March 3, 1993

Docket No. 50-237

Mr. Thomas J. Kovach
Nuclear Licensing Manager
Commonwealth Edison Company-Suite 300
OPUS West III
1400 OPUS Place
Downers Grove, Illinois 60515

Dear Mr. Kovach:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M84564)

The Commission has issued the enclosed Amendment No. 123 to Facility Operating License No. DPR-19 for Dresden, Unit 2. The amendment is in response to your application dated September 14, 1992.

The amendment revises the pressure/temperature (P/T) limits in Section 3.6 of the Technical Specifications (TS) to correct a deficiency in the P/T limits currently in the TS identified by you.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original Signed By:

John F. Stang, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosure:
1. Amendment No. 123 to DPR-19
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION:	Docket File	NRC & Local PDRs	J. Stang
J. Roe	J. Zwolinski	J. Dyer	C. Moore
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OC/LFDCB	PDIII-2 p/f	B. Clayton RIII	

* Please see previous concurrence

OFC	LA:PDIII-2	PM:PDIII-2	D:PDIII-2	OGC*	
NAME	CMOORE	JSTANG	JDYER	CPW	
DATE	3/3/93	3/3/93	3/4/93	02/16/93	

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P PDR

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

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NAME	CMOORE	JSTANG	JDYER	CPW	
DATE	3/3/93	3/3/93	3/4/93	02/16/93	

Mr. Thomas J. Kovach
Commonwealth Edison Company

Dresden Nuclear Power Station
Unit Nos. 2 and 3

cc:

Michael I. Miller, Esquire
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U. S. Nuclear Regulatory Commission
Resident Inspectors Office
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Chairman
Board of Supervisors of
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A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Byron L. Siegel, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosure:
1. Amendment No. to DPR-19
2. Safety Evaluation

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OFC	LA:PDIII-2	PM:PDIII-2	D:PDIII-2	OGC	
NAME	CMOORE	BSIEGEL for	JDYER	OPW	
DATE	/ /93	2/ /93	/ /93	2/1/93	

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 123
License No. DPR-19

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated September 14, 1992, 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. DPR-19 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 123 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James E. Dyer

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 3, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 123

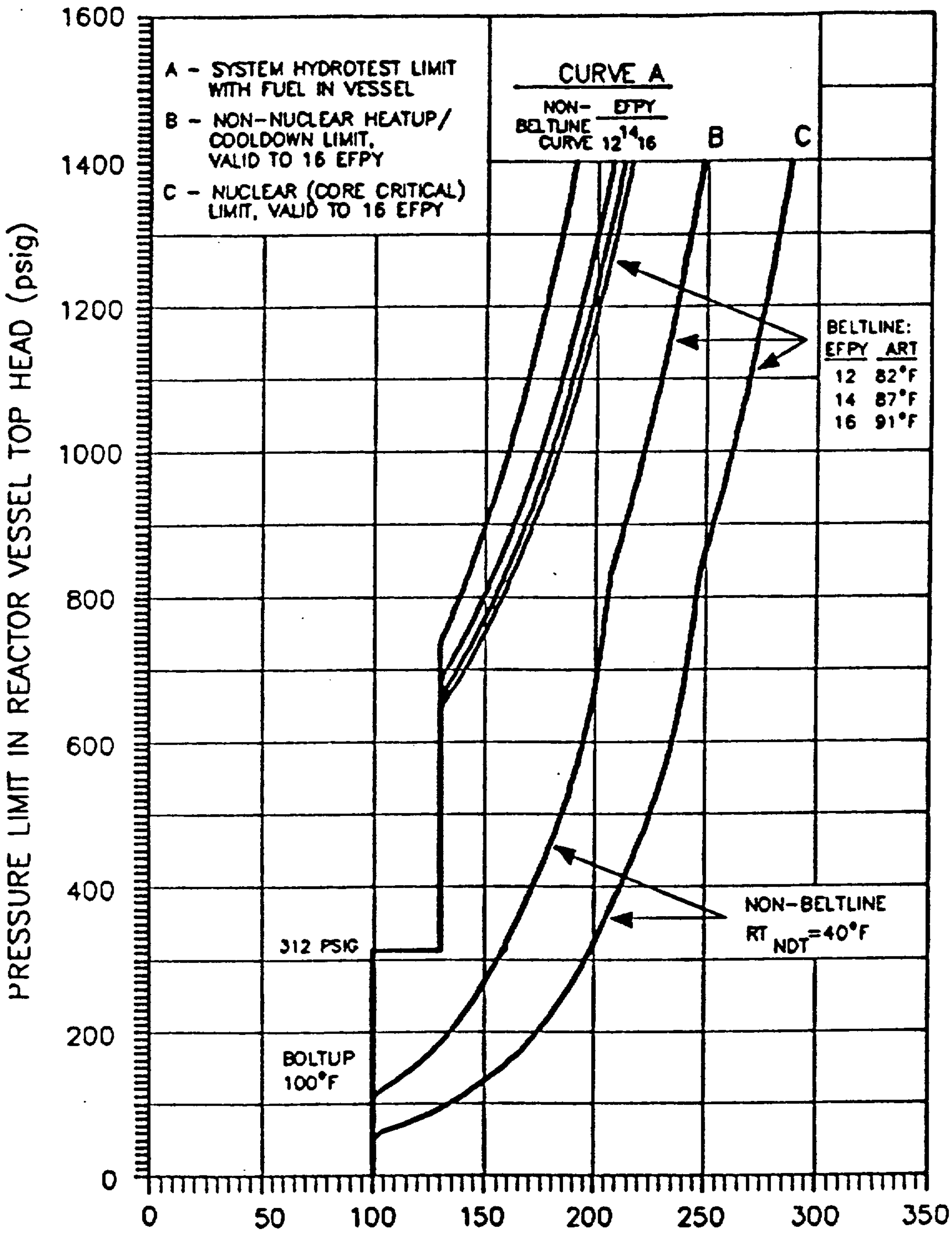
FACILITY OPERATING LICENSE NO. DPR-19

DOCKET NO. 50-237

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
3/4.6-23	3/4.6-23
B 3/4.6-26	B 3/4.6-26
B 3/4.6-26a	B 3/4.6-26a

MINIMUM REACTOR VESSEL METAL TEMPERATURE (°F)



MINIMUM REACTOR VESSEL METAL TEMPERATURE (°F)

FIGURE 3.6.1.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

B. Pressurization Temperature - The reactor vessel 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, pressure-temperature limits have been established for the operating conditions to which the reactor vessel can be subjected. Figure 3.6.1 presents the pressure-temperature curves for those operating conditions; Inservice Hydrostatic Testing (Curve A), Non-Nuclear Heatup/Cooldown (Curve B), and Core Critical Operation (Curve C). These curves have been established to be in conformance with Appendix G to 10 CFR 50 and Regulatory Guide 1.99, Revision 2, and take into account the change in reference nil-ductility transition temperature (RT_{NDT}) as a result of neutron embrittlement. The adjusted reference temperature (ART) of the limiting vessel material is used to account for irradiation effects.

Three vessel regions are considered for the development of the pressure-temperature curves: 1) the core beltline region; 2) the non-beltline region (other than the closure flange region); and 3) the closure flange region. The beltline region is defined as that region of the reactor vessel that directly surrounds the effective height of the reactor core (between the bottom and the top of active fuel), and is subject to an RT_{NDT} adjustment to account for irradiation embrittlement. The non-beltline and closure flange regions receive insufficient fluence to necessitate an RT_{NDT} adjustment. These regions contain components which include; the reactor vessel nozzles, closure flanges, top and bottom head plates, control rod drive penetrations, and shell plates that do not directly surround the reactor core. Although the closure flange region is a non-beltline region, it (the closure flange region) is treated separately for the development of the pressure-temperature curves to address 10 CFR 50 Appendix G requirements.

In evaluating the adequacy of the steel which comprises the reactor vessel, it is necessary that the following be established: 1) the RT_{NDT} for all vessel and adjoining materials; 2) the relationship between RT_{NDT} and integrated neutron flux (fluence, at energies greater than one Mev); and 3) the fluence at the location of a postulated flaw.

Boltup Temperature

The initial RT_{NDT} of the main closure flanges, the shell and head materials connecting to these flanges, the connecting welds and the vertical electroslag welds which terminate immediately below the vessel flange have an RT_{NDT} of 40°F. Therefore, the minimum allowable boltup temperature is established as 100°F (RT_{NDT} + 60°F) which includes a 60°F conservatism required by the original ASME Code of construction.

Curve A - Hydrotesting

As indicated in Curve A of Figure 3.6.1 for system hydrotesting, the minimum metal temperature of the reactor vessel shell is 100°F for reactor pressures less than 312 psig. This 100°F minimum boltup temperature is based on an RT_{NDT} of 40°F for the electroslag weld immediately below the vessel flange and a 60°F conservatism required by the original ASME Code of construction.

At reactor pressures greater than 312 psig the minimum vessel metal temperature is established as 130°F. The 130°F minimum temperature is based on a closure flange region RT_{NDT} of 40°F and a 90°F conservatism required by 10 CFR 50 Appendix G for pressure in excess of 20% of the preservice hydrostatic test pressure (1563 psig).

At approximately 650 psig the effects of pressurization are more limiting than the boltup stresses at the closure flange region, hence a family of non-linear curves intersect the 130°F vertical line. Belt-line as well as non-beltline curves have been provided to allow separate monitoring of the two regions. Beltline curves as a function of vessel exposure for 12, 14, and 16 effective full power years (EFPY) are presented to allow the use of the appropriate curve up to 16 EFPY of operation.

Curve B - Non-Nuclear Heatup/Cooldown

Curve B of Figure 3.6.1 applies during heatups with non-nuclear heat (e.g., recirculation pump heat) and during cooldowns when the reactor is not critical (e.g., following a scram). The curve provides the minimum reactor vessel metal temperatures based on the most limiting vessel stress.

As indicated by the vertical 100°F line, the boltup stresses at the closure flange region are most limiting for reactor pressures below approximately 110 psig. For reactor pressures greater than approximately 110 psig, pressurization and thermal stresses become more limiting than the boltup stresses, which is reflected by the non-linear portion of Curve B. The non-linear portion of the curve is dependent on non-beltline and beltline regions, with the beltline region temperature limits having been adjusted to account for vessel irradiation (up to a vessel exposure of 16 EFPY). The non-beltline region is limiting between approximately 110 psig and 830 psig. Above approximately 830 psig, the beltline region becomes limiting.

Curve C - Core Critical Operation

Curve C, the core critical operation curve shown in Figure 3.6.1, is generated in accordance with 10 CFR 50 Appendix G which requires core critical pressure-temperature limits to be 40°F above any Curve A or B limits. Since Curve B is more limiting, Curve C is Curve B plus 40°F.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 123 TO FACILITY OPERATING LICENSE NO. DPR-19

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNIT 2

DOCKET NO. 50-237

1.0 INTRODUCTION

On September 14, 1992, the Commonwealth Edison Company (CECo, the licensee) requested permission to revise the pressure/temperature (P/T) limits in Section 3.6 of the Dresden Nuclear Power Station, Unit 2, Technical Specifications (TS). The P/T limits were requested for 16 effective full power years (EFPY). As of July 1, 1992, Dresden, Unit 2, has operated to about 12 EFPY.

On July 2, 1992, the licensee informed the NRC that the P/T limits in the Dresden, Unit 2, TS require revision. This was determined as a result of the licensee's review of the reactor vessel material data in response to Generic Letter (GL) 92-01. The licensee's evaluation using Appendix E to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) determined that the reactor vessel has adequate safety margin against brittle fracture; however, as an interim measure, the licensee has used the more conservative P/T limits of Dresden Unit 3 for Dresden Unit 2 operation.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the American Society for Testing and Materials (ASTM) Standards, and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); Regulatory Guide (RG) 1.99, Revision 2; Standard Review Plan (SRP) Section 5.3.2; and GL 88-11. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," recommends RG 1.99, Revision 2, be used in calculating P/T limits, unless the use of different methods can be justified.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide TS for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions for operation be included in the TS. The P/T limits are among the limiting conditions for operation in the TS for all commercial nuclear plants in the U.S. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART). Generic Letter 88-11 requested that licensees use the methods in RG 1.99, Revision 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

2.0 EVALUATION

The licensee selected properties among all beltline materials in Dresden and Quad Cities reactors that would give the highest ART. The licensee used (1) the chemical composition of longitudinal seam weld, PQ-1300, in Dresden Unit 3, (2) the initial nil ductility transition reference temperature (RT_{ndt}) of the longitudinal weld in Dresden Unit 2 (Reference 1), and (3) the predicted neutron fluence of Dresden Unit 3 at 16 EFPY. The result of this conservative approach is that the limiting material, from which the proposed P/T limits were constructed, has 0.3% copper, 0.33% nickel, and an initial RT_{ndt} of 40 °F. The neutron fluence used was 1.8E17 n/cm2 at 1/4T (T = reactor vessel beltline thickness). The licensee calculated a limiting ART of 91 °F at 1/4T which the staff confirmed to be correct using the RG 1.99 method.

Besides reviewing the licensee's ART calculations, the staff also calculated the ART for each beltline material in the Dresden Unit 2 reactor vessel using the material data in the licensee's surveillance reports and FSAR. The staff determined that the highest ART is 59.5 °F at 1/4T based on a neutron fluence of 1.2E17 n/cm2 at 16 EFPY (Reference 8). The limiting beltline material was the lower intermediate shell, B4065-1, with 0.23% copper, 0.52% nickel, and an initial RT_{ndt} of 20 °F.

The licensee's ART of 91 °F is more conservative than the staff's ART of 59.5 °F and is acceptable. Substituting the ART of 91 °F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet paragraph IV.A.2 of Appendix G to 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Paragraph IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120 °F for normal operation and by 90 °F for hydrostatic pressure tests and leak tests. Paragraph IV.A.3 of Appendix G states "an exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the pre-service system hydrostatic test pressure. In this case the minimum permissible temperature is 60 °F (33 °C) above the reference temperature of the closure flange regions that are highly stressed by the bolt preload." Based on the flange reference temperature of 20 °F, the staff has determined that the proposed P/T limits satisfy paragraph IV.A.3 of Appendix G.

The staff concludes that the proposed P/T limits for heatup, cooldown, leak test, and criticality are valid through 16 EFPY because the limits conform to paragraphs IV.A.2, IV.A.3, & IV.A.4 of Appendix G to 10 CFR Part 50. The proposed P/T limits also satisfy GL 88-11 because the licensee used the method in RG 1.99, Revision 2, to construct the limits. Hence, the proposed P/T limits may be incorporated into the Dresden, Unit 2, TS.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has 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 the amendment involves no significant hazards consideration, and there has been no public comment on such finding (57 FR 55578). Accordingly, the 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 the amendment.

5.0 CONCLUSION

The Commission 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: John Tsao

Date: March 3, 1993

6.0 REFERENCES

1. Appendix F of Dresden, Unit 2, Final Safety Analysis Report.

2. NUREG-0800, Standard Review Plan, Section 5.3.2: Pressure-Temperature Limits.

3. Letter from P. L. Piet (CECo) to T. E. Murley (USNRC), subject: Application to Amendment to Facility Operating License DPR-19, Appendix A, Technical Specifications; Proposed Amendment to Figure 3.6.1, "Minimum Reactor Vessel Metal Temperature," September 14, 1992.

4. Letter from R. Stols (CECo) to T. E. Murley (USNRC), Subject: Application for Amendment to Facility Operating Licenses DPR-19, DPR-25, DPR-29, and DPR-30, October 10, 1989.

5. Letter from R. Stols (CECo) to T. E. Murley (USNRC), Subject: Application for Amendment to Facility Operating Licenses DPR-19, DPR-25, DPR-29, and DPR-30, October 23, 1989.

6. Letter from R. Stols (CECo) to T. E. Murley (USNRC), Subject: Response to Request for Additional Information, March 23, 1990.

7. G. F. Rieger and G. H. Henderson, "Dresden Nuclear Power Station Unit One and Unit Two, Mechanical Properties of Irradiated Reactor Vessel Material Surveillance Specimens," NEDC-12585, May 1975.

8. E. B. Norris, "Dresden Nuclear Power Station Unit 2 Reactor Vessel Irradiation Surveillance Program, Analysis of Capsule 8," SWRI 06-6901-002, March 1983.

9. E. O. Fromm, et al., "Dresden Nuclear Plant Reactor Pressure Vessel Surveillance Program: Unit No. 2 Capsule Basket Assembly No. 5," BCL-585-10, May 8, 1979.

10. J. S. Perrin, et al., "Dresden Nuclear Plant Reactor Pressure Vessel Surveillance Program: Unit No. 2 Neutron Dosimeter Monitor, Unit No. 2 Capsule Basket Assembly No. 2, and Unit No. 3 Capsule Basket Assembly No. 12," BCL-585-3, September 15, 1977.