarify, with specificity, the basis for the withdrawal schedule.

A copy of the Safety Evaluation supporting this amendment is enclosed. Notice of Issuance will be included in the Commission's Monthly Notice in the Federal Register.

The information requested in this letter affects fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

Sincerely,

ORIGINAL SIGNED BY
JOHN F. STOLZ

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

Enclosures:
1. Amendment No. 81
2. Safety Evaluation

cc w/enclosures:
See next page

ORB#4:DL	ORB#4:DL	MTB:DL	ORB#4:DL	AD:OR:DL	OELD
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Toledo Edison Company

cc w/enclosure(s):

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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. 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:

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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, 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

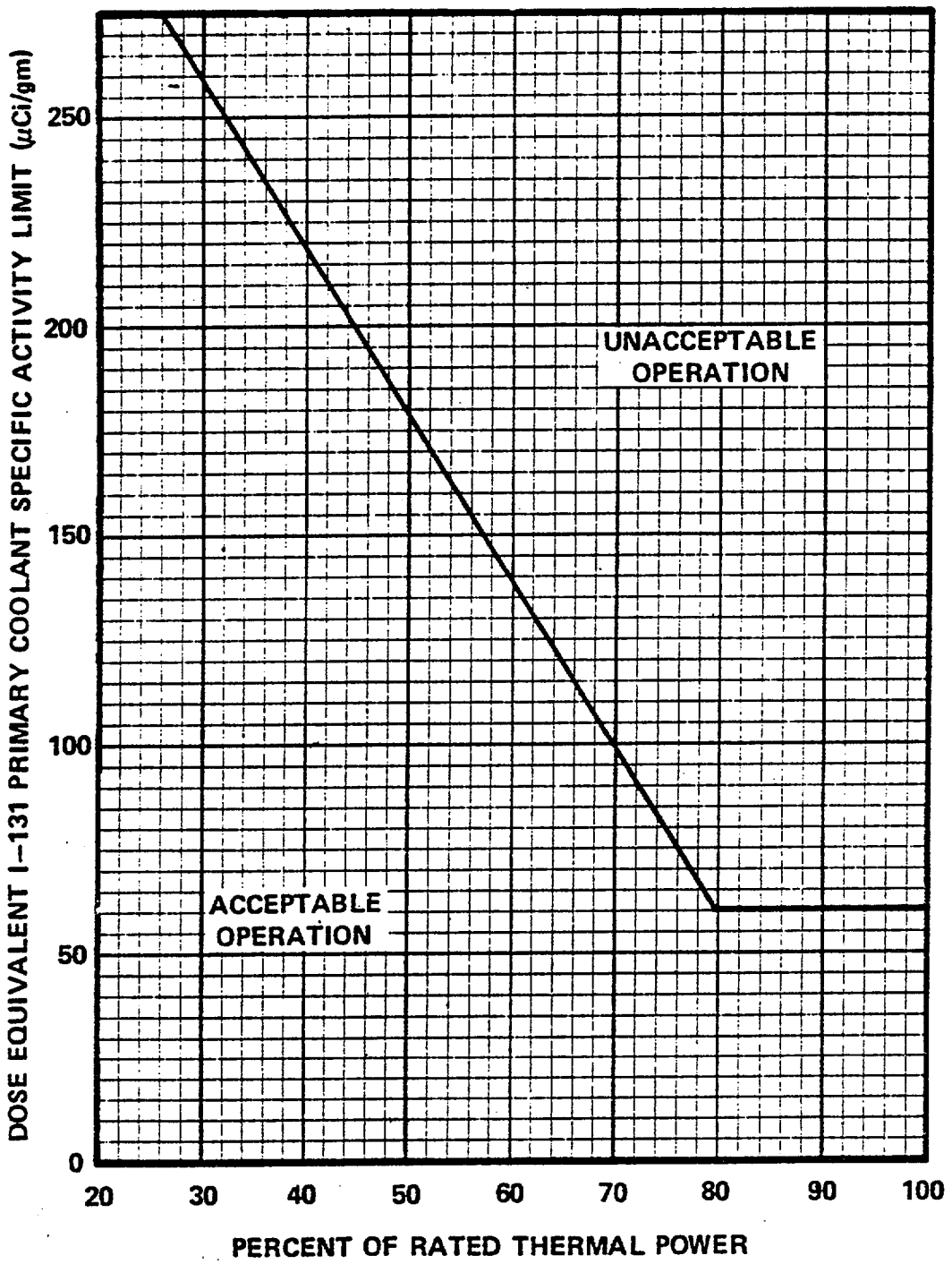


FIGURE 3.4-1

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.

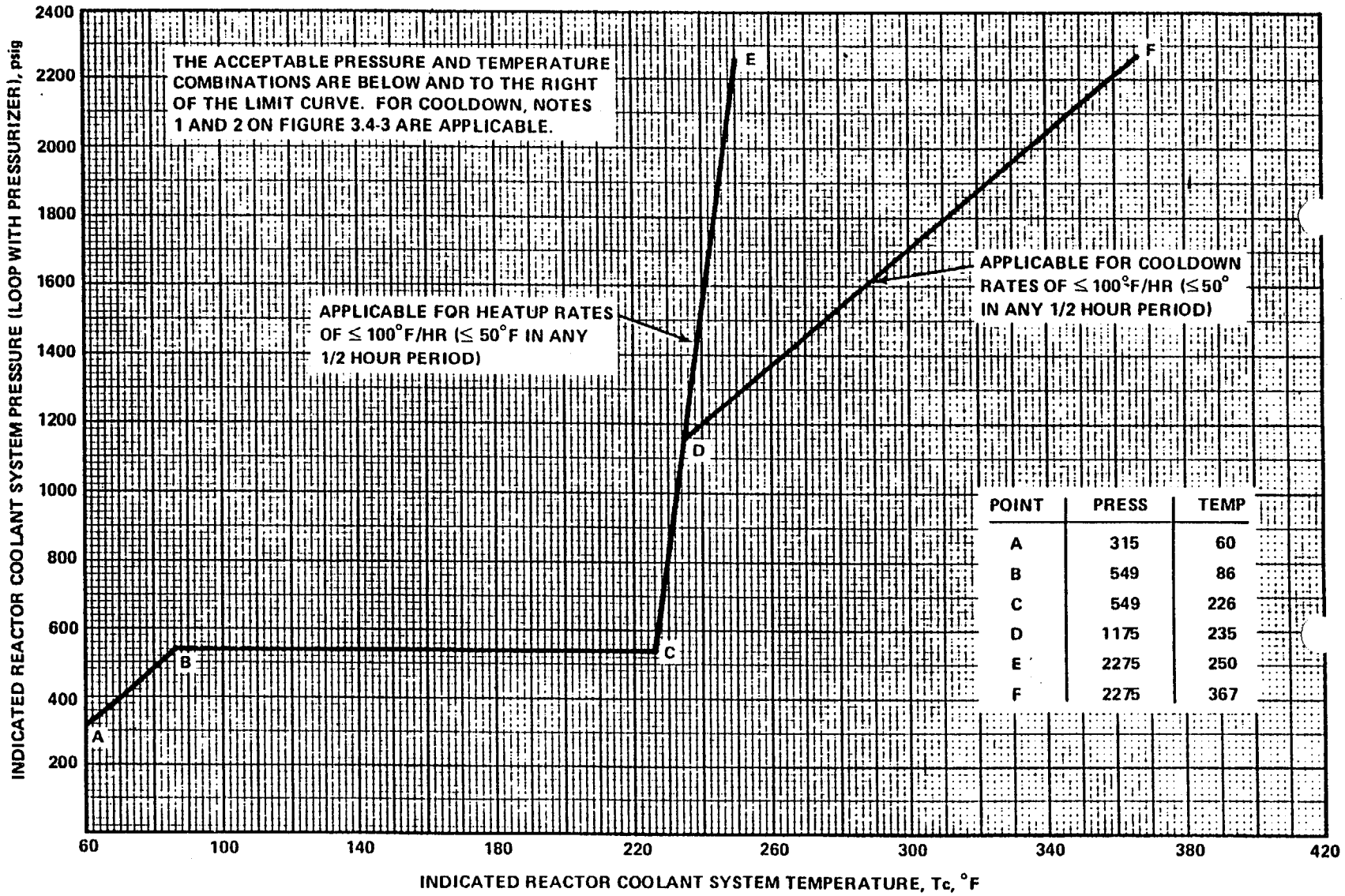
ACTION:

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 3.4-4 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.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO.81 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

Introduction

By letter dated August 27, 1984, the Toledo Edison Company (TED) submitted an application to revise the Davis-Besse Nuclear Power Station, Unit No. 1, Technical Specifications. The application requested a revision to the withdrawal schedule for the reactor vessel material surveillance specimen capsules, deletion of references to specific irradiation capsule locations, and clarification of the surveillance requirement related to the specimens.

Discussion and Evaluation

All light-water nuclear power reactors must meet the fracture toughness requirements and material surveillance requirements for the reactor coolant pressure boundary set forth in Appendices G and H to Part 50 to Title 10, Code of Federal Regulations (see §10 CFR 50.60). Appendix H, "Reactor Vessel Material Surveillance Program Requirements", specifies, among other requirements, that the requirements of ASTM E 185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels", must be met. Appendix H requires also that the proposed surveillance specimen capsule withdrawal schedule be approved prior to its implementation.

ASTM E 185-82 recommendations for the number of surveillance capsules and their withdrawal schedule are based on the predicted transition temperature shift at the reactor vessel inside surface. For reactor vessels with a shift greater than 200°F, the capsule program must include a minimum of five capsules. The ASTM standard prescribes the removal schedule in terms of Effective Full Power Years (EFPY) and end-of-life neutron fluence at various locations through the reactor vessel wall, except for the first capsule to be removed, which is in terms of fluence and predicted temperature shift of the encapsulated materials.

The removal schedule presently specified in Table 4.4-5 of the Davis-Besse, Unit 1, Technical Specifications is based on accumulated fluence of the irradiated capsules but the removal interval is given in terms of operating cycles. However, the relationships between accumulated fluence and operating cycle is not fixed but, rather, can be altered by other operational considerations, two major considerations being revised fuel loading schemes

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and longer operating periods between refuelings. With the beginning of Cycle 5, Toledo Edison Company intends to operate the Davis-Besse station on an 18-month cycle and is planning to utilize a low-leakage core design. Therefore, Toledo Edison Company has proposed a change to Table 4.4-5. The proposed change incorporates the ASTM E 185-82 recommended withdrawal schedule applicable to a program with a minimum required number of capsules of five. The proposed table does not give values for accumulated neutron fluence and does not reference capsule locations within the reactor vessel.

The safety evaluation prepared by Toledo Edison Company states that the specifics of the Davis-Besse surveillance program are contained in the Babcock and Wilcox Owners Group Report, BAW-1543, "Integrated Reactor Vessel Material Surveillance Program". The peak neutron fluence to be received by the Davis-Besse reactor vessel is estimated to be 1.6×10^{19} n/cm^2 (E>1MeV) according to BAW-1834, "Analysis of Capsule TE 1-B, the Toledo Edison Company, Davis-Besse Nuclear Power Station Unit 1 - Reactor Vessel Material Surveillance Program." This report presents the results of examination of the second capsule removed from the Davis-Besse reactor vessel at the end of the third fuel cycle. The remaining surveillance capsules will be irradiated to various neutron fluences, the highest being the estimated peak neutron fluence received by the vessel.

We have compared the expected neutron fluence to be received by each capsule in the surveillance program to that required by ASTM E 185-82 and find that the capsule withdrawal schedule identified in BAW-1543, Rev. 2, meets the intent of the ASTM specification and is acceptable.

Because Cycle 5 marks the start of 18-month operating cycles, the licensee has withdrawn the third capsule at the end of the fourth cycle. The Technical Specifications presently require removal of this capsule at the end of the fifth cycle. The proposed withdrawal schedules submitted by Toledo Edison Company in the August 27, 1984 application do not specifically identify the neutron fluence target which will determine the actual withdrawal time. We find that withdrawal of the third capsule at the end of Cycle 4 meets the neutron fluence target given in BAW-1543, Rev. 2. We also find that the withdrawal times reported for the first two capsules also meet the neutron fluence target. However, the withdrawal times for the remaining capsules are not sufficiently specific to ensure that the target neutron fluences for these capsules, as given in BAW-1543, Rev. 2, will be met. Therefore, we consider the proposed withdrawal schedule for the first three capsules only acceptable. We are withholding a determination on the schedules for the remaining capsules pending clarification by the licensee of the neutron fluence to be received by the capsules. This clarification must be submitted not later than 180 days prior to the refueling outage (transition into mode 5) prior to Cycle 6 operation.

The proposed changes to the Technical Specifications also would delete reference to capsule adder locations. This change would allow a modified capsule shuffle scheme, described in BAW-1543, Rev. 2. We have reviewed this revised shuffle scheme and find it acceptable. We also find that it is unnecessary to include this detail in the Technical Specifications.

Toledo Edison Company proposes to modify Section 4.4.9.1.2 by adding clarification that this surveillance requirements refers to surveillance specimens which are representative of the reactor vessel materials. The Davis-Besse reactor is used to irradiate other material specimens used for research purposes. The proposed change clarifies the scope of this surveillance requirement. We find the clarification acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

CONCLUSION

We have 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.

Dated: December 17, 1984

The following NRC personnel contributed to this Safety Evaluation: Barry Elliot and Albert De Agazio