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**NewYorkPower Authority**

**James Knubel**
Senior Vice President and
Chief Nuclear Officer

April 27, 2000
IPN-00-033

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: **Indian Point 3 Nuclear Power Plant**
**Docket No. 50-286**
**Proposed Technical Specification Changes Associated With Pressure-Temperature and Overpressure Protection System Limits for Up To 16.2 Effective Full Power Years**

References:

1. NRC letter, G. Wunder to J. Knubel, "Issuance of Amendment for Indian Point Nuclear Generating Unit No. 3 (TAC No. M99928)," (Amendment 179), dated April 10, 1998.

2. NRC letter, G. Wunder to J. Knubel, "Request for Additional Information Regarding Proposed Changes to Pressure and Temperature (P/T) Curves (TAC No. M99928)," dated January 14, 1998.

3. NYPA letter, J. Knubel to NRC (IPN-98-015), "Response to Request for Additional Information on Proposed Technical Specification Changes Regarding Pressure-Temperature Limits," dated February 6, 1998.

Dear Sir:

This amendment request seeks to revise Sections 3.1 and 4.3 of the Indian Point 3 Technical Specifications. Specifically, the Pressure-Temperature (P/T) and Overpressure Protection System (OPS) limits are being revised from 13.3 Effective Full Power Years (EFPYs) to 16.2 EFPYs. The current limit of 13.3 EFPYs was approved by the NRC in Amendment 179 (Reference 1).

The fluence value calculations, used for the basis of the revised P/T curves issued by Amendment 179, were performed using the ENDF/B-IV and ENDF/B-V cross section databases. The NRC stated in Reference 2 that the use of these databases could possibly result in an underestimation of fluence values by as much as 15%. Therefore, the Authority added a 15% penalty to the lifetime fluence values used to calculate the current P/T curves to compensate for this potential underestimation (Reference 3). In order to eliminate the fluence

A001

penalty currently incurred, the Authority contracted Westinghouse to recalculate the fluence values for Indian Point 3 using the ENDF/B-VI cross section database. The results of the calculations using ENDF/B-VI (Attachment IV) confirm that the fluence values used for Amendment 179 contained an extra 15% of conservatism. Further, this calculation specifically modeled the effect of the low-leakage cores utilized at IP3. Therefore, the use of the newly calculated fluence values allows the Authority to extend the existing Technical Specification P/T curves from 13.3 EFPYs to 16.2 EFPYs.

Enclosed for filing is the signed original of a document entitled, "Application for Amendment to Operating License," together with Attachments I and II, comprising a statement of the proposed changes to the Technical Specifications and the associated Safety Evaluation. Attachment III contains a markup of the revised TS pages (for information only). Attachment V includes the Authority's calculation of the 16.2 EFPY expiration. This calculation has been independently reviewed and concurred with by the reactor vessel manufacturer, ABB-Combustion Engineering.

This change will be incorporated into sections 3.4.3 and 3.4.12 of the Improved Technical Specifications. The same figures will be included in 3.4.3 and 3.4.12 to reflect the new validity period as 16.2 EFPYs.

In accordance with 10 CFR 50.91, a copy of this revised application and the associated attachments are being submitted to the designated New York State official.

Indian Point 3 is currently operating with Technical Specification requirements which are valid up to 13.3 EFPYs. Indian Point 3 is expected to exceed 13.3 EFPYs at approximately October 26, 2000. Therefore, approval of this amendment request is respectfully requested by September 26, 2000 to allow time for amendment implementation. If you have any questions, please contact Ms. C. D. Faison.

Very truly yours,

J. Knubel
Senior Vice President and
Chief Nuclear Officer

Attachments: next page

Attachments:

Application for Amendment to Operating License

I. Proposed Revisions to Technical Specification Changes Associated with Pressure-Temperature and Overpressure Protection System Limits for up to 16.2 Effective Full Power Years

II. Safety Evaluation of Proposed Technical Specification Changes Associated with Pressure-Temperature and Overpressure Protection System Limits for Up to 16.2 Effective Full Power Years

III. Mark-up of Revised Technical Specification Pages Associated with Amendment Request Regarding Pressure-Temperature and Overpressure Protection System Limits for Up to 16.2 Effective Full Power Years (For Information Only)

IV. Westinghouse Report

V. NYPA Calculation

cc: Regional Administrator
U.S. Nuclear Regulatory Commission
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King of Prussia, PA 19406

Resident Inspector's Office
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BEFORE THE UNITED STATES
NUCLEAR REGULATORY COMMISSION

| | | |
|---|---|---|
| In the Matter of | ) | |
| POWER AUTHORITY OF THE STATE OF NEW YORK | ) | Docket No. 50-286 |
| Indian Point 3 Nuclear Power Plant | ) | |

**APPLICATION FOR AMENDMENT TO OPERATING LICENSE**

Pursuant to Section 50.90 of the regulations of the Nuclear Regulatory Commission (NRC), the Power Authority of the State of New York, as holder of Facility Operating License No. DPR-64, hereby applies for an Amendment to the Technical Specifications contained in Appendix A of the license.

This amendment request seeks to revise Sections 3.1 and 4.3 of the Technical Specifications. Specifically, the Pressure-Temperature (P/T) and Overpressure Protection System (OPS) limits are being revised from 13.3 Effective Full Power Years (EFPYs) to 16.2 EFPYs. The current limit of 13.3 EFPYs was approved by the NRC in Amendment 179.

The proposed changes to the Technical Specifications are included as Attachment I to this application. The Safety Evaluation is included as Attachment II.

POWER AUTHORITY OF THE
STATE OF NEW YORK

By
J. Knubel
Senior Vice President and
Chief Nuclear Officer

STATE OF NEW YORK
COUNTY OF WESTCHESTER

Subscribed and Sworn to before me
this 27th day of April 2000.

Notary Public

EILEEN E. O'CONNOR
Notary Public, State of New York
No. 4991062
Qualified in Westchester County
Commission Expires January 21, 2002

Attachment I to IPN-00-033

# PROPOSED REVISIONS TO TECHNICAL SPECIFICATION CHANGES ASSOCIATED WITH PRESSURE-TEMPERATURE AND OVERPRESSURE PROTECTION SYSTEM LIMITS FOR UP TO 16.2 EFFECTIVE FULL POWER YEARS

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

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| Secondary Side Limitations for RCP Start With Secondary Side Hotter Than Primary, 16.2 EFPY | 3.1.A-1 |
| Maximum Allowable Nominal PORV Setpoint for the Low Temperature Overpressure Protection System (OPS), 16.2 EFPY | 3.1.A-2 |
| Deleted | 3.1.A-3 |
| Deleted | 3.1.A-4 |
| Pressurizer Limitations for OPS Inoperable (Up to one charging pump capable of feeding RCS), 16.2 EFPY | 3.1.A-5 |
| Pressurizer Limitations for OPS Inoperable (up to three charging pumps and/or one safety injection pump capable of feeding RCS), 16.2 EFPY | 3.1.A-6 |
| Reactor Coolant System Heatup Limitations | 3.1-1 |
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| Primary Coolant Specific Activity Limit vs. Percent of Rated Thermal Power | 3.1-3 |
| Gross Electrical Output - 1" HG Backpressure | 3.4-1 |
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| Spent Fuel Pit Region 1 Fuel Type Definition | 3.8-1 |
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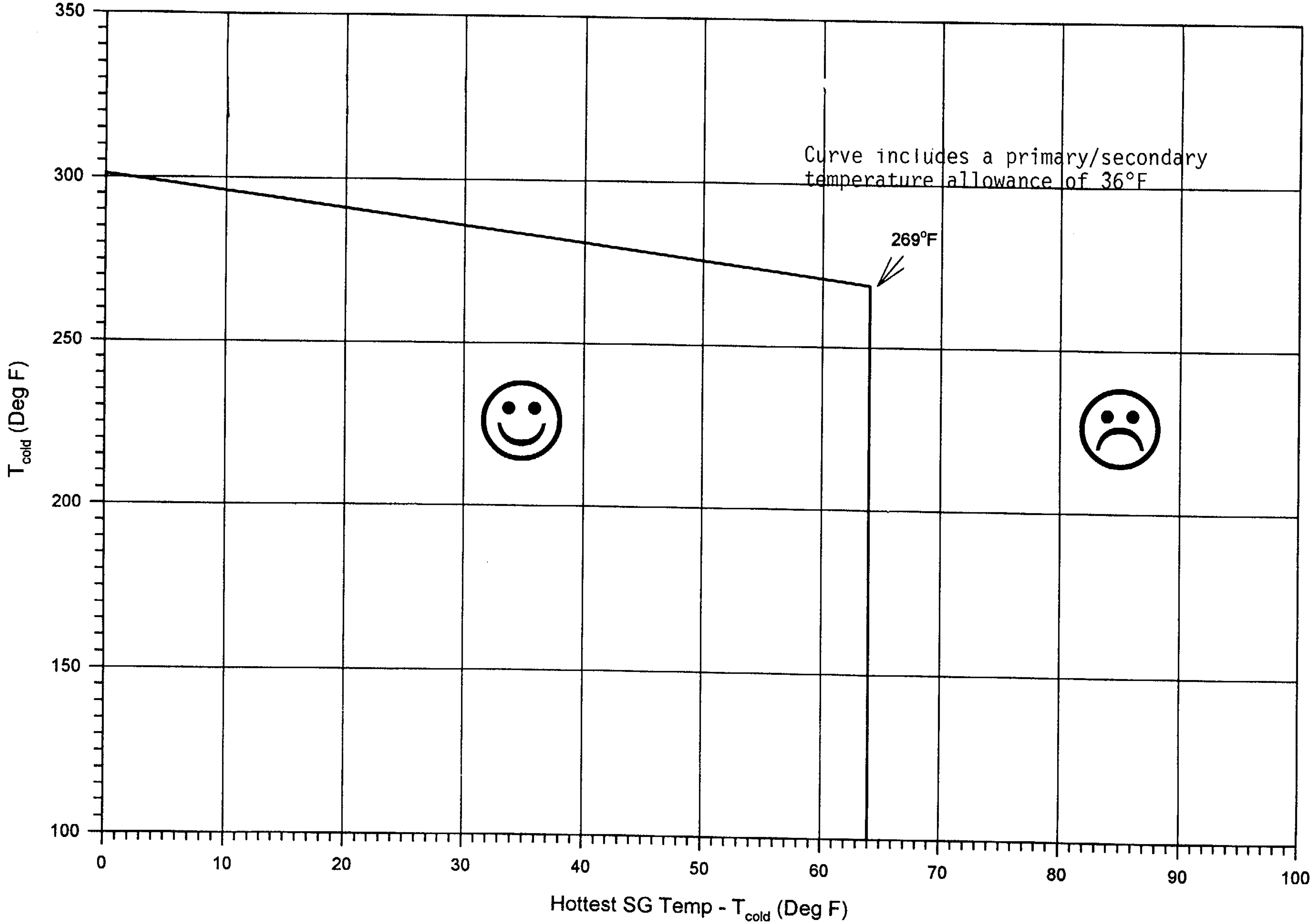
| RCS Limitations for Hydrostatic Testing, 16.2 EFPY | 4.3-1 |

FIGURE 3.1.A-1

Secondary Side Limitations for RCP Start With Secondary Side Hotter Than Primary, 16.2 EFPY

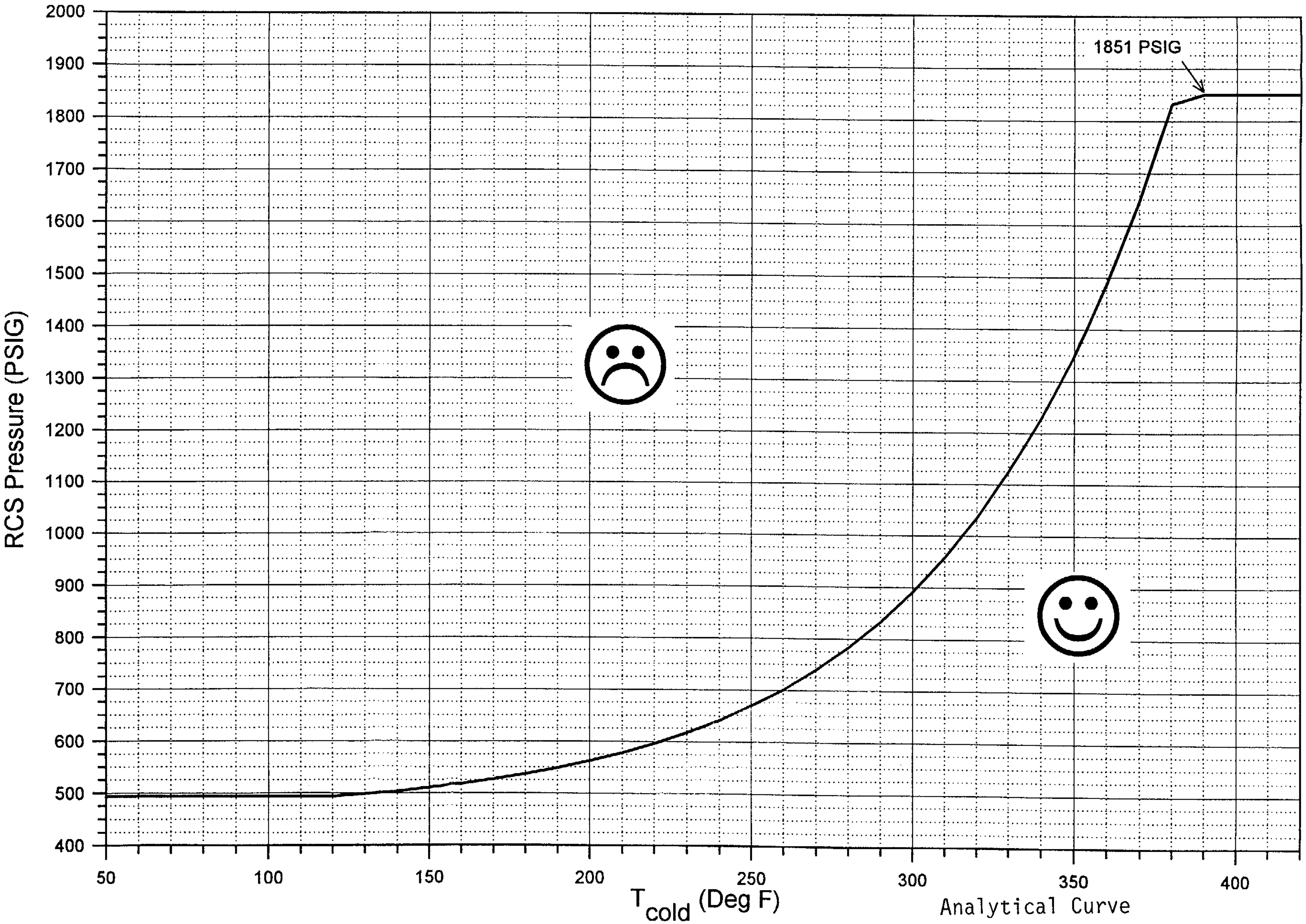





Amendment No. ~~67, 121, 179,~~

FIGURE 3.1.A-2

Maximum Allowable Nominal PORV Setpoint for the Low Temperature Overpressure Protection System (OPS), 16.2 EFPY




Analytical Curve



Amendment No. ~~67, 121, 179,~~

FIGURE 3.1.A-5

Pressurizer Limitations for OPS Inoperable (Up to one charging pump capable of feeding RCS), 16.2 EFPY

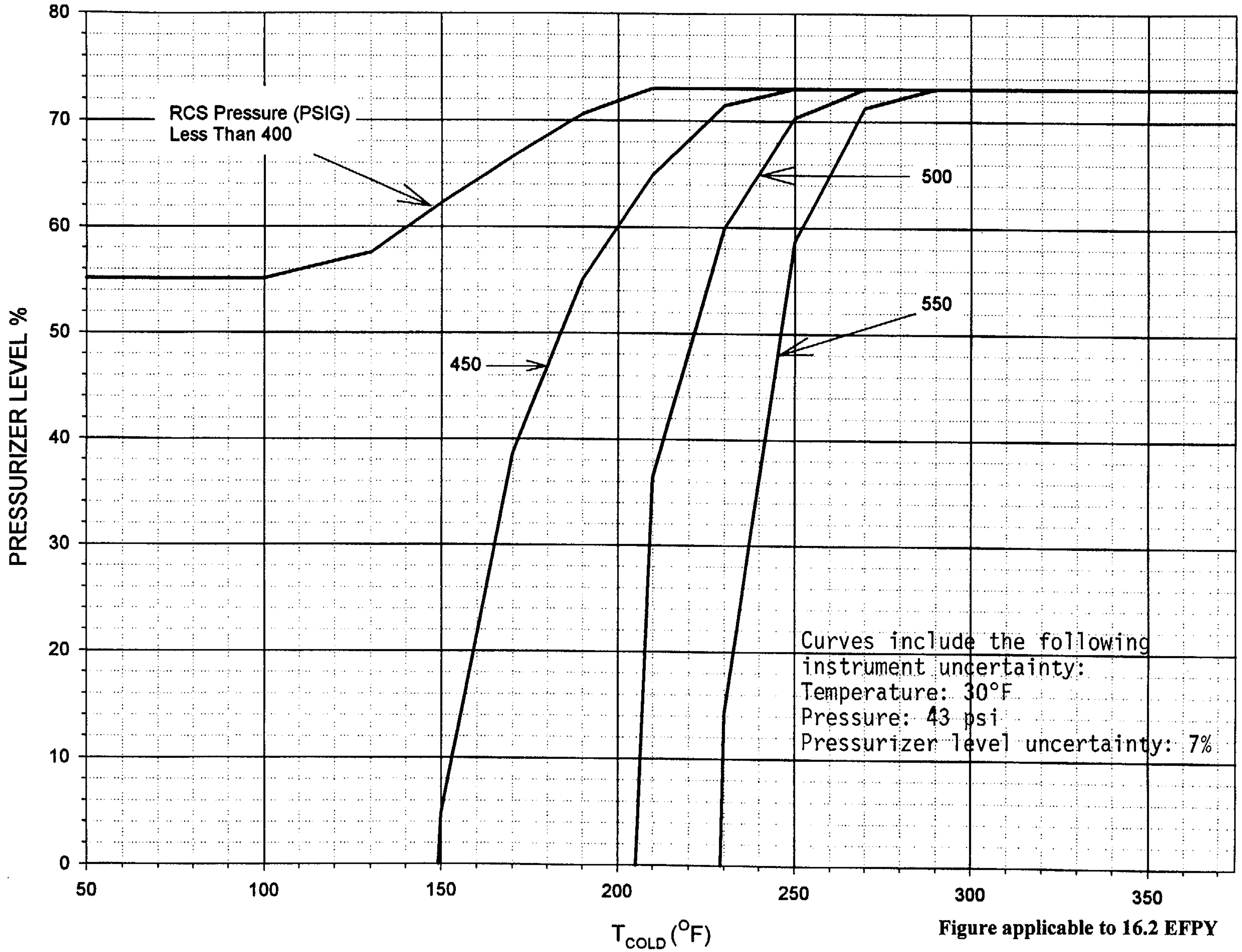



**Figure applicable to 16.2 EFPY**

Curves represent maximum allowable pressurizer level for the conditions defined.



Amendment No. ~~67, 101, 121, 179,~~

FIGURE 3.1.A-6

Pressurizer Limitations for OPS Inoperable
(up to three charging pumps and/or one safety injection pump capable of feeding RCS), 16.2 EFPY

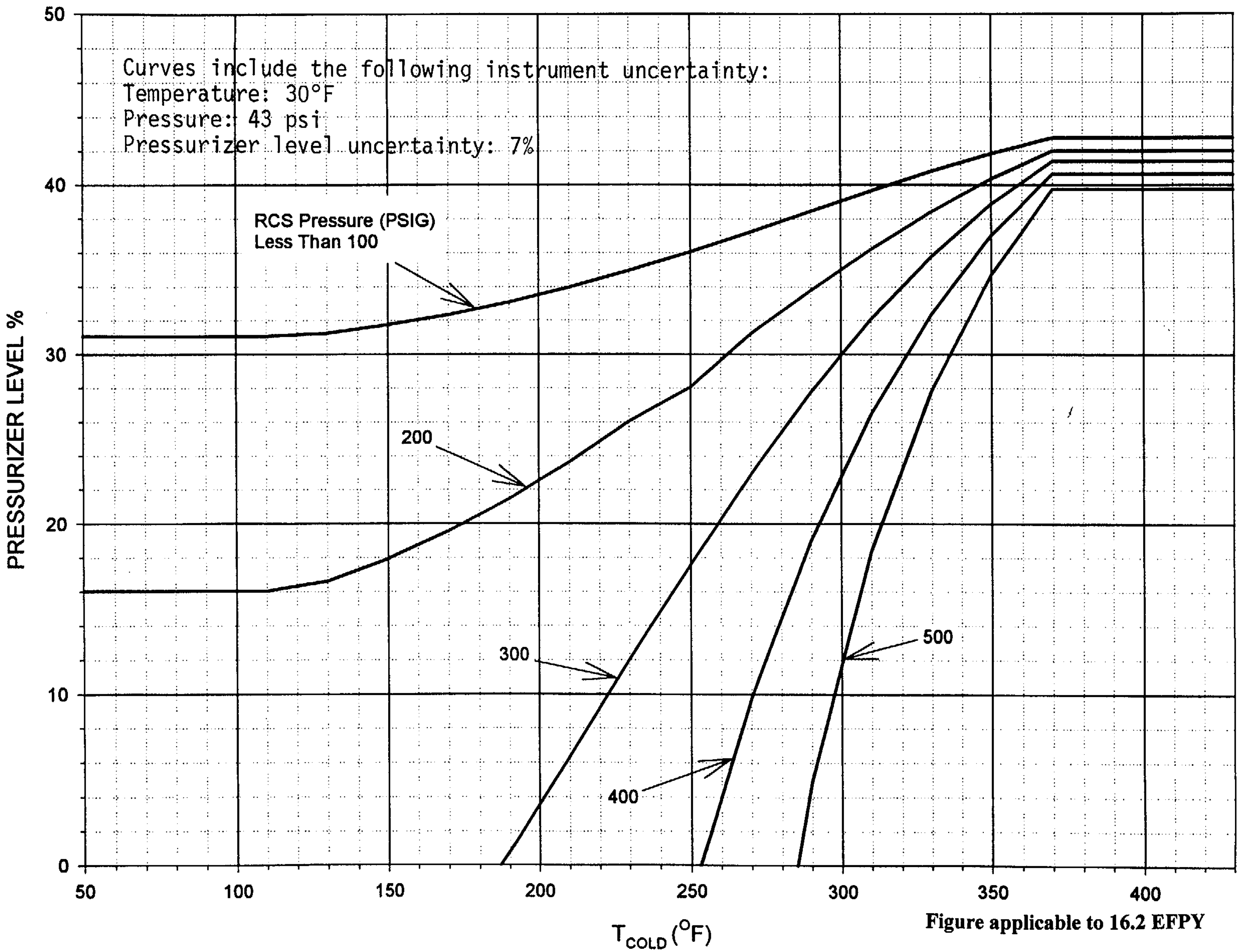



**Figure applicable to 16.2 EFPY**

Curves represent maximum allowable pressurizer level for the conditions defined.



Amendment No. ~~67, 10, 121, 179,~~

B. HEATUP AND COOLDOWN

Specifications

1. The reactor coolant temperature and pressure and system heatup and cooldown rates averaged over one hour (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2 for the service period up to 16.2 effective full-power years (EFPYs). The heatup and cooldown rates shall not exceed 60°F/hr and 100°F/hr respectively.

    a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

2. The limit lines shown in Figure 3.1-1 and Figure 3.1-2 shall be recalculated periodically using methods discussed in the Basis and results of surveillance specimens as covered in Specification 4.2. The order of specimen removal may be modified based on the results of testing of previously removed specimens.

3. The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

4. The pressurizer heatup and cooldown rates averaged over one hour shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

5. Reactor Coolant System integrity tests shall be performed in accordance with Section 4.3.

Basis

Fracture Toughness Properties

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the Summer 1965 Section III of the ASME Boiler and Pressure Vessel Code [6] and ASTM E185 [5] and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1972 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code [1], and the calculation methods described in WCAP-7924 [2].

The first reactor vessel material surveillance capsule was removed during the 1978 refueling outage. This capsule has been tested by Westinghouse Corporation and the results have been evaluated and reported [7]. Similar reports were prepared for the surveillance capsules [10, 8] removed in 1982 and 1987. Based on the Westinghouse evaluation, heatup and cooldown curves (Figures 3.1-1 and 3.1-2) were developed for up to 16.2 EFPYs of reactor operation.

Generic Letter 88-11 requested that licensees use the methodology of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", to predict the effect of neutron radiation on reactor vessel materials as required by paragraph V.A. of 10 CFR part 50, Appendix G. Capsule Z was analyzed [8] and new pressure-temperature curves were developed using this methodology.

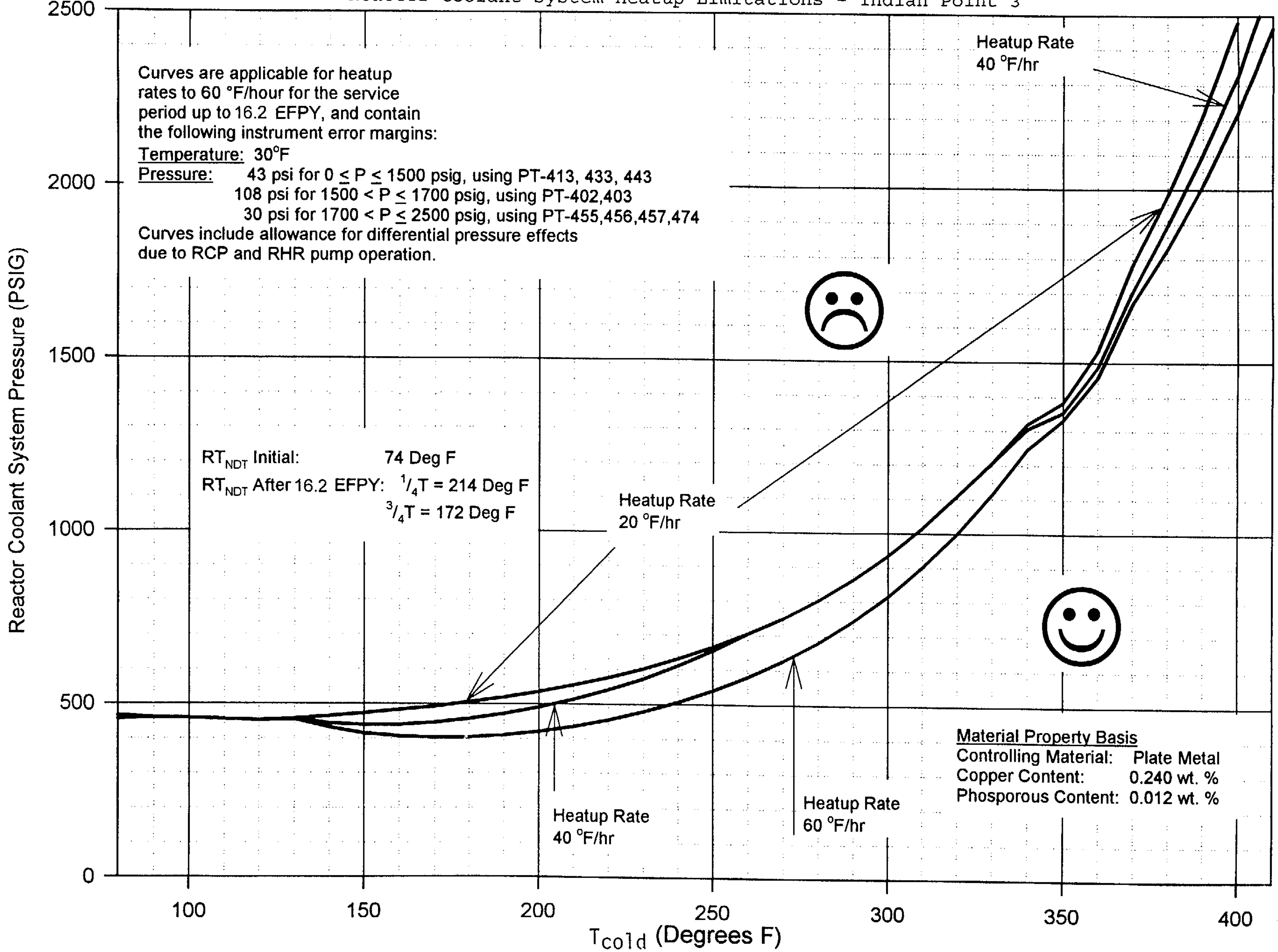
The maximum value in $RT_{NDT}$ after 16.2 EFPYs of operation is projected to be $214^0$F at the 1/4 T and $172^0$F at the 3/4 T vessel wall locations for Plate B2803-3 the controlling plate. Plate B2803-3 was also the controlling plate for the operating period up to 13.3 EFPYs.

Heatup and cooldown limit curves are calculated using the most limiting value of $RT_{NDT}$ at the end of 16.2 years of service life. The 16.2 year service life period is chosen such that the limiting $RT_{NDT}$ at the 1/4 T location in the core region is higher than the $RT_{NDT}$ of the limiting unirradiated material. This service period assures that all components in the Reactor Coolant System will be operated conservatively in accordance with Code recommendations.

The highest $RT_{NDT}$ of the core region material is determined by adding the radiation induced $\Delta RT_{NDT}$ for the applicable time period to the original $RT_{NDT}$ shown in Table Q4.2-1 [3].

FIGURE 3.1-1

Reactor Coolant System Heatup Limitations - Indian Point 3

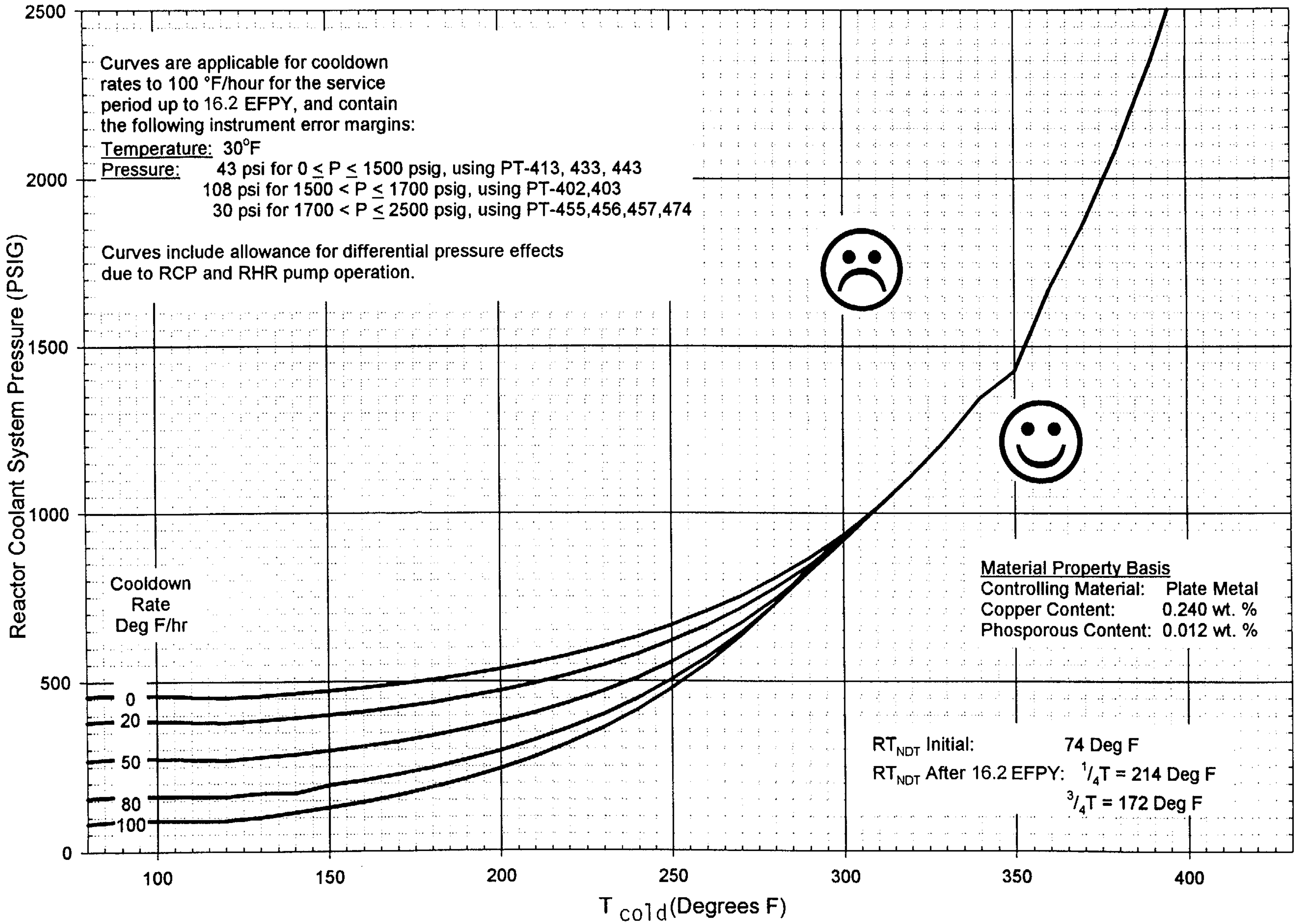





Amendment No. ~~28, 109, 121, 179,~~

FIGURE 3.1-2

Reactor Coolant System Cooldown Limitations - Indian Point 3






Amendment No. ~~28, 109, 121, 179,~~

## 4.3 REACTOR COOLANT SYSTEM (RCS) TESTING

A. Reactor Coolant System Integrity Testing

Applicability

Applies to test requirements for Reactor Coolant System integrity.

Objective

To specify tests for Reactor Coolant System integrity after the system is closed following refueling, repair, replacement or modification.

Specification

a) The Reactor Coolant System shall be tested for leakage at normal operating pressure prior to plant startup following each refueling outage, in accordance with the requirements of ASME Section XI.

b) Testing of repairs, replacements or modifications for the Reactor Coolant System shall meet the requirements of ASME Section XI.

c) The Reactor Coolant System leak test temperature-pressure relationship shall be in accordance with the limits of Figure 4.3-1 for heatup for the first 16.2 EFPYs of operations. Figure 4.3-1 will be recalculated periodically. Allowable pressures during cooldown from the leak test temperature shall be in accordance with Figure 3.1-2.

Basis

Leak test of the Reactor Coolant System is required by the ASME Boiler and Pressure Vessel Code, Section XI, to ensure leak tightness of the system during operation. The test frequency and conditions are specified in the Code.

For repairs on components, the thorough non-destructive testing gives a very high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. In all cases, the leak test will assure leak tightness during normal operation.

The inservice leak test temperatures are shown on Figure 4.3-1. The temperatures are calculated in accordance with ASME Code Section III, Appendix G. This Code requires that a safety factor of 1.5 times the stress intensity factor caused by pressure be applied to the calculation.

For the first 16.2 effective full power years, it is predicted that the highest $RT_{NDT}$ in the core region taken at the 1/4 thickness will be 214°F. The temperature determined by methods of ASME Code Section III for 1989 psig is 134°F above this $RT_{NDT}$ and for 2485 psig (maximum) is 153°F above this $RT_{NDT}$. The minimum inservice leak test temperature requirements for periods up to 16.2 effective full power years are shown on Figure 4.3-1[2].

The heatup limits specified on the heatup curve, Figure 4.3-1, must not be exceeded while the reactor coolant system is being heated to the inservice leak test temperature. For cooldown from the leak test temperature, the limitations of Figure 3.1-2 must not be exceeded. Figures 4.3-1 and 3.1-2 are recalculated periodically, using methods discussed in the Basis for Specification 3.1.B and results of surveillance specimens, as covered in Specification 4.2.

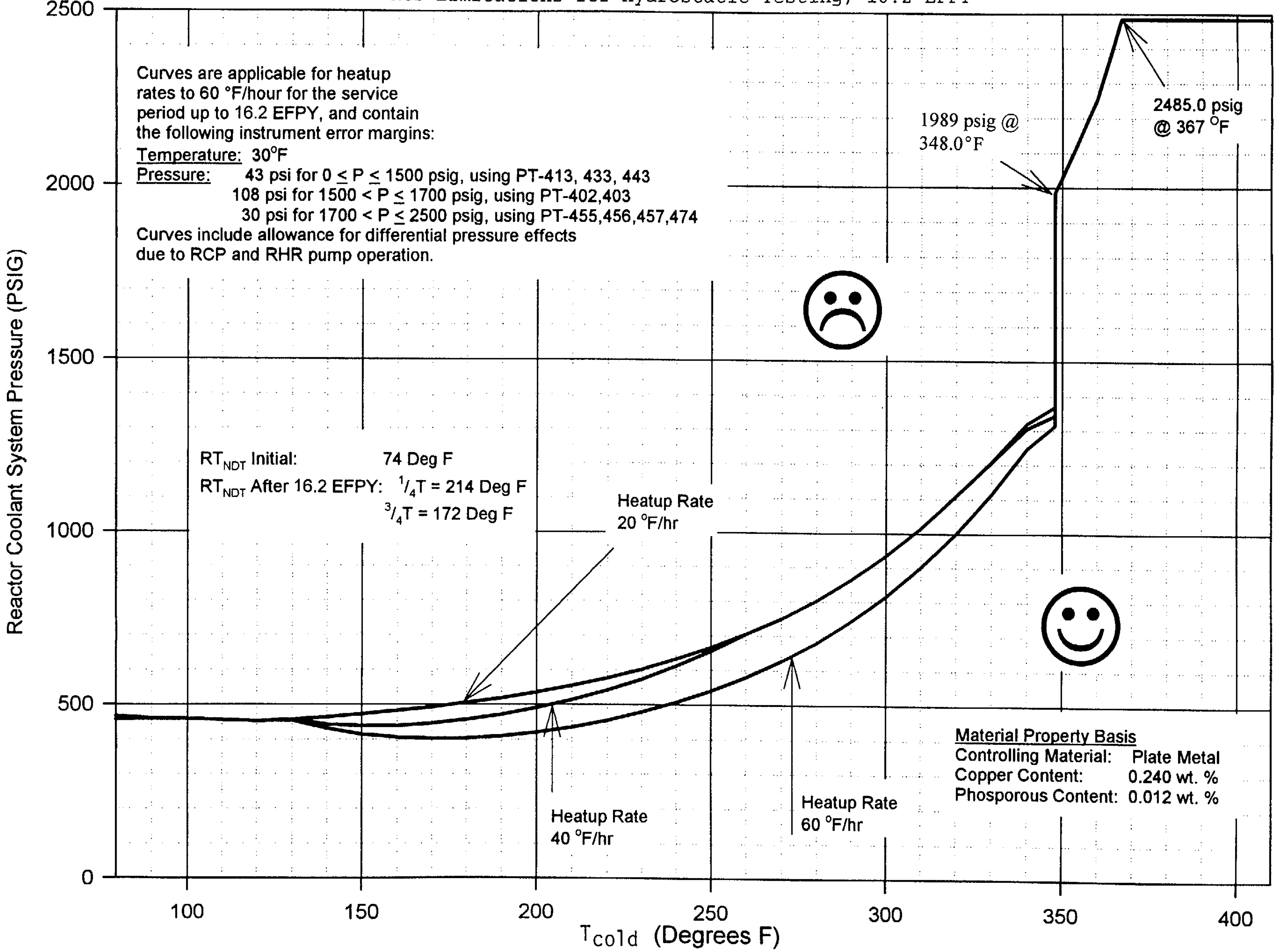
Reference

1. FSAR, Section 4.

2. "Indian Point Unit 3 Final Report on Appendix G Reactor Vessel Pressure-Temperature Limits" ABB-Combustion Engineering, July 24, 1990

FIGURE 4.3-1

RCS Limitations for Hydrostatic Testing, 16.2 EFPY






Amendment ~~28, 109, 121, 179,~~

# SAFETY EVALUATION OF PROPOSED TECHNICAL SPECIFICATION CHANGES ASSOCIATED WITH PRESSURE-TEMPERATURE AND OVERPRESSURE PROTECTION SYSTEM LIMITS FOR UP TO 16.2 EFFECTIVE FULL POWER YEARS

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

# SAFETY EVALUATION RELATED TO PROPOSED TECHNICAL SPECIFICATION CHANGES ASSOCIATED WITH PRESSURE-TEMPERATURE AND OVERPRESSURE PROTECTION SYSTEM LIMITS FOR UP TO 16.2 EFFECTIVE FULL POWER YEARS

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## I. Description of Changes

This amendment request seeks to revise Sections 3.1 and 4.3 of the Technical Specifications. Specifically, the Pressure-Temperature (P/T) and Overpressure Protection System (OPS) limits are being revised from 13.3 Effective Full Power Years (EFPYs) to 16.2 EFPYs. The current limit of 13.3 EFPYs was approved by the NRC in Amendment 179 (Reference 1). The specific changes are as follows.

1. **List of Figures, Page viii**

*Replace:*

"Secondary Side Limitations for RCP Start With Secondary Side Hotter Than Primary, 13.3 EFPY"

"Maximum Allowable Nominal PORV Setpoint for the Low Temperature Overpressure Protection System (OPS), 13.3 EFPY"

"Pressurizer Limitations for OPS Inoperable (Up to one charging pump capable of feeding RCS), 13.3 EFPY"

"Pressurizer Limitations for OPS Inoperable (up to three charging pumps and/or one safety injection pump capable of feeding RCS), 13.3 EFPY"

"RCS Limitations for Hydrostatic Testing, 13.3 EFPY"

*With:*

"Secondary Side Limitations for RCP Start With Secondary Side Hotter Than Primary, 16.2 EFPY"

"Maximum Allowable Nominal PORV Setpoint for the Low Temperature Overpressure Protection System (OPS), 16.2 EFPY"

"Pressurizer Limitations for OPS Inoperable (Up to one charging pump capable of feeding RCS), 16.2 EFPY"

"Pressurizer Limitations for OPS Inoperable (up to three charging pumps and/or one safety injection pump capable of feeding RCS), 16.2 EFPY"

"RCS Limitations for Hydrostatic Testing, 16.2 EFPY"

2. **Figure 3.1.A-1**

*Replace current figure title with:*

"Secondary Side Limitations for RCP Start With Secondary Side Hotter Than Primary, 16.2 EFPY"

3. **Figure 3.1.A-2**

*Replace current figure title with:*

"Maximum Allowable Nominal PORV Setpoint for the Low Temperature Overpressure Protection System (OPS), 16.2 EFPY"

4. **Figure 3.1.A-5**

*Replace current figure title with:*

"Pressurizer Limitations for OPS Inoperable (Up to one charging pump capable of feeding RCS), 16.2 EFPY"

*Replace caption in low right-hand corner with:*

"Figure applicable to 16.2 EFPY"

5. **Figure 3.1.A-6**

*Replace current figure title with:*

"Pressurizer Limitations for OPS Inoperable (up to three charging pumps and/or one safety injection pump capable of feeding RCS), 16.2 EFPY"

*Replace caption in low right-hand corner with:*

"Figure applicable to 16.2 EFPY"

6. **Section 3.1.B.1, page 3.1-17**

*Replace:*

"The reactor coolant temperature and pressure ... for the service period up to 13.3 effective full-power years (EFPYs)."

*With:*

"The reactor coolant temperature and pressure ... for the service period up to 16.2 effective full-power years (EFPYs)."

7. **TS Bases, page 3.1-18**

*Replace:*

"Based on the Westinghouse evaluation, heatup and cooldown curves (Figures 3.1-1 and 3.1-2) were developed for up to 13.3 EFPYs of reactor operation ... The maximum value in $RT_{NDT}$ after 13.3 EFPYs of operation is projected to be ... Heatup and cooldown limit curves are calculated using the most limiting value of $RT_{NDT}$ at the end of 13.3 years of service life. The 13.3 year service life period is chosen such that ..."

*With:*

"Based on the Westinghouse evaluation, heatup and cooldown curves (Figures 3.1-1 and 3.1-2) were developed for up to 16.2 EFPYs of reactor operation ... The maximum value in $RT_{NDT}$ after 16.2 EFPYs of operation is projected to be ... Heatup and cooldown limit curves are calculated using the most limiting value of RTNDT at the end of 16.2 years of service life. The 16.2 year service life period is chosen such that ..."

8. **Figure 3.1-1**

*Replace:*

"Curves are applicable for heatup rates to 60°F/hour for the service period up to 13.3 EFPY ..."

"$RT_{NDT}$ After 13.3 EFPY: ¼ T=214 Deg F"

*With:*

"Curves are applicable for heatup rates to 60°F/hour for the service period up to 16.2 EFPY ..."

"$RT_{NDT}$ After 16.2 EFPY: ¼ T=214 Deg F"

9. **Figure 3.1-2**

*Replace:*

"Curves are applicable for cooldown rates to 100°F/hour for the service period up to 13.3

EFPY ..."

"$RT_{NDT}$ After 13.3 EFPY: ¼ T=214 Deg F"

*With:*

"Curves are applicable for cooldown rates to 100°F/hour for the service period up to 16.2 EFPY ..."

"$RT_{NDT}$ After 16.2 EFPY: ¼ T=214 Deg F"

10. **Section 4.3.A.c, page 4.3-1**

*Replace:*

"The Reactor Coolant System leak test temperature-pressure relationship shall be in accordance with the limits of Figure 4.3-1 for heatup for the first 13.3 EFPYs of operations."

*With:*

"The Reactor Coolant System leak test temperature-pressure relationship shall be in accordance with the limits of Figure 4.3-1 for heatup for the first 16.2 EFPYs of operations."

11. **TS Bases, page 4.3-2**

*Replace:*

"For the first 13.3 effective full power years, it is predicted that the ... The minimum inservice leak test temperature requirements for periods up to 13.3 effective full power years are shown ..."

*With:*

"For the first 16.2 effective full power years, it is predicted that the ... The minimum inservice leak test temperature requirements for periods up to 16.2 effective full power years are shown ..."

12. **Figure 4.3-1**

*Replace current figure title with:*

"RCS Limitations for Hydrostatic Testing, 16.2 EFPY"

*Replace:*

"Curves are applicable for heatup rates to 60°F/hour for the service period up to 13.3 EFPY ... "

"$RT_{NDT}$ After 13.3 EFPY: ¼ T=214 Deg F"

*With:*

"Curves are applicable for heatup rates to 60°F/hour for the service period up to 16.2 EFPY ... "

"$RT_{NDT}$ After 16.2 EFPY: ¼ T=214 Deg F"

## II. Purpose of Proposed Changes

The Technical Specification changes proposed by this amendment revise the P/T and OPS limits to allow continued operation beyond the current TS limit of 13.3 EFPYs and up to 16.2 EFPYs.

## III. Safety Implication of Proposed Changes

This TS amendment request seeks to revise the EFPY limit associated with the P/T and OPS curves. The pressure-temperature limit curves define an acceptable region for normal plant operation. They limit the pressure and temperature changes during RCS heatup and cooldown to within the design assumptions and the stress limits for cyclic operation.

The low temperature overpressurization protection (LTOP) system controls reactor coolant system (RCS) pressure at low temperatures so the integrity of the reactor coolant pressure boundary is not compromised by violating the pressure and temperature limits of 10 CFR 50, Appendix G.

The current Indian Point Unit 3 Technical Specifications associated with Reactor Coolant System (RCS) heatup and cooldown are based on a designated expiration date of 13.3 Effective Full-Power Years (EFPY), which corresponds to a calculated limiting Adjusted Reference Temperature (ART) of 214°F at the 1/4 vessel thickness and 172°F at the 3/4 vessel thickness . These curves were developed by ABB-Combustion Engineering (Reference 4) and were based on the analyzed results of the three reactor vessel surveillance capsules removed to date (References 5, 6, 7).

The capsule analysis reports were originally performed using the ENDF/B-IV cross-section library and were later re-evaluated using the ENDF/B-V library (Reference 8). The resultant fluxes and fluences were then applied to a determination of chemistry factor for the limiting plate (the Indian Point Unit 3 vessel is plate-limited, not weld-limited) and to a general prediction of total fluence versus vessel service life. It is important to note that the Authority has pursued an aggressive program of vessel flux reduction since 1985, which dramatically reduces the neutron flux on the reactor vessel. However, no formal analysis had been performed modeling the effects of the lower-leakage cores on the reactor vessel. Therefore, the Authority utilized very conservative estimates of flux reduction in its most recent submittal; the projected flux was reduced based on the reduction in measured excore current from cycle to cycle and then increased by ten percent to provide margin.

Subsequent to submittal, the Commission expressed concern that the ENDF/B-V library analysis might be non-conservative with respect to calculated flux on the capsule centers and the vessel walls. In addition, they requested that the Authority re-calculate the analytical Chemistry Factor (CF) based on the result of both transverse and longitudinal capsule surveillance measurement. In response to the Commission's request, the Authority re-evaluated the CF and penalized all flux and fluence values from initial plant startup by fifteen percent.

On April 25, 2000, Westinghouse presented the Authority with a report (Reference 9) reanalyzing the projected flux and fluence at the measured capsule centers and for the reactor vessel inner wall. The revised calculation utilized the cross-section library of ENDF/B-VI and the results are summarized in Table 1. The capsule fluences, it should be noted, are nearly identical to those used in the previous submittal. Their statistical combination results in an analytical CF of 167.91° F, which is slightly, lower than the previously calculated CF of 168.12°F. This means that the calculated ART as a function of burnup will be slightly lower than what was used in the past.

Furthermore, the fluence as a function of cycle burnup at the limiting vessel wall location (45 degrees) is shown to be significantly less than what has been submitted to the Commission in the past. This is due to the elimination of the 15 percent penalty, but more significantly, it is due to the effect of the low-leakage cores, which have been specifically modeled in the Westinghouse re-analysis. Prior to this time, as noted above, the effects of the low-leakage cores had either been ignored or represented by a conservative estimate.

The ABB-CE report (Reference 4) identifies that the current Technical Specification heatup and cooldown curves are based on a limiting ART of 214°F at the 1/4 thickness and a total inner radius surface fluence of 6.702E+18 n/cm2. NYPA Calculation IP3-CALC-RV-03197 (Reference 10) shows that this corresponds to a vessel service lifetime of 16.2 EFPY, presuming that average flux for Cycles 11 and beyond is no greater than that of Cycle 10. This is assured by the installation of hafnium flux suppressors into the IP3 core during the most recent refueling outage. ABB-CE independently reviewed the NYPA calculation in draft form and concurred with the results (Reference 11). NYPA has incorporated all ABB-CE input into the final version of the calculation (Reference 10).

Since it is shown that the existing curves bound plant operation from both the standpoint of ART and of total fluence, the Authority requests that the curves be maintained unchanged through a plant service life of 16.2 EFPY. Furthermore, since the Low Temperature Overpressurization Protection System (LTOPS) limit curves are based directly on the heatup and cooldown curves, the Authority requests that these curves be retained with an extended expiration date of 16.2

EFPY. The fifteen percent penalty can be eliminated as a result of the revised analysis, and the standing CF of 168.12°F is conservative and may be retained. In conclusion, the actual curves, and the methodology utilized to generate these curves, will remain the same. This methodology was reviewed and approved by the NRC as part of TS Amendment 179 (Reference 1).

**Table 1**

**SUMMARY OF RESULTS OF ENDF/B-VI REANALYSIS**
**INDIAN POINT UNIT 3**

**Capsule Fluences**

| Capsule | Fluence (previous) ( E19 n/cm2) | Fluence (revised) (E19 n/cm2) |
|---|---|---|
| T | 0.312 | 0.312 |
| Y | 0.724 | 0.731 |
| Z | 1.040 | 1.05 |

**Cycle Flux and Fluence**

| Cycle | EFPY Total | Average Flux (E10 n/cm2-s) | Maximum Fluence (E18 n/cm2) |
|---|---|---|---|
| 1 | 1.37 | 1.95 | 0.843 |
| 2 | 2.23 | 2.23 | 1.45 |
| 3 | 3.28 | 2.13 | 2.15 |
| 4 | 4.41 | 1.63 | 2.73 |
| 5 | 5.55 | 1.34 | 3.21 |
| 6 | 6.73 | 1.21 | 3.67 |
| 7 | 7.81 | 0.960 | 3.99 |
| 8 | 8.94 | 0.973 | 4.34 |
| 9 | 10.49 | 0.918 | 4.79 |
| 10 | 12.28 | 0.867 | 5.28 |

## IV. No Significant Hazards Evaluation

Consistent with the criteria of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based on the following information.

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously analyzed?

Response:

The proposed license amendment does not involve a significant increase in the probability or consequences of a previously analyzed accident. This amendment proposes to extend the EFPY limit from 13.3 to 16.2 for the pressure-temperature and overpressure protection system limit curves. This extension in EFPYs is the result of new fluence values calculated using the ENDF/B-VI database. The methodology used to generate the P/T and OPS limit curves was approved by the NRC in Amendment 179 (Reference 1) and is not being changed by this amendment.

(2) Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response:

The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously analyzed. The P/T and OPS limit curves are being extended through 16.2 EFPYs based on new fluence values calculated using the ENDF/B-VI database. These changes do not affect the way the pressure-temperature or OPS limits provide plant protection and no physical plant alterations are necessary.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response:

The proposed amendment does not involve a significant reduction in a margin of safety. The P/T and OPS limits curves were developed using methodology approved by the NRC for Amendment 179 (Reference 1). This amendment request seeks to revise only the EFPY limits associated with these curves. The new EFPY limits are based upon revised fluence values obtained using the ENDF/B-VI database.

## V. Implementation of Proposed Changes

This amendment request meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) as follows:

(i) The amendment involves no significant hazards consideration.

As described in Section IV of this evaluation, the proposed changes involve no significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed changes do not involve plant equipment or the way the plant operates and therefore do not affect the types or amounts of effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes do not involve plant physical changes, or introduce any new mode of plant operation. Therefore, there is no significant increase in individual or cumulative occupational radiation exposure.

## Section V - Conclusions

The incorporation of these changes: a) will not increase the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; b) will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; c) will not reduce the margin of safety as defined in the bases for any technical specification; and d) involves no significant hazards considerations as defined in 10 CFR 50.92.

**Section VI - References**


1. NRC letter, G. Wunder to J. Knubel, "Issuance of Amendment for Indian Point Nuclear Generating Unit No. 3 (TAC NO. M99928)," (Amendment 179), dated April 10, 1998.

2. NRC letter, G. Wunder to J. Knubel, "Request for Additional Information Regarding Proposed Changes to Pressure and Temperature (P/T) Curves (TAC No. M99928), dated January 14, 1998.

3. NYPA letter to NRC (IPN-98-015), "Response to Request for Additional Information on Proposed Technical Specification Changes Regarding Pressure-Temperature Limits, dated February 6, 1998.

4. PENG-98-014, "Revised ART Determination for Indian Point Unit 3," ABB-Combustion Engineering, January 23, 1998.

5. WCAP-9491, "Analysis of Capsule T from the Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program," Westinghouse Electric Corp., April 1979.

6. WCAP-10300, "Analysis of Capsule Y from the Power Authority of the State of New York, Indian Point Unit 3 Reactor Vessel Radiation Surveillance Program," Westinghouse, March 1983.

7. WCAP-11815, "Analysis of Capsule Z from the New York Power Authority, Indian Point Unit 3 Reactor Vessel Irradiation Surveillance Program," Westinghouse, March 1988.

8. WCAP-14044, "Westinghouse Surveillance Capsule Neutron Fluence Reevaluation," Westinghouse, April 1994.

9. Westinghouse letter INT-00-211 "Evaluation of Reactor Vessel Flux and Fluence Calculations," April 25, 2000.

10. NYPA Calculation IP3-CALC-RV-03197, Revision 0, " Revised Reactor Vessel Neutron Fluence Calculation for April 2000."

11. ABB-Combustion Engineering letter PS-2000-0012, Rev. 0, dated April 24, 2000, "Review of Draft Calculation on Applicability for Indian Point Unit 3 Pressure/Temperature Limits."

Attachment III to IPN-00-033

**MARK-UP OF REVISED TECHNICAL SPECIFICATION PAGES ASSOCIATED WITH AMENDMENT REQUEST REGARDING PRESSURE-TEMPERATURE AND OVERPRESSURE PROTECTION SYSTEM LIMITS FOR UP TO 16.2 EFFECTIVE FULL POWER YEARS**

(FOR INFORMATION ONLY)

NOTE 1: Deletions are shown in ~~strikeout~~, and additions are shown in **bold**.

NOTE 2: Previous amendment revision bars are not shown.

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

LIST OF FIGURES


| No. Title | Figure |
|---|---|
| Core Limits - Four Loop Operation | 2.1-1 |
| Secondary Side Limitations for RCP Start With Secondary Side Hotter Than Primary, ~~13.3~~ **16.2** EFPY | 3.1.A-1 |
| Maximum Allowable Nominal PORV Setpoint for the Low Temperature Overpressure Protection System (OPS), ~~13.3~~ **16.2** EFPY | 3.1.A-2 |
| Deleted | 3.1.A-3 |
| Deleted | 3.1.A-4 |
| Pressurizer Limitations for OPS Inoperable (Up to one charging pump capable of feeding RCS), ~~13.3~~ **16.2** EFPY | 3.1.A-5 |
| Pressurizer Limitations for OPS Inoperable (up to three charging pumps and/or one safety injection pump capable of feeding RCS), ~~13.3~~ **16.2** EFPY | 3.1.A-6 |
| Reactor Coolant System Heatup Limitations | 3.1-1 |
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| Primary Coolant Specific Activity Limit vs. Percent of Rated Thermal Power | 3.1-3 |
| Gross Electrical Output - 1" HG Backpressure | 3.4-1 |
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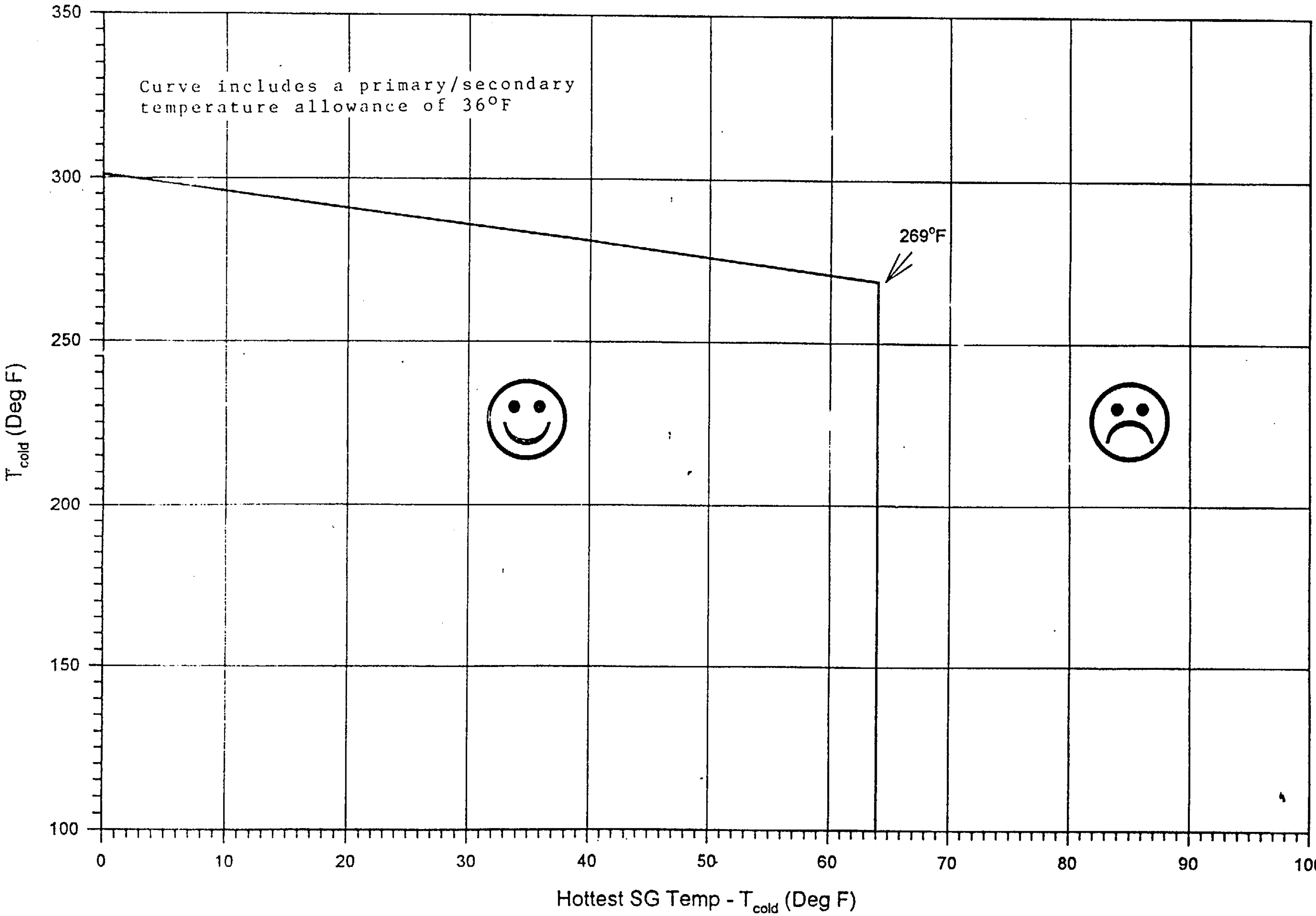
| Spent Fuel Pit Region 1 Fuel Type Definition | 3.8-1 |
| Region 2 Burnup Requirements for Fuel Assembly Storage in the Spent Fuel Pit | 3.8-2 |
| Maximum Density Spent Fuel Pit Racks - Regions and Indexing | 3.8-3 |
| Permissible Rod Misalignment vs. Step Counter Demand Position, > 85% of Rated Thermal Power | 3.10-1 |
| RCS Limitations for Hydrostatic Testing, ~~13.3~~ **16.2** EFPY | 4.3-1 |

FIG' 3.1.A-1

Secondary Side Limitations for RCP Start With
Secondary Side Hotter Than Primary, ~~13.3 EFPY~~ 16.2 EFPY

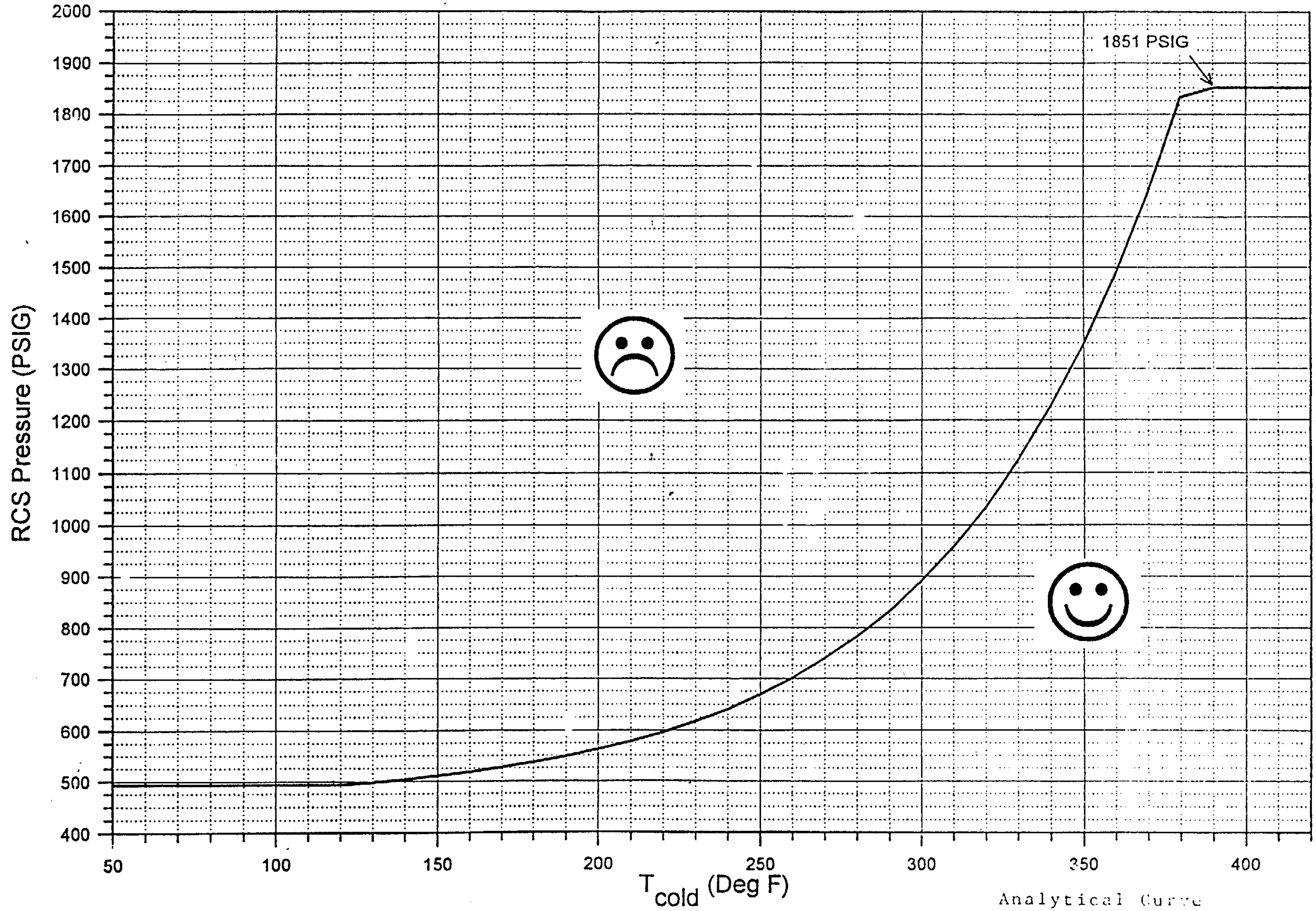





Amendment No. 67, 121, 179

Fi 3.1.A-2

Maximum Allowable Nominal PORV Setpoint for the Low Temperature Overpressure Protection System (OPS), ~~13.3 EFPY~~ 16.2 EFPY




Analytical Curve



Amendment No. ~~67~~, ~~121~~, 179

FIGURE 3.1.A-5

Pressurizer Limitations for OPS Inoperable
(Up to one charging pump capable of feeding RCS), ~~13.3 EFPY~~ 16.2 EFPY

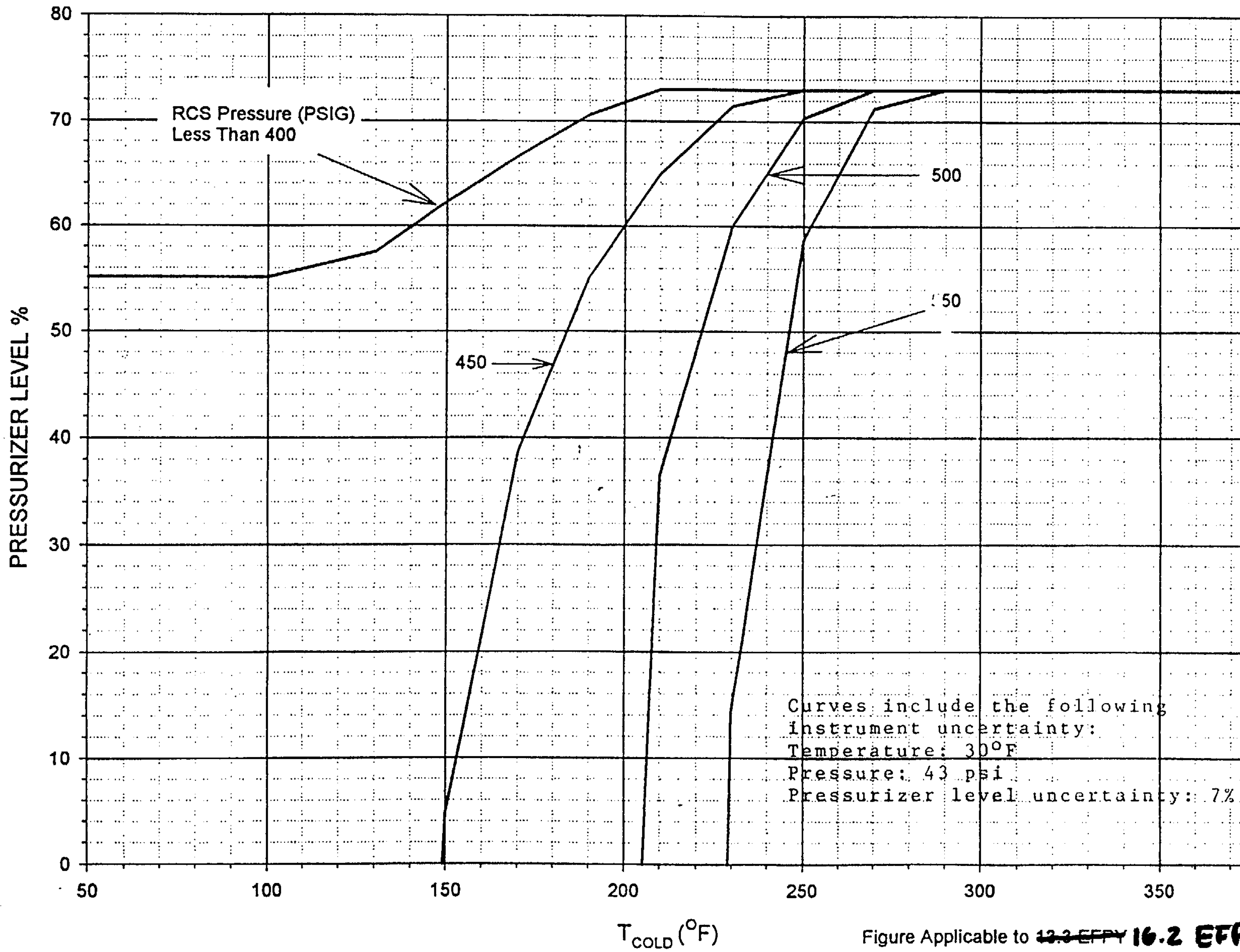



Figure Applicable to ~~13.3 EFPY~~ 16.2 EFPY

Curves represent maximum allowable pressurizer level for the conditions defined.

FIGURE 3.1.A-6

Pressurizer Limitations for OPS Inoperable (up to three charging pumps and/or one safety injection pump capable of feeding RCS), ~~13.3 EFPY~~ 16.2 EFPY

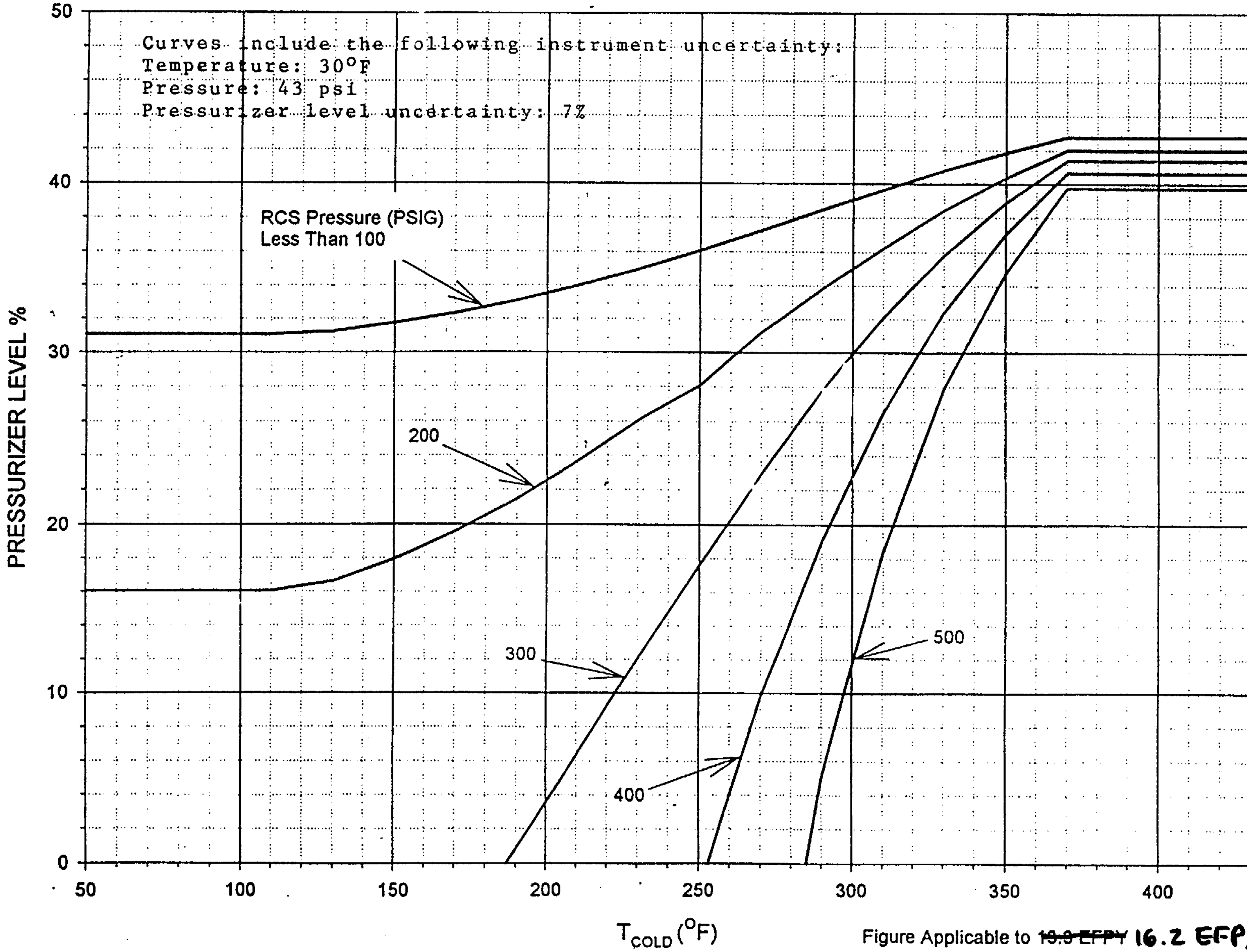



Figure Applicable to ~~13.3 EFPY~~ 16.2 EFPY



Curves represent maximum allowable pressurizer level for the conditions defined.

Amendment No. ~~67~~, ~~101~~, ~~121~~, 179

B. HEATUP AND COOLDOWN

Specifications

1. The reactor coolant temperature and pressure and system heatup and cooldown rates averaged over one hour (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2 for the service period up to ~~13.3~~ **16.2** effective full-power years (EFPYs). The heatup and cooldown rates shall not exceed 60°F/hr and 100°F/hr respectively.

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

2. The limit lines shown in Figure 3.1-1 and Figure 3.1-2 shall be recalculated periodically using methods discussed in the Basis and results of surveillance specimens as covered in Specification 4.2. The order of specimen removal may be modified based on the results of testing of previously removed specimens.

3. The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

4. The pressurizer heatup and cooldown rates averaged over one hour shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

5. Reactor Coolant System integrity tests shall be performed in accordance with Section 4.3.

Basis

Fracture Toughness Properties

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the Summer 1965 Section III of the ASME Boiler and Pressure Vessel Code (6) and ASTM E185 (5) and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1972 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code (1), and the calculation methods described in WCAP-7924 (2).

The first reactor vessel material surveillance capsule was removed during the 1978 refueling outage. This capsule has been tested by Westinghouse Corporation and the results have been evaluated and reported [7]. Similar reports were prepared for the surveillance capsules [10, 8] removed in 1982 and 1987. Based on the Westinghouse evaluation, heatup and cooldown curves (Figures 3.1-1 and 3.1-2) were developed for up to ~~13.3~~ **16.2** EFPYs of reactor operation.

Generic Letter 88-11 requested that licensees use the methodology of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", to predict the effect of neutron radiation on reactor vessel materials as required by paragraph V.A. of 10 CFR part 50, Appendix G. Capsule Z was analyzed [8] and new pressure-temperature curves were developed using this methodology.

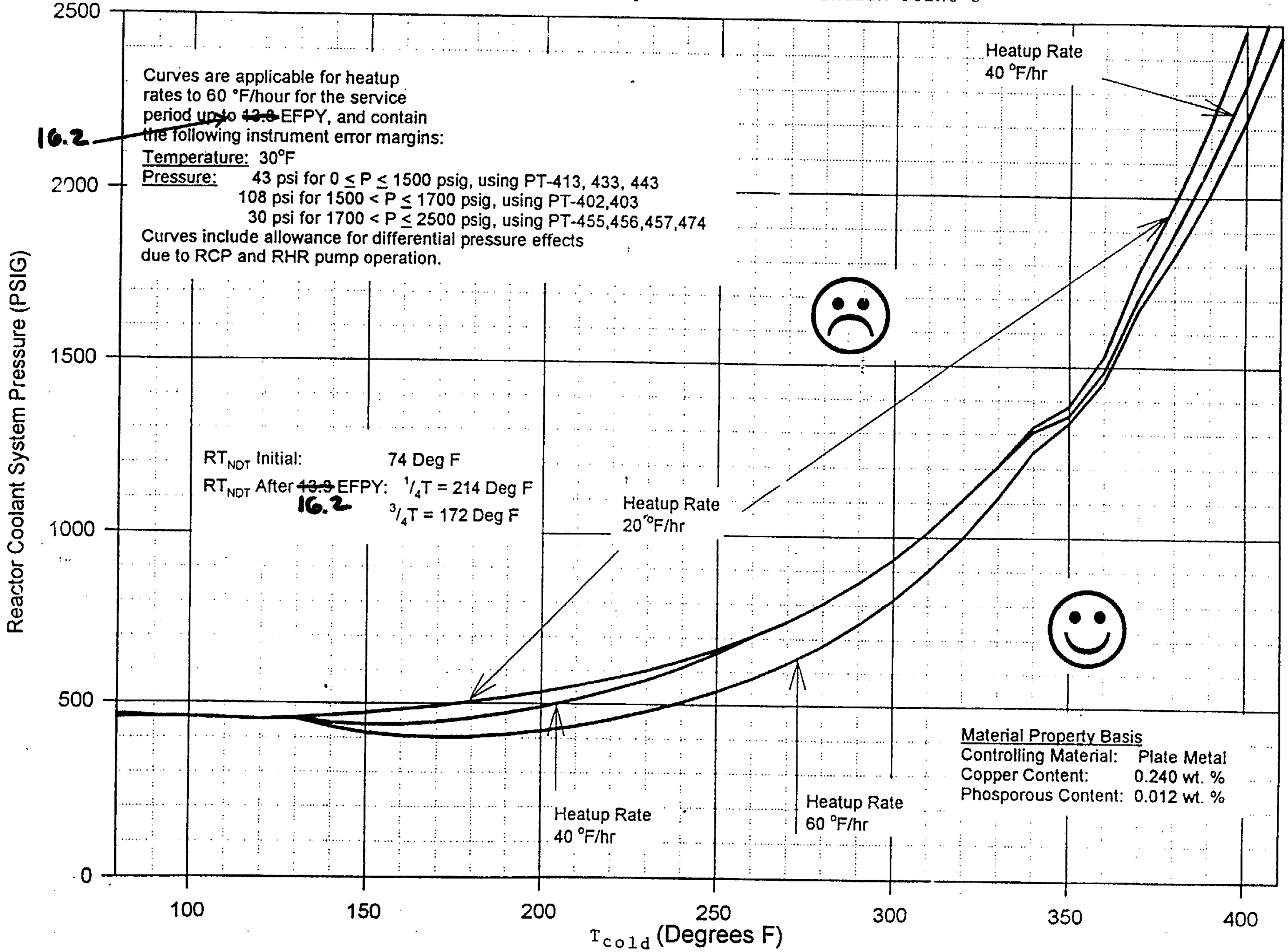
The maximum value in $RT_{NDT}$ after ~~13.3~~ **16.2** EFPYs of operation is projected to be $214^0$F at the 1/4 T and $172^0$F at the 3/4 T vessel wall locations for Plate B2803-3 the controlling plate. Plate B2803-3 was also the controlling plate for the operating period up to 13.3 EFPYs.

Heatup and cooldown limit curves are calculated using the most limiting value of $RT_{NDT}$ at the end of ~~13.3~~ **16.2** years of service life. The ~~13.3~~ **16.2** year service life period is chosen such that the limiting $RT_{NDT}$ at the 1/4 T location in the core region is higher than the $RT_{NDT}$ of the limiting unirradiated material. This service period assures that all components in the Reactor Coolant System will be operated conservatively in accordance with Code recommendations.

The highest $RT_{NDT}$ of the core region material is determined by adding the radiation induced $\Delta RT_{NDT}$ for the applicable time period to the original $RT_{NDT}$ shown in Table Q4.2-1 [3].

FIGURE 3.1-1

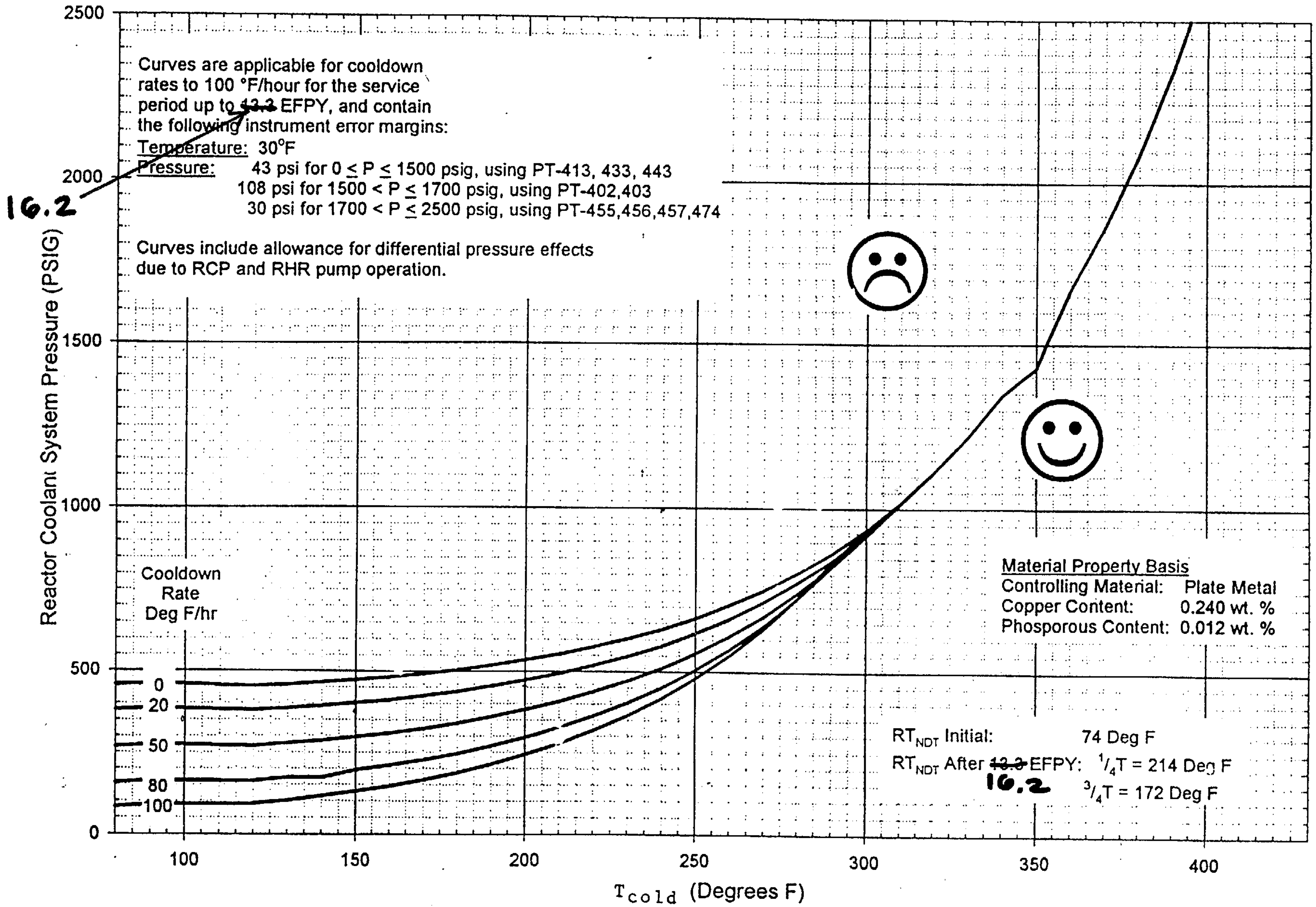
Reactor Coolant System Heatup Limitations - Indian Point 3






Amendment No. ~~28~~, ~~109~~, ~~121~~, 179

FIGURE 3.1-2
Reactor Coolant System Cooldown Limitations - Indian Point 3






Amendment No. ~~28~~, ~~109~~, ~~121~~, 179

4.3 REACTOR COOLANT SYSTEM (RCS) TESTING

A. Reactor Coolant System Integrity Testing

Applicability

Applies to test requirements for Reactor Coolant System integrity.

Objective

To specify tests for Reactor Coolant System integrity after the system is closed following refueling, repair, replacement or modification.

Specification

a) The Reactor Coolant System shall be tested for leakage at normal operating pressure prior to plant startup following each refueling outage, in accordance with the requirements of ASME Section XI.

b) Testing of repairs, replacements or modifications for the Reactor Coolant System shall meet the requirements of ASME Section XI.

c) The Reactor Coolant System leak test temperature-pressure relationship shall be in accordance with the limits of Figure 4.3-1 for heatup for the first ~~13.3~~ **16.2** EFPYs of operations. Figure 4.3-1 will be recalculated periodically. Allowable pressures during cooldown from the leak test temperature shall be in accordance with Figure 3.1-2.

Basis

Leak test of the Reactor Coolant System is required by the ASME Boiler and Pressure Vessel Code, Section XI, to ensure leak tightness of the system during operation. The test frequency and conditions are specified in the Code.

For repairs on components, the thorough non-destructive testing gives a very high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. In all cases, the leak test will assure leak tightness during normal operation.

The inservice leak test temperatures are shown on Figure 4.3-1. The temperatures are calculated in accordance with ASME Code Section III, Appendix G. This Code requires that a safety factor of 1.5 times the stress intensity factor caused by pressure be applied to the calculation.

For the first ~~13.3~~ **16.2** effective full power years, it is predicted that the highest $RT_{NDT}$ in the core region taken at the 1/4 thickness will be 214EF. The temperature determined by methods of ASME Code Section III for 1989 psig is 134°F above this $RT_{NDT}$ and for 2485 psig (maximum) is 153°F above this $RT_{NDT}$. The minimum inservice leak test temperature requirements for periods up to ~~13.3~~ **16.2** effective full power years are shown on Figure 4.3-1[(2)].

The heatup limits specified on the heatup curve, Figure 4.3-1, must not be exceeded while the reactor coolant system is being heated to the inservice leak test temperature. For cooldown from the leak test temperature, the limitations of Figure 3.1-2 must not be exceeded. Figures 4.3-1 and 3.1-2 are recalculated periodically, using methods discussed in the Basis for Specification 3.1.B and results of surveillance specimens, as covered in Specification 4.2.

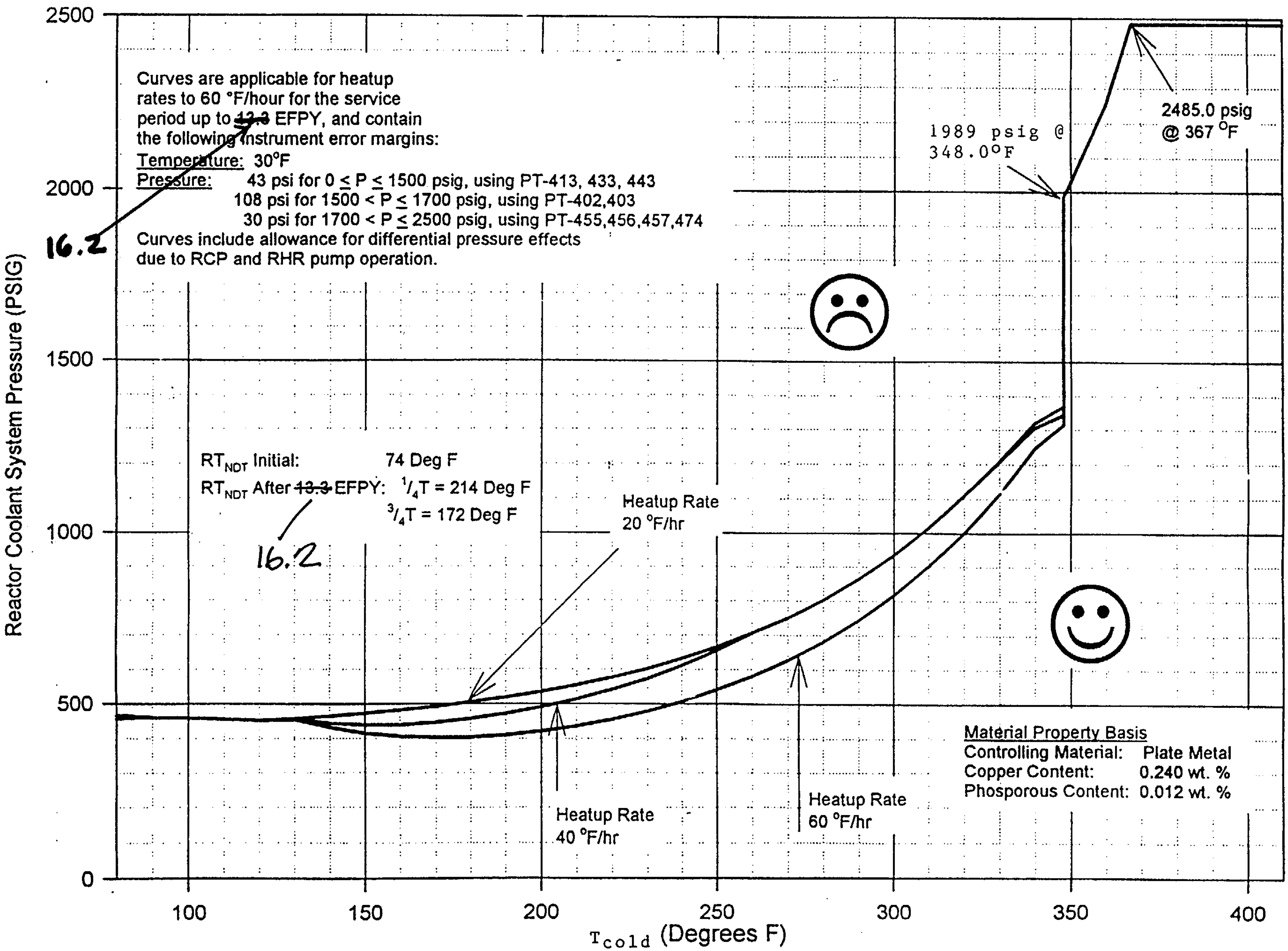
Reference

1. FSAR, Section 4.

2. "Indian Point Unit 3 Final Report on Appendix G Reactor Vessel Pressure-Temperature Limits" ABB-Combustion Engineering, July 24, 1990

FIGURE ..3-1

RCS Limitations for Hydrostatic Testing, ~~13.3 EFPY~~ 16.2 EFPY






Amendment No. ~~28~~, ~~109~~, ~~121~~, 179

Attachment IV to IPN-00-033

**WESTINGHOUSE REPORT**

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

Westinghouse Electric Company, LLC

Box 355
Pittsburgh, Pennsylvania 15230-0355

INT-00-211

Mr. Floyd Gumble
New York Power Authority
123 Main Street
White Plains, New York 10601

April 25, 2000

**New York Power Authority**
Indian Point Unit 3
EVALUATION OF REACTOR VESSEL FLUX AND FLUENCE CALCULATIONS"
(Power Authority Purchase Order No. S-95-76193, Blanket Order Task Number 0001)

Dear Gerry,

The attachment to this letter is the final report describing the Westinghouse re-evaluation of neutron flux and fluence for the Indian Point Unit 3 (IP3) reactor vessel and surveillance capsules. The attached report is considered non-proprietary and may be used in the New York Power Authority submittal to the NRC.

All Power Authority comments have been incorporated into this final version with the exception of the comment regarding Reference 3 as cited in the attached report. Reference 3, which refers to the DOORS 3.1 computer code package, is correct as listed. The DORT two-dimensional discrete ordinates transport code is one of the modules contained in the DOORS system of codes.

This completes our scope of work under Task 0001 of blanket order S-95-76193.

If you have any questions or comments, please contact Stan Anderson (412-374-) or me (412-374-4707).

Sincerely,
Robert R. Laubham

Robert R. Laubham
Supervisory Engineer

| | | |
|---|---|---|
| cc: | George Grochowski | White Plains |
| | Pete Kokolakis | White Plains |
| | Charlene Faison | White Plains |
| | Gerry Canavan | White Plains |
| | Ken Peters | IP3 Site |
| | Dave Lewis | Westinghouse, Danbury, CT |
| | NYPA Document Control, c/o Dave O'Brien | IP3 Site |
| | IP2 Letter Log c/o Kathleen Scandinaro | Westinghouse, Waltz Mill Site |

INT-00-211
April 25, 2000

| | | |
|---|---|---|
| bcc: | Stan Anderson | EC E4-61 |
| | Gary Brassart | EC E4-61 |
| | Rick Easterling | EC E4-08 |
| | Steve Ira | Waltz Mill |
| | Bob Laubham | EC E4-08 |

Internal Reference:

1. SAE-REA-00-606, dated April 25, 2000, S.Anderson.

# ATTACHMENT TO INT-00-211

## APRIL 25, 2000

# INDIAN POINT UNIT 3
# PRESSURE VESSEL NEUTRON EXPOSURE EVALUATION

## 1.0 Introduction

This report describes a discrete ordinates $S_n$ transport analysis performed for the Indian Point Unit 3 reactor to determine the neutron radiation environment within the reactor pressure vessel and surveillance capsules. In this evaluation, fast neutron exposure parameters in terms of fast neutron fluence (E > 1.0 MeV) and iron atom displacements (dpa) were established on a plant and fuel cycle specific basis for the first ten reactor operating cycles. In addition, neutron dosimetry sensor sets from the first three surveillance capsules withdrawn from the Indian Point Unit 3 reactor were re-analyzed using current dosimetry evaluation methodology. The results of this dosimetry re-evaluation were then used to validate the plant specific neutron transport calculations.

The use of fast neutron fluence (E > 1.0 MeV) to correlate measured material property changes to the neutron exposure of the material has traditionally been accepted for development of damage trend curves as well as for the implementation of trend curve data to assess vessel condition. In recent years, however, it has been suggested that an exposure model that accounts for differences in neutron energy spectra between surveillance capsule locations and positions within the vessel wall could lead to an improvement in the uncertainties associated with damage trend curves as well as to a more accurate evaluation of damage gradients through the reactor vessel wall.

Because of this potential shift away from a threshold fluence toward an energy dependent damage function for data correlation, ASTM Standard Practice E853, "Analysis and Interpretation of Light-Water Reactor Surveillance Results," recommends reporting displacements per iron atom (dpa) along with fluence (E > 1.0 MeV) to provide a data base for future reference. The energy dependent dpa function to be used for this evaluation is specified in ASTM Standard Practice E693, "Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements per Atom." The application of the dpa parameter to the assessment of embrittlement gradients through the thickness of the reactor vessel wall has already been promulgated in Revision 2 to Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials." Therefore, in keeping with the philosophy espoused in the current standards governing pressure vessel exposure evaluations, dpa data is also included in this report.

All of the calculations and dosimetry evaluations described in this report were based on the latest available nuclear cross-section data derived from ENDF/B-VI and made use of the latest available calculational tools. Furthermore, the neutron transport and dosimetry evaluation methodologies follow the guidance and meet the requirements of Draft Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."[1] Additionally, the methods used to determine the pressure vessel neutron exposure are consistent with the NRC approved methodology described in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure

Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," January 1996.[2]

## 2.0 Neutron Transport Calculations

In performing the fast neutron exposure evaluations for the Indian Point Unit 3 surveillance capsules and reactor vessel, plant specific forward transport calculations were carried out using the following three-dimensional flux synthesis technique:

$$\phi(r,\theta,z) = \phi(r,\theta) * \frac{\phi(r,z)}{\phi(r)}$$

where $\phi(r,\theta,z)$ is the synthesized three-dimensional neutron flux distribution, $\phi(r,\theta)$ is the transport solution in r,θ geometry, $\phi(r,z)$ is the two-dimensional solution for a cylindrical reactor model using the actual axial core power distribution, and $\phi(r)$ is the one-dimensional solution for a cylindrical reactor model using the same source per unit height as that used in the r,θ two-dimensional calculation.

For the Indian Point Unit 3 analysis, all of the transport calculations were carried out using the DORT two-dimensional discrete ordinates code Version 3.1[3] and the BUGLE-96 cross–section library[4]. The BUGLE-96 library provides a 67 group coupled neutron-gamma ray cross-section data set produced specifically for light water reactor application. In these analyses, anisotropic scattering was treated with a $P_5$ legendre expansion and the angular discretization was modeled with an $S_{16}$ order of angular quadrature.

A plan view of the r,θ model of the Indian Point Unit 3 reactor geometry at the core midplane is shown in Figure 2-1. Since the reactor exhibits approximate octant symmetry only a 0° to 45° sector is depicted. In addition to the core, reactor internals, pressure vessel and primary biological shield, the model also included explicit representations of the surveillance capsules, the pressure vessel cladding, and the insulation located external to the pressure vessel.

From a neutronic standpoint the inclusion of the surveillance capsules and associated support structure in the analytical model is significant. Since the presence of the capsules and structure has a marked impact on the magnitude of the neutron flux as well as on the relative neutron and gamma ray spectra at dosimetry locations within the capsules, a meaningful evaluation of the radiation environment internal to the capsules can be made only when these perturbation effects are properly accounted for in the analysis.

In developing the r,θ analytical model of the reactor geometry shown in Figure 2-1, nominal design dimensions were employed for the various structural components. Likewise, water temperatures and, hence, coolant density in the reactor core and downcomer regions of the reactor were taken to be representative of full power operating conditions. The reactor core itself was treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids, guide tubes, etc. The r,θ geometric mesh description of the reactor model shown in

Figure 2-1 consisted of 170 radial by 67 azimuthal intervals. Mesh sizes were chosen to assure that proper convergence of the inner iterations was achieved on a pointwise basis. The pointwise inner iteration flux convergence criterion utilized in the r,θ calculations was set at a value of 0.001.

A section view of the r,z model of the Indian Point Unit 3 reactor is shown in Figure 2-2. The model extended radially from the centerline of the reactor core out to a location interior to the primary biological shield and over an axial span from an elevation 1 foot below the active fuel to 1 foot above the active fuel. As in the case of the r,θ model, nominal design dimensions and full power coolant densities were employed in the calculations. In this case, the homogenous core region was treated as an equivalent cylinder with a volume equal to that of the active core zone. The stainless steel former plates located between the core baffle and core barrel regions were also explicitly included in the model. The r,z geometric mesh description of the reactor model shown in Figure 2-2 consisted of 153 radial by 90 axial intervals. Mesh sizes were chosen to assure that proper convergence of the inner iterations was achieved on a pointwise basis. The pointwise inner iteration flux convergence criterion utilized in the r,z calculations was also set at a value of 0.001.

The one-dimensional radial model used in the synthesis procedure consisted of the same 153 radial mesh intervals included in the r,z model. Thus, radial synthesis factors could be determined on a meshwise basis throughout the entire geometry.

The core power distributions used in the plant specific transport analysis for the Indian Point Unit 3 reactor were taken from the appropriate fuel cycle design reports for Cycles 1 through 10[5 through 14]. The data extracted from the design reports represented cycle average relative assembly powers, burnups, and axial distributions. Therefore, the calculated results provided data in terms of fuel cycle averaged neutron flux which, when multiplied by the appropriate fuel cycle length, in turn yielded the incremental fast neutron exposure for each fuel cycle. In constructing, the core source distributions, the energy distribution of the source was based on an appropriate fission split for uranium and plutonium isotopes; and from that fission split, composite values of energy release per fission, neutron yield per fission, and fission spectrum were determined.

Calculated fast neutron (E > 1.0 MeV) and iron atom displacement exposures for the Indian Point Unit 3 reactor are presented in Tables 2-1 through 2-6 for plant operation through the conclusion of the tenth fuel cycle. Data are presented at several azimuthal angles at the pressure vessel clad/base metal interface as well as at the geometric center of the in-vessel surveillance capsules. In all cases exposure data are presented in terms of both fluence (E > 1.0 MeV) and dpa.

In Tables 2-1 through 2-4, cycle specific maximum neutron exposures at the pressure vessel clad/base metal interface are given at azimuthal angles of 0°, 15°, 30°, and 45° relative to the core cardinal axes. The data provided in Tables 2-1 through 2-4 were taken at the maximum exposure location based on the three-dimensional synthesized solutions. In Tables 2-5 and 2-6, neutron exposure levels at the geometric center of surveillance capsules located at 4° and 40° relative to the core cardinal axes are provided. In this case, the data are taken at the axial elevation of the midplane of the active fuel region. This location was chosen in order to provide a direct comparison with measurements obtained from in-vessel surveillance capsules that are centered on the core midplane.

Figure 2-1

Indian Point Unit 3 r,θ Reactor Geometry

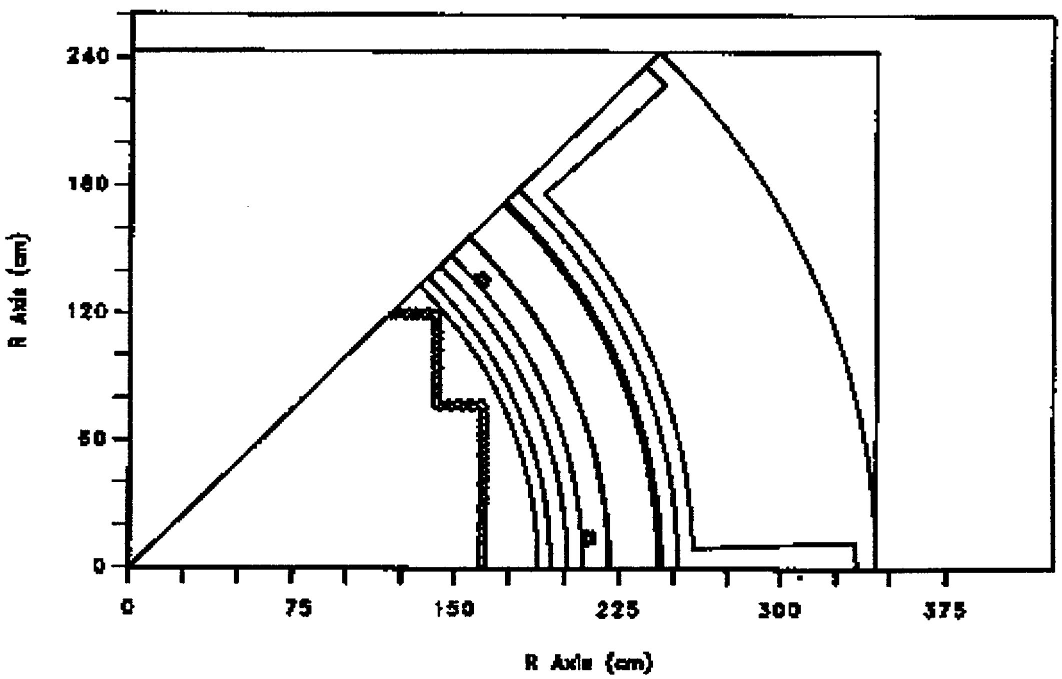


Figure 2-2

Indian Point Unit 3 r,z Reactor Geometry

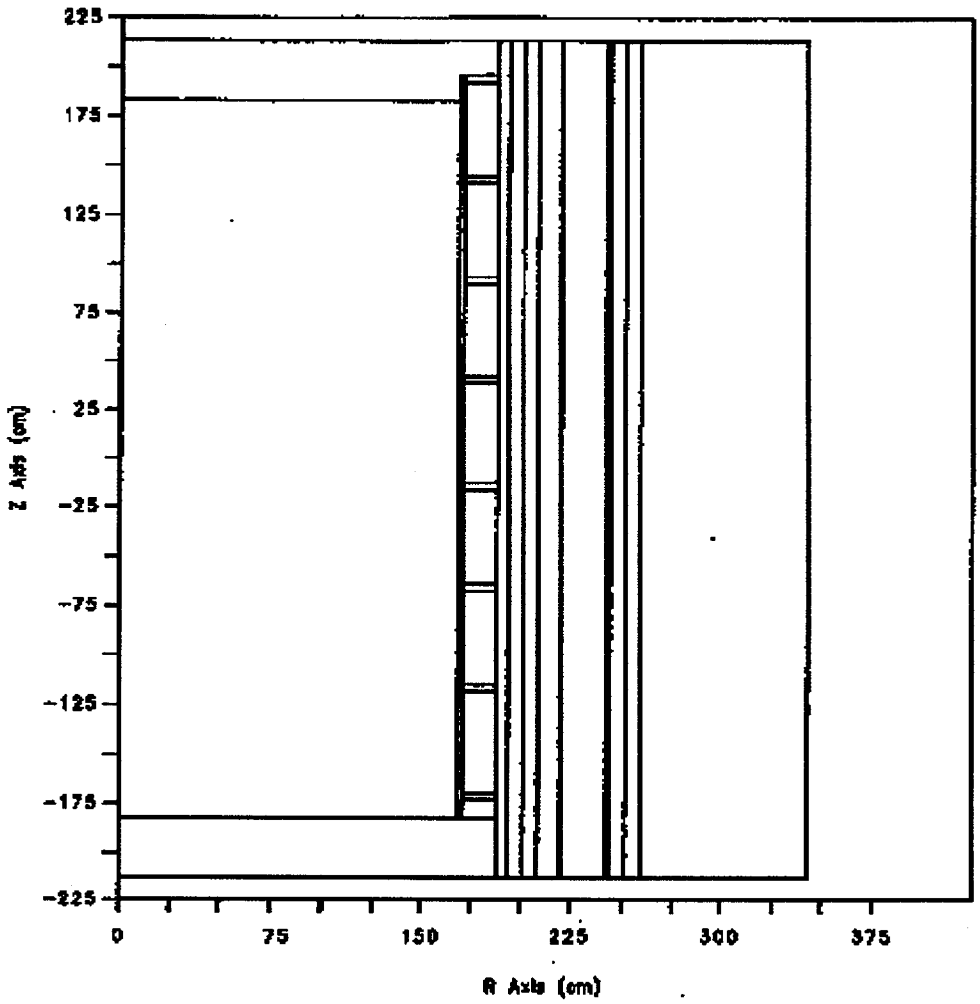


Table 2-1

Indian Point Unit 3
Maximum Fast Neutron Exposure at the Reactor Vessel
Clad/Base Metal Interface – 0 Degree Azimuthal Angle

| Irradiation Interval | Elapsed Irradiation Time [EFPY] | Fluence (E > 1.0 MeV) Average Flux [n/cm2-s] | Fluence (E > 1.0 MeV) Maximum Fluence [n/cm2] | Iron Displacements Average [dpa/s] | Iron Displacements Maximum [dpa] |
|---|---|---|---|---|---|
| **Cycle 1** | 1.37 | 6.46E+09 | 2.80E+17 | 1.05E-11 | 4.54E-04 |
| **Cycle 2** | 2.23 | 6.71E+09 | 4.61E+17 | 1.09E-11 | 7.48E-04 |
| **Cycle 3** | 3.28 | 8.40E+09 | 7.41E+17 | 1.36E-11 | 1.20E-03 |
| **Cycle 4** | 4.41 | 7.29E+09 | 9.99E+17 | 1.18E-11 | 1.62E-03 |
| **Cycle 5** | 5.55 | 6.65E+09 | 1.24E+18 | 1.08E-11 | 2.01E-03 |
| **Cycle 6** | 6.73 | 6.03E+09 | 1.46E+18 | 9.78E-12 | 2.37E-03 |
| **Cycle 7** | 7.81 | 5.29E+09 | 1.64E+18 | 8.60E-12 | 2.67E-03 |
| **Cycle 8** | 8.94 | 6.64E+09 | 1.88E+18 | 1.08E-11 | 3.05E-03 |
| **Cycle 9** | 10.49 | 5.96E+09 | 2.17E+18 | 9.63E-12 | 3.52E-03 |
| **Cycle 10** | 12.28 | 4.22E+09 | 2.41E+18 | 6.84E-12 | 3.91E-03 |

Table 2-2

Indian Point Unit 3
Maximum Fast Neutron Exposure at the Reactor Vessel
Clad/Base Metal Interface – 15 Degree Azimuthal Angle

| Irradiation Interval | Elapsed Irradiation Time [EFPY] | Fluence (E > 1.0 MeV) Average Flux [n/cm2-s] | Maximum Fluence [n/cm2] | Iron Displacements Average [dpa/s] | Maximum [dpa] |
|---|---|---|---|---|---|
| **Cycle 1** | 1.37 | 1.04E+10 | 4.50E+17 | 1.67E-11 | 7.21E-04 |
| **Cycle 2** | 2.23 | 1.07E+10 | 7.39E+17 | 1.71E-11 | 1.18E-03 |
| **Cycle 3** | 3.28 | 1.32E+10 | 1.18E+18 | 2.12E-11 | 1.89E-03 |
| **Cycle 4** | 4.41 | 1.08E+10 | 1.56E+18 | 1.72E-11 | 2.50E-03 |
| **Cycle 5** | 5.55 | 9.52E+09 | 1.90E+18 | 1.52E-11 | 3.05E-03 |
| **Cycle 6** | 6.73 | 8.88E+09 | 2.24E+18 | 1.42E-11 | 3.58E-03 |
| **Cycle 7** | 7.81 | 9.31E+09 | 2.55E+18 | 1.49E-11 | 4.08E-03 |
| **Cycle 8** | 8.94 | 1.03E+10 | 2.92E+18 | 1.64E-11 | 4.67E-03 |
| **Cycle 9** | 10.49 | 7.89E+09 | 3.31E+18 | 1.26E-11 | 5.29E-03 |
| **Cycle 10** | 12.28 | 6.47E+09 | 3.67E+18 | 1.04E-11 | 5.87E-03 |

Table 2-3

Indian Point Unit 3
Maximum Fast Neutron Exposure at the Reactor Vessel
Clad/Base Metal Interface – 30 Degree Azimuthal Angle

| | Elapsed | Fluence (E > 1.0 MeV) | | Iron Displacements | |
|---|---|---|---|---|---|
| | Irradiation | Average | Maximum | | |
| Irradiation | Time | Flux | Fluence | Average | Maximum |
| Interval | [EFPY] | [n/cm2-s] | [n/cm2] | [dpa/s] | [dpa] |
| Cycle 1 | 1.37 | 1.30E+10 | 5.63E+17 | 2.10E-11 | 9.08E-04 |
| Cycle 2 | 2.23 | 1.37E+10 | 9.33E+17 | 2.21E-11 | 1.51E-03 |
| Cycle 3 | 3.28 | 1.53E+10 | 1.44E+18 | 2.46E-11 | 2.33E-03 |
| Cycle 4 | 4.41 | 1.18E+10 | 1.86E+18 | 1.90E-11 | 3.00E-03 |
| Cycle 5 | 5.55 | 9.70E+09 | 2.21E+18 | 1.56E-11 | 3.56E-03 |
| Cycle 6 | 6.73 | 9.43E+09 | 2.56E+18 | 1.52E-11 | 4.13E-03 |
| Cycle 7 | 7.81 | 9.65E+09 | 2.89E+18 | 1.45E-11 | 4.62E-03 |
| Cycle 8 | 8.94 | 1.05E+10 | 3.27E+18 | 1.39E-11 | 5.12E-03 |
| Cycle 9 | 10.49 | 7.55E+09 | 3.63E+18 | 1.22E-11 | 5.71E-03 |
| Cycle 10 | 12.28 | 7.14E+09 | 4.04E+18 | 1.15E-11 | 6.36E-03 |

Table 2-4

Indian Point Unit 3
Maximum Fast Neutron Exposure at the Reactor Vessel
Clad/Base Metal Interface – 45 Degree Azimuthal Angle

| | Elapsed | Fluence (E > 1.0 MeV) | | Iron Displacements | |
|---|---|---|---|---|---|
| | Irradiation | Average | Maximum | | |
| **Irradiation** | **Time** | **Flux** | **Fluence** | **Average** | **Maximum** |
| **Interval** | **[EFPY]** | **[n/cm2-s]** | **[n/cm2]** | **[dpa/s]** | **[dpa]** |
| **Cycle 1** | 1.37 | 1.95E+10 | 8.43E+17 | 3.14E-11 | 1.36E-03 |
| **Cycle 2** | 2.23 | 2.23E+10 | 1.45E+18 | 3.60E-11 | 2.33E-03 |
| **Cycle 3** | 3.28 | 2.13E+10 | 2.15E+18 | 3.44E-11 | 3.48E-03 |
| **Cycle 4** | 4.41 | 1.63E+10 | 2.73E+18 | 2.64E-11 | 4.41E-03 |
| **Cycle 5** | 5.55 | 1.34E+10 | 3.21E+18 | 2.16E-11 | 5.19E-03 |
| **Cycle 6** | 6.73 | 1.21E+10 | 3.67E+18 | 1.96E-11 | 5.93E-03 |
| **Cycle 7** | 7.81 | 9.60E+09 | 3.99E+18 | 1.55E-11 | 6.45E-03 |
| **Cycle 8** | 8.94 | 9.73E+09 | 4.34E+18 | 1.57E-11 | 7.02E-03 |
| **Cycle 9** | 10.49 | 9.18E+09 | 4.79E+18 | 1.48E-11 | 7.74E-03 |
| **Cycle 10** | 12.28 | 8.67E+09 | 5.28E+18 | 1.40E-11 | 8.53E-03 |

Table 2-5

Indian Point Unit 3
Fast Neutron Exposure at the Surveillance Capsule Geometric Center
4 Degree Azimuthal Angle – Core Midplane Elevation

| Irradiation Interval | Elapsed Irradiation Time [EFPY] | Fluence (E > 1.0 MeV) Average Flux [n/cm2-s] | Fluence (E > 1.0 MeV) Maximum Fluence [n/cm2] | Iron Displacements Average [dpa/s] | Iron Displacements Maximum [dpa] |
|---|---|---|---|---|---|
| **Cycle 1** | 1.37 | 2.12E+10 | 9.18E+17 | 3.35E-11 | 1.45E-03 |
| **Cycle 2** | 2.23 | 2.27E+10 | 1.53E+18 | 3.59E-11 | 2.42E-03 |
| **Cycle 3** | 3.28 | 2.80E+10 | 2.46E+18 | 4.43E-11 | 3.90E-03 |
| **Cycle 4** | 4.41 | 2.41E+10 | 3.32E+18 | 3.81E-11 | 5.25E-03 |
| **Cycle 5** | 5.55 | 2.21E+10 | 4.11E+18 | 3.49E-11 | 6.50E-03 |
| **Cycle 6** | 6.73 | 2.00E+10 | 4.86E+18 | 3.15E-11 | 7.68E-03 |
| **Cycle 7** | 7.81 | 1.77E+10 | 5.46E+18 | 2.80E-11 | 8.64E-03 |
| **Cycle 8** | 8.94 | 2.23E+10 | 6.26E+18 | 3.52E-11 | 9.90E-03 |
| **Cycle 9** | 10.49 | 1.95E+10 | 7.21E+18 | 3.08E-11 | 1.14E-02 |
| **Cycle 10** | 12.28 | 1.39E+10 | 8.00E+18 | 2.20E-11 | 1.26E-02 |

Table 2-6

Indian Point Unit 3
Fast Neutron Exposure at the Surveillance Capsule Geometric Center
40 Degree Azimuthal Angle – Core Midplane Elevation

| Irradiation Interval | Elapsed Irradiation Time [EFPY] | Fluence (E > 1.0 MeV) Average Flux [n/cm2-s] | Maximum Fluence [n/cm2] | Iron Displacements Average [dpa/s] | Maximum [dpa] |
|---|---|---|---|---|---|
| **Cycle 1** | 1.37 | 6.67E+10 | 2.89E+18 | 1.11E-10 | 4.79E-03 |
| **Cycle 2** | 2.23 | 7.86E+10 | 5.01E+18 | 1.30E-10 | 8.32E-03 |
| **Cycle 3** | 3.28 | 7.54E+10 | 7.52E+18 | 1.25E-10 | 1.25E-02 |
| **Cycle 4** | 4.41 | 5.67E+10 | 9.53E+18 | 9.41E-11 | 1.58E-02 |
| **Cycle 5** | 5.55 | 4.62E+10 | 1.12E+19 | 7.67E-11 | 1.86E-02 |
| **Cycle 6** | 6.73 | 4.19E+10 | 1.28E+19 | 6.95E-11 | 2.12E-02 |
| **Cycle 7** | 7.81 | 3.35E+10 | 1.39E+19 | 5.55E-11 | 2.31E-02 |
| **Cycle 8** | 8.94 | 3.38E+10 | 1.51E+19 | 5.62E-11 | 2.51E-02 |
| **Cycle 9** | 10.49 | 3.14E+10 | 1.66E+19 | 5.21E-11 | 2.76E-02 |
| **Cycle 10** | 12.28 | 2.99E+10 | 1.83E+19 | 4.96E-11 | 3.04E-02 |

## 3.0 Neutron Dosimetry Evaluations

### 3.1 Sensor Reaction Rate Determinations

In this section, the results of the evaluations of the three neutron sensor sets withdrawn as a part of the Indian Point Unit 3 Reactor Vessel Materials Surveillance Program are presented. The capsule designation, location within the reactor, and time of withdrawal of each of these dosimetry sets were as follows:

| Capsule ID | Azimuthal Location | Withdrawal Time | Irradiation Time [efps] |
|---|---|---|---|
| T | 40° | End of Cycle 1 | 4.33E+07 |
| Y | 40° | End of Cycle 3 | 1.04E+08 |
| Z | 40° | End of Cycle 5 | 1.75E+08 |

The passive neutron sensors included in the Indian Point Unit 3 surveillance capsules are summarized as follows:

| Sensor Material | Reaction of Interest | Capsule T | Capsule Y | Capsule Z |
|---|---|---|---|---|
| Copper | $Cu^{63}(n,\alpha)Co^{60}$ | X | X | X |
| Iron | $Fe^{54}(n,p)Mn^{54}$ | X | X | X |
| Nickel | $Ni^{58}(n,p)Co^{58}$ | X | X | X |
| Uranium-238 | $U^{238}(n,f)Cs^{137}$ | | X | |
| Neptunium-237 | $Np^{237}(n,f)Cs^{137}$ | | X | |
| Cobalt-Aluminum | $Co^{59}(n,\gamma)Co^{60}$ | X | X | X |

The relative locations of the neutron sensors within the capsules are shown in Figure 3-1 The copper, nickel, and cobalt-aluminum monitors, in wire form, were placed in holes drilled in spacers at several axial levels within the capsules. The cadmium shielded uranium and neptunium fission monitors were accommodated within the dosimeter block located near the center of the capsule. The iron sensors were obtained by cutting small samples from individual charpy specimens taken from several locations within the surveillance capsules.

The use of passive monitors such as those listed above does not yield a direct measure of the energy dependent neutron flux at the point of interest. Rather, the activation or fission process is a measure of the integrated effect that the time and energy dependent neutron flux has on the target material over the course of the irradiation period. An accurate assessment of the average neutron flux level incident on the various monitors may be derived from the activation measurements only if the irradiation parameters are well known. In particular, the following variables are of interest:

- The measured specific activity of each monitor,
- The physical characteristics of each monitor,
- The operating history of the reactor,
- The energy response of each monitor, and
- The neutron energy spectrum at the monitor location.

The radiometric counting of each of the Indian Point Unit 3 dosimetry data sets was accomplished by Westinghouse using established ASTM procedures. Following sample preparation and weighing, the activity of each monitor was determined by means of a high resolution gamma spectrometer. For the copper, iron, nickel, and cobalt-aluminum sensors, these analyses were performed by direct counting of each of the individual samples. In the case of the uranium and neptunium fission sensors, the analyses were carried out by direct counting preceded by disolution and chemical separation of cesium from the sensor material.

The irradiation history of the reactor over the irradiation period experienced by Capsules T, Y, and Z was obtained from NUREG-0020, "Licensed Operating Reactors Status Summary Report". The operating data were obtained on a monthly basis from reactor startup to the end of the dosimetry evaluation period. For the sensor sets utilized in the surveillance capsules, the half-lives of the product isotopes are long enough that a monthly histogram describing reactor operation has proven to be an adequate representation for use in radioactive decay corrections for the reactions of interest in the exposure evaluations.

Having the measured specific activities, the operating history of the reactor, and the physical characteristics of the sensors, reaction rates referenced to full power operation were determined from the following equation:

$$R = \frac{A}{N_o F Y \sum_{j=1}^{n} \frac{P_j}{P_{ref}} C_j (1 - e^{-\lambda t_j}) e^{\lambda t_d}}$$

where:

| | | |
|---|---|---|
| A | = | measured specific activity (dps/g) |
| R | = | reaction rate averaged over the irradiation period and referenced to operation at a core power level of $P_{ref}$ (rps/nucleus). |
| $N_0$ | = | number of target element atoms per gram of sensor. |
| F | = | weight fraction of the target isotope in the sensor material. |
| Y | = | number of product atoms produced per reaction. |
| $P_j$ | = | average core power level during irradiation period j (MW). |
| $P_{ref}$ | = | maximum or reference core power level of the reactor (MW). |
| $C_j$ | = | calculated ratio of $\phi(E \geq 1.0$ MeV) during irradiation period j to the time weighted average $\phi(E \geq 1.0$ MeV) over the entire irradiation period. |
| $\lambda$ | = | decay constant of the product isotope ($s^{-1}$). |
| $t_j$ | = | length of irradiation period j (s). |
| $t_d$ | = | decay time following irradiation period j (s). |

and the summation is carried out over the total number of monthly intervals comprising the irradiation period.

In the above equation, the ratio $P_j/P_{ref}$ accounts for month by month variation of power level within a given fuel cycle. The ratio $C_j$ is calculated for each fuel cycle using the methodology described in Section 2.0 of this report and accounts for the change in sensor reaction rates caused by variations in flux level due to changes in core power spatial distributions from fuel cycle to fuel cycle. For a single cycle irradiation $C_j$ = 1.0. However, for multiple cycle irradiations, particularly those employing low leakage fuel management, the additional $C_j$ correction must be utilized. This additional correction can be quite significant for sensor sets that have been irradiated for many fuel cycles in a reactor that has transitioned from non-low leakage to low leakage fuel management.

Prior to using the measured reaction rates in the least squares adjustment procedure discussed later in this section, additional corrections were made to $U^{238}$ measurements to account for the presence of $U^{235}$ impurities in the sensors as well as to adjust for the build-in of plutonium isotopes over the course of the irradiation. These corrections were location and fluence dependent and were derived from the plant specific discrete ordinates analysis described in Section 2.0. Corrections were also made to the $U^{238}$ and $Np^{237}$ sensor reaction rates to account for gamma ray induced fission reactions that occurred over the course of the irradiation. These photo-fission corrections were, likewise, location dependent and were based on the transport calculations described in Section 2.0. The correction factors applied to the Indian Point Unit 3 fission sensor reaction rates are summarized as follows:

| Correction | Capsule T | Capsule Y | Capsule Z |
|---|---|---|---|
| $U^{235}$ Impurity/Pu Build-in | Not Applicable | 0.856 | Not Applicable |
| $U^{238}(\gamma,f)$ | Not Applicable | 0.958 | Not Applicable |
| Net $U^{238}$ Correction | Not Applicable | 0.820 | Not Applicable |
| $Np^{237}(\gamma,f)$ | Not Applicable | 0.985 | Not Applicable |

These factors were applied in a multiplicative fashion to the decay corrected fission sensor reaction rates.

Results of the sensor reaction rate determinations for Capsules T, Y, and Z are given in Tables 3.1-1 through 3.1-4. In Tables 3.1-1 through 3.1-3, the measured specific activities, gradient corrected specific activities, and decay corrected reaction rates are listed for Capsules T, Y, and Z, respectively. A summary of the reaction rates for each capsule is provided in Table 3.1-4. The data listed in Table 3.1-4 are indexed to the geometric center of the respective capsules and included all corrections for $U^{235}$ impurities, Pu build-in, and photo-fission effects.

Figure 3-1

Surveillance Capsule Geometry

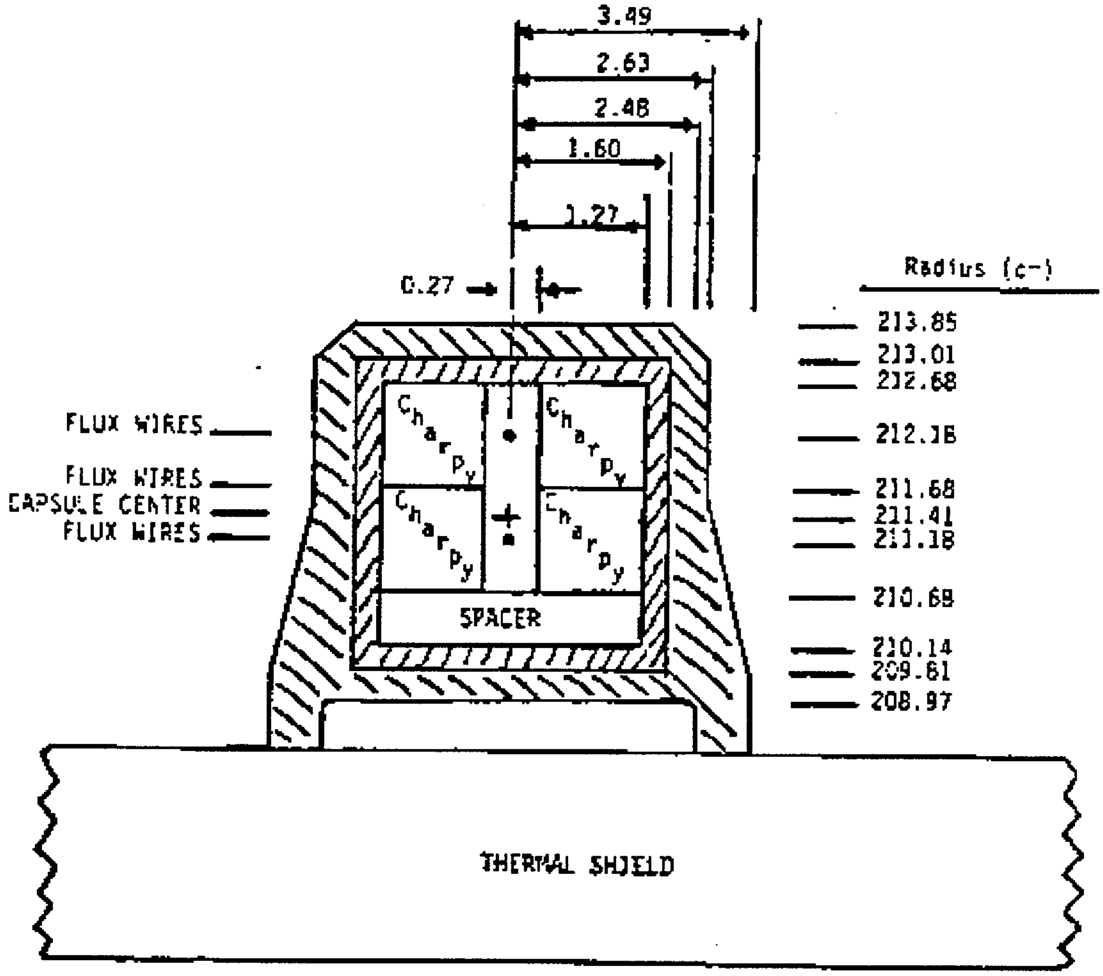


Table 3.1-1

Measured Sensor Specific Activities and Reaction Rates
Capsule T

| Sample ID | Foil ID | Foil Type | Measured Activity [dps/g] | Saturated Activity [dps/g] | Radially Adjusted Saturated Activity [dps/g] | Radially Adjusted Reaction Rate [rps/atom] | Average Reaction Rate [rps/atom] | Corrected Average Reaction Rate [rps/atom] |
|---|---|---|---|---|---|---|---|---|
| **78-3097** | Top | Cu | 4.870E+04 | 3.243E+05 | 3.097E+05 | 4.724E-17 | | |
| **78-3103** | Bottom | Cu | 4.590E+04 | 3.056E+05 | 2.919E+05 | 4.453E-17 | 4.589E-17 | 4.589E-17 |
| **78-3104** | W-40 | Fe | 1.370E+06 | 3.489E+06 | 3.325E+06 | 5.270E-15 | | |
| **78-3107** | W-37 | Fe | 1.250E+06 | 3.183E+06 | 3.034E+06 | 4.809E-15 | | |
| **78-3108** | B-9 | Fe | 1.340E+06 | 3.412E+06 | 3.252E+06 | 5.155E-15 | | |
| **78-3105** | A-32 | Fe | 1.170E+06 | 2.979E+06 | 3.444E+06 | 5.460E-15 | | |
| **78-3106** | TA-58 | Fe | 1.090E+06 | 2.776E+06 | 3.209E+06 | 5.086E-15 | | |
| **78-3109** | AT-54 | Fe | 1.180E+06 | 3.005E+06 | 3.474E+06 | 5.506E-15 | 5.214E-15 | 5.214E-15 |
| **78-3100** | Middle | Ni | 8.110E+06 | 4.985E+07 | 4.750E+07 | 6.801E-15 | 6.801E-15 | 6.801E-15 |
| **78-3095** | Top | Co -Bare | 7.630E+06 | 5.081E+07 | 4.949E+07 | 3.229E-12 | | |
| **78-3098** | Middle | Co -Bare | 6.840E+06 | 4.555E+07 | 4.436E+07 | 2.894E-12 | | |
| **78-3101** | Bottom | Co -Bare | 7.680E+06 | 5.114E+07 | 4.981E+07 | 3.250E-12 | 3.124E-12 | 3.124E-12 |
| **78-3096** | Top | Co - Cd | 3.080E+06 | 2.051E+07 | 2.363E+07 | 1.541E-12 | | |
| **78-3099** | Middle | Co - Cd | 3.030E+06 | 2.018E+07 | 2.324E+07 | 1.516E-12 | | |
| **78-3102** | Bottom | Co - Cd | 3.020E+06 | 2.011E+07 | 2.317E+07 | 1.511E-12 | 1.523E-12 | 1.523E-12 |

Table 3.1-2

Measured Sensor Specific Activities and Reaction Rates
Capsule Y

| Sample ID | Foil ID | Foil Type | Measured Activity [dps/g] | Saturated Activity [dps/g] | Radially Adjusted Saturated Activity [dps/g] | Radially Adjusted Reaction Rate [rps/atom] | Average Reaction Rate [rps/atom] | Corrected Average Reaction Rate [rps/atom] |
|---|---|---|---|---|---|---|---|---|
| **82-2066** | Top | Cu | 8.580E+04 | 3.121E+05 | 2.980E+05 | 4.546E-17 | | |
| **82-2072** | Bottom | Cu | 8.930E+04 | 3.248E+05 | 3.102E+05 | 4.732E-17 | 4.639E-17 | 4.639E-17 |
| | W-16 | Fe | 1.050E+06 | 3.144E+06 | 2.997E+06 | 4.750E-15 | | |
| | W-12 | Fe | 1.040E+06 | 3.114E+06 | 2.968E+06 | 4.705E-15 | | |
| | W-9 | Fe | 1.060E+06 | 3.174E+06 | 3.025E+06 | 4.796E-15 | | |
| | AT-37 | Fe | 8.980E+05 | 2.689E+06 | 3.109E+06 | 4.928E-15 | | |
| | AT-33 | Fe | 8.700E+05 | 2.605E+06 | 3.012E+06 | 4.774E-15 | | |
| | AT-30 | Fe | 8.140E+05 | 2.438E+06 | 2.818E+06 | 4.467E-15 | 4.737E-15 | 4.737E-15 |
| **82-2069** | Middle | Ni | 4.050E+06 | 4.631E+07 | 4.413E+07 | 6.318E-15 | 6.318E-15 | 6.318E-15 |
| **82-2063** | Middle | U-238 | 3.320E+05 | 4.763E+06 | 4.763E+06 | 3.128E-14 | 3.128E-14 | 2.564E-14 |
| **82-2062** | Middle | Np-237 | 2.340E+06 | 3.357E+07 | 3.357E+07 | 2.142E-13 | 2.142E-13 | 2.109E-13 |
| **82-2064** | Top | Co -Bare | 1.520E+07 | 5.528E+07 | 5.384E+07 | 3.513E-12 | | |
| **82-2067** | Middle | Co -Bare | 1.420E+07 | 5.165E+07 | 5.030E+07 | 3.282E-12 | | |
| **82-2070** | Bottom | Co -Bare | 1.520E+07 | 5.528E+07 | 5.384E+07 | 3.513E-12 | 3.436E-12 | 3.436E-12 |
| **82-2065** | Top | Co - Cd | 5.560E+06 | 2.022E+07 | 2.330E+07 | 1.520E-12 | | |
| **82-2068** | Middle | Co - Cd | 5.360E+06 | 1.949E+07 | 2.246E+07 | 1.465E-12 | | |
| **82-2071** | Bottom | Co - Cd | 5.790E+06 | 2.106E+07 | 2.426E+07 | 1.583E-12 | 1.523E-12 | 1.523E-12 |

Table 3.1-3

Measured Sensor Specific Activities and Reaction Rates
Capsule Z

| Sample ID | Foil ID | Foil Type | Measured Activity [dps/g] | Saturated Activity [dps/g] | Radially Adjusted Saturated Activity [dps/g] | Radially Adjusted Reaction Rate [rps/atom] | Average Reaction Rate [rps/atom] | Corrected Average Reaction Rate [rps/atom] |
|---|---|---|---|---|---|---|---|---|
| **87-2679** | Top | Cu | 1.010E+05 | 2.968E+05 | 2.834E+05 | 4.324E-17 | | |
| **87-2686** | Bottom | Cu | 9.920E+04 | 2.915E+05 | 2.784E+05 | 4.246E-17 | 4.285E-17 | 4.285E-17 |
| **87-2799** | W-64 | Fe | 9.920E+05 | 2.976E+06 | 2.837E+06 | 4.496E-15 | | |
| **87-2798** | W-61 | Fe | 9.930E+05 | 2.979E+06 | 2.839E+06 | 4.501E-15 | | |
| **87-2797** | TI | Fe | 9.660E+05 | 2.898E+06 | 2.762E+06 | 4.379E-15 | | |
| **87-2796** | AT-82 | Fe | 7.940E+05 | 2.382E+06 | 2.754E+06 | 4.366E-15 | | |
| **87-2795** | AT-78 | Fe | 8.250E+05 | 2.475E+06 | 2.861E+06 | 4.536E-15 | | |
| **87-2794** | A-56 | Fe | 7.650E+05 | 2.295E+06 | 2.653E+06 | 4.206E-15 | 4.414E-15 | 4.414E-15 |
| **87-2682** | Middle | Ni | 4.030E+06 | 4.098E+07 | 3.906E+07 | 5.591E-15 | 5.591E-15 | 5.591E-15 |
| **87-2677** | Top | Co -Bare | 1.470E+07 | 4.319E+07 | 4.207E+07 | 2.745E-12 | | |
| **87-2680** | Middle | Co -Bare | 1.360E+07 | 3.996E+07 | 3.892E+07 | 2.539E-12 | | |
| **87-2684** | Bottom | Co -Bare | 1.490E+07 | 4.378E+07 | 4.264E+07 | 2.782E-12 | 2.689E-12 | 2.689E-12 |
| **87-2578** | Top | Co - Cd | 5.900E+06 | 1.734E+07 | 1.997E+07 | 1.303E-12 | | |
| **87-2681** | Middle | Co - Cd | 5.820E+06 | 1.710E+07 | 1.970E+07 | 1.285E-12 | | |
| **87-2685** | Bottom | Co - Cd | 5.920E+06 | 1.739E+07 | 2.004E+07 | 1.307E-12 | 1.298E-12 | 1.298E-12 |

Table 3.1-4

Summary of Sensor Reaction Rates from Capsules T, Y, and Z

| | Reaction Rate [rps/atom] | | |
|---|---|---|---|
| **Reaction** | Capsule T | Capsule Y | Capsule Z |
| $Cu^{63}(n,\alpha)Co^{60}$ | 4.589E-17 | 4.639E-17 | 4.285E-17 |
| $Fe^{54}(n,p)Mn^{54}$ | 5.214E-15 | 4.737E-15 | 4.414E-15 |
| $Ni^{58}(n,p)Co^{58}$ | 6.801E-15 | 6.318E-15 | 5.591E-15 |
| $U^{238}(n,f)Cs^{137}$ Cd | | 2.564E-14 | |
| $Np^{237}(n,f)Cs^{137}$ Cd | | 2.109E-13 | |
| $Co^{59}(n,\gamma)Co^{60}$ | 3.124E-12 | 3.436E-12 | 2.689E-12 |
| $Co^{59}(n,\gamma)Co^{60}$ Cd | 1.523E-12 | 1.523E-12 | 1.298E-12 |

Note: Cd Indicates that the sensor was cadmium covered.

### 3.2 Least Squares Evaluation of Sensor Sets

Least squares adjustment methods provide the capability of combining the measurement data with the neutron transport calculation resulting in a Best Estimate neutron energy spectrum with associated uncertainties. Best Estimates for key exposure parameters such as $\phi(E > 1.0\ \text{MeV})$ or dpa/s along with their uncertainties are then easily obtained from the adjusted spectrum. In general, the least squares methods, as applied to surveillance capsule dosimetry evaluations, act to reconcile the measured sensor reaction rate data, dosimetry reaction cross-sections, and the calculated neutron energy spectrum within their respective uncertainties. For example,

$$R_i \pm \delta_{R_i} = \sum_g (\sigma_{ig} \pm \delta_{\sigma_{ig}})(\phi_g \pm \delta_{\phi_g})$$

relates a set of measured reaction rates, $R_i$, to a single neutron spectrum, $\phi_g$, through the multigroup dosimeter reaction cross-section, $\sigma_{ig}$, each with an uncertainty $\delta$. The primary objective of the least squares evaluation is to produce unbiased estimates of the neutron exposure parameters at the location of the measurement.

For the least squares evaluation of the Indian Point Unit 3 surveillance capsule dosimetry, The FERRET code[15] was employed to combine the results of the plant specific neutron transport calculations and sensor set reaction rate measurements to determine best estimate values of exposure parameters ($\phi(E > 1.0\ \text{MeV})$ and dpa) along with associated uncertainties for the three in-vessel capsules withdrawn to date.

The application of the least squares methodology requires the following input:

1 - The calculated neutron energy spectrum and associated uncertainties at the measurement location.

2 - The measured reaction rates and associated uncertainty for each sensor contained in the multiple foil set.

3 - The energy dependent dosimetry reaction cross-sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

For the Indian Point Unit 3 application, the calculated neutron spectrum was obtained from the results of plant-specific neutron transport calculations described in Section 2.0 of this report. The sensor reaction rates were derived from the measured specific activities using the procedures described in Section 3.1. The dosimetry reaction cross-sections and uncertainties were obtained from the SNLRML dosimetry cross-section library[16]. The SNLRML library is an evaluated dosimetry reaction cross-section compilation recommended for use in LWR evaluations by ASTM Standard E1018, "Application of ASTM Evaluated Cross-Section Data File, Matrix E 706 (IIB)".

The uncertainties associated with the measured reaction rates, dosimetry cross-sections, and calculated neutron spectrum were input to the least squares procedure in the form of variances and covariances. The assignment of the input uncertainties

followed the guidance provided in ASTM Standard E 944, "Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance.
The following provides a summary of the uncertainties associated with the least squares evaluation of the Indian Point Unit 3 surveillance capsule sensor sets:

Reaction Rate Uncertainties

The overall uncertainty associated with the measured reaction rates includes components due to the basic measurement process, the irradiation history corrections, and the corrections for competing reactions. A high level of accuracy in the reaction rate determinations is assured by utilizing laboratory procedures that conform to the ASTM National Consensus Standards for reaction rate determinations for each sensor type.

After combining all of these uncertainty components, the sensor reaction rates derived from the counting and data evaluation procedures were assigned the following net uncertainties for input to the least squares evaluation:

| Reaction | Uncertainty |
|---|---|
| $Cu^{63}(n,\alpha)Co^{60}$ | 5% |
| $Fe^{54}(n,p)Mn^{54}$ | 5% |
| $Ni^{58}(n,p)Co^{58}$ | 5% |
| $U^{238}(n,f)Cs^{137}$ $Np^{237}(n,f)Cs^{137}$ | 10% |
| $Co^{59}(n,\gamma)Co^{60}$ | 10% |
| | 5% |

These uncertainties are given at the 1$\sigma$ level.

Dosimetry Cross-Section Uncertainties

The reaction rate cross-sections used in the least squares evaluations were taken from the SNLRML library. This data library provides reaction cross-sections and associated uncertainties, including covariances, for 66 dosimetry sensors in common use. Both cross-sections and uncertainties are provided in a fine multigroup structure for use in least squares adjustment applications. These cross-sections were compiled from the most recent cross-section evaluations and they have been tested with respect to their accuracy and consistency for least squares evaluations. Further, the library has been empirically tested for use in fission spectra determination as well as in the fluence and energy characterization of 14 MeV neutron sources. Detailed discussions of the contents of the SNLRML library along with the evaluation process for each of the sensors is provided in Reference 16.

For sensors included in the Indian Point Unit 3 surveillance, the following uncertainties in the fission spectrum averaged cross-sections are provided in the SNLRML documentation package.

| Reaction | Uncertainty |
|---|---|
| $Cu^{63}(n,\alpha)Co^{60}$ | 4.08-4.16% |
| $Fe^{54}(n,p)Mn^{54}$ | 3.05-3.11% |
| $Ni^{58}(n,p)Co^{58}$ | 4.49-4.56% |

| | |
|---|---|
| $U^{238}(n,f)Cs^{137}$ $Np^{237}(n,f)Cs^{137}$<br>$Co^{59}(n,\gamma)Co^{60}$ | 0.54-0.64%<br>10.32-10.97%<br>0.79-3.59% |

These tabulated ranges provide an indication of the dosimetry cross-section uncertainties associated with the sensor sets used in LWR irradiations.

Calculated Neutron Spectrum

The neutron spectrum input to the least squares adjustment procedure was obtained directly from the results of plant specific transport calculations for each surveillance capsule location. The spectrum at each location was input in an absolute sense (rather than as simply a relative spectral shape). Therefore, within the constraints of the assigned uncertainties, the calculated data were treated equally with the measurements.

While the uncertainties associated with the reaction rates were obtained from the measurement procedures and counting benchmarks and the dosimetry cross-section uncertainties were supplied directly with the SNLRML library, the uncertainty matrix for the calculated spectrum was constructed from the following relationship:

$$M_{g'g} = R_n^2 + R_g * R_{g'} * P_{g'g}$$

where $R_n$ specifies an overall fractional normalization uncertainty and the fractional uncertainties $R_{g'}$ and $R_g$ specify additional random groupwise uncertainties that are correlated with a correlation matrix given by:

$$P_{g'g} = [1 - \vartheta] * \delta_{g'g} + \vartheta * e^{-H}$$

where:

$$H = \frac{(g - g')^2}{2\gamma^2}$$

The first term in the correlation matrix equation specifies purely random uncertainties, while the second term describes the short range correlations over a group range $\gamma$ ($\theta$ specifies the strength of the latter term). The value of $\delta$ is 1.0 when g = g' and 0.0 otherwise.

The set of parameters defining the input covariance matrix for the Indian Point Unit 3 calculated spectra was as follows:

| | |
|---|---|
| Flux Normalization Uncertainty ($R_n$) | 15% |
| Flux Group Uncertainties ($R_g$, $R_{g'}$) | |
| (E > 0.0055 MeV) | 15% |
| (0.68 eV < E < 0.0055 MeV) | 29% |
| (E < 0.68 eV) | 52% |

| | |
|---|---|
| Short Range Correlation ($\theta$) | |
| (E > 0.0055 MeV) | 0.9 |
| (0.68 eV < E < 0.0055 MeV) | 0.5 |
| (E < 0.68 eV) | 0.5 |
| Flux Group Correlation Range ($\gamma$) | |
| (E > 0.0055 MeV) | 6 |
| (0.68 eV < E < 0.0055 MeV) | 3 |
| (E < 0.68 eV) | 2 |

Results of the least squares evaluation of the three sensor sets withdrawn from the Indian Point Unit 3 reactor are provided in Tables 3.2-1 and 3.2-2. In Table 3.2-1, measured, calculated, and best estimate sensor reaction rates are given for Capsules T, Y, and Z. The improvement in the fit of the adjusted spectra to the measurements is evident for all three capsule data sets. Prior to the application of the adjustment procedure M/C ratios for individual foil reactions ranged from 0.91 to 1.18, while after the adjustment M/BE ratios ranged from 0.95 to 1.06. Thus, demonstrating a significant improvement in the data fits.

In Table 3.2-2, the calculated and best estimate exposure rates and integrated exposures of Capsules T, Y, and Z are given. Data are provided in terms of both fluence (E > 1.0 MeV) and iron atom displacements. The best estimate fluence values and associated uncertainties are recommended for use in data correlation studies for measured materials properties from test specimens removed from Capsules T, Y, and Z.

Table 3.2-1

Comparison of Measured, Calculated, and Best Estimate Reaction Rates at the Surveillance Capsule Center

**Surveillance Capsule T**

| | Reaction Rate [rps/atom] | | | | |
|---|---|---|---|---|---|
| **Reaction** | **Measured** | **Calculated** | **Best Estimate** | **M/C** | **M/BE** |
| $Cu^{63}(n,\alpha)Co^{60}$ | 4.59E-17 | 4.20E-17 | 4.59E-17 | 1.09 | 1.00 |
| $Fe^{54}(n,p)Mn^{54}$ | 5.21E-15 | 4.65E-15 | 5.11E-15 | 1.12 | 1.02 |
| $Ni^{58}(n,p)Co^{58}$ | 6.80E-15 | 6.41E-15 | 6.93E-15 | 1.06 | 0.98 |
| $Co^{59}(n,\gamma)Co^{60}$ | 3.12E-12 | 2.67E-12 | 3.11E-12 | 1.17 | 1.00 |
| $Co^{59}(n,\gamma)Co^{60}$ Cd | 1.52E-12 | 1.39E-12 | 1.52E-12 | 1.09 | 1.00 |

**Surveillance Capsule Y**

| | Reaction Rate [rps/atom] | | | | |
|---|---|---|---|---|---|
| **Reaction** | **Measured** | **Calculated** | **Best Estimate** | **M/C** | **M/BE** |
| $Cu^{63}(n,\alpha)Co^{60}$ | 4.64E-17 | 4.53E-17 | 4.51E-17 | 1.02 | 1.03 |
| $Fe^{54}(n,p)Mn^{54}$ | 4.74E-15 | 5.05E-15 | 4.82E-15 | 0.94 | 0.98 |
| $Ni^{58}(n,p)Co^{58}$ | 6.32E-15 | 6.95E-15 | 6.58E-15 | 0.91 | 0.96 |
| $U^{238}(n,f)Cs^{137}$ Cd | 2.56E-14 | 2.50E-14 | 2.42E-14 | 1.02 | 1.06 |
| $Np^{237}(n,f)Cs^{137}$ Cd | 2.11E-13 | 1.96E-13 | 2.02E-13 | 1.07 | 1.04 |
| $Co^{59}(n,\gamma)Co^{60}$ | 3.44E-12 | 2.91E-12 | 3.41E-12 | 1.18 | 1.01 |
| $Co^{59}(n,\gamma)Co^{60}$ Cd | 1.52E-12 | 1.52E-12 | 1.53E-12 | 1.00 | 1.00 |

**Surveillance Capsule Z**

| | Reaction Rate [rps/atom] | | | | |
|---|---|---|---|---|---|
| **Reaction** | **Measured** | **Calculated** | **Best Estimate** | **M/C** | **M/BE** |
| $Cu^{63}(n,\alpha)Co^{60}$ | 4.28E-17 | 4.04E-17 | 4.17E-17 | 1.06 | 1.03 |
| $Fe^{54}(n,p)Mn^{54}$ | 4.41E-15 | 4.47E-15 | 4.35E-15 | 0.99 | 1.01 |
| $Ni^{58}(n,p)Co^{58}$ | 5.59E-15 | 6.16E-15 | 5.84E-15 | 0.91 | 0.95 |
| $Co^{59}(n,\gamma)Co^{60}$ | 2.69E-12 | 2.56E-12 | 2.68E-12 | 1.05 | 1.00 |
| $Co^{59}(n,\gamma)Co^{60}$ Cd | 1.30E-12 | 1.34E-12 | 1.30E-12 | 0.97 | 1.00 |

Table 3.2-2

Comparison of Calculated and Best Estimate
Exposure Parameters at the Surveillance Capsule Center

**Time Averaged Exposure Rates**

| | $\phi$(E > 1.0 MeV) [n/cm$^2$-s] | | | |
|---|---|---|---|---|
| | Calculated | Best Estimate | Uncertainty | BE/C |
| Capsule T | 6.66E+10 | 7.20E+10 | 7% | 1.08 |
| Capsule Y | 7.25E+10 | 7.05E+10 | 6% | 0.97 |
| Capsule Z | 6.39E+10 | 5.97E+10 | 7% | 0.94 |

| | Iron Atom Displacements [dpa/s] | | | |
|---|---|---|---|---|
| | Calculated | Best Estimate | Uncertainty | BE/C |
| Capsule T | 1.11E-10 | 1.19E-10 | 9% | 1.08 |
| Capsule Y | 1.20E-10 | 1.19E-10 | 7% | 0.99 |
| Capsule Z | 1.06E-10 | 1.00E-10 | 9% | 0.94 |

**Integrated Capsule Exposure**

| | $\Phi$ (E > 1.0 MeV) [n/cm$^2$] | | | |
|---|---|---|---|---|
| | Calculated | Best Estimate | Uncertainty | BE/C |
| Capsule T | 2.88E+18 | 3.12E+18 | 7% | 1.08 |
| Capsule Y | 7.52E+18 | 7.31E+18 | 6% | 0.97 |
| Capsule Z | 1.12E+19 | 1.05E+19 | 7% | 0.94 |

| | Iron Atom Displacements [dpa] | | | |
|---|---|---|---|---|
| | Calculated | Best Estimate | Uncertainty | BE/C |
| Capsule T | 4.79E-03 | 5.15E-03 | 9% | 1.08 |
| Capsule Y | 1.25E-02 | 1.24E-02 | 7% | 0.99 |
| Capsule Z | 1.86E-02 | 1.75E-02 | 9% | 0.94 |

## 4.0 Comparisons of Measurement and Calculation

In this section, comparisons of the measurement results from the three surveillance capsules withdrawn to date with the corresponding analytical predictions at the measurement locations are provided. These comparisons are given on two levels. In the first instance, calculations of individual sensor reaction rates are compared directly with the corresponding values obtained from the measured specific activities. In the second case, calculations of fast neutron exposure rates in terms of $\phi(E > 1.0 \text{ MeV})$ and dpa/s are compared with the best estimate results obtained from the least squares evaluation of the three capsule dosimetry results. These two levels of comparison yield consistent and similar results with all measurement to calculation comparisons falling well within the 20% limits specified as the acceptance criteria in DG-1053.

In the case of the direct comparison of measured and calculated sensor reaction rates, the M/C comparisons for fast neutron reactions range from 0.91–1.12 for the 11 samples included in the data set. In the comparisons of best estimate and calculated fast neutron exposure parameters, the corresponding BE/C comparisons range from 0.94 –1.08 for the three surveillance capsules withdrawn to date.

Based on these comparisons, it is concluded that the data comparisons validate the use of the calculated fast neutron exposures provided in Section 2.0 of this report for use in the assessment of the condition of the materials comprising the beltline region of the Indian Point Unit 3 reactor pressure vessel.

## 5.0 References

1 - Draft Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," September 1999 draft version.

2 - J. D. Andrachek, et. al., "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," WCAP-14040-NP-A, January 1996.

3 - RSICC Computer Code Collection CCC-650, "DOORS 3.1, One, Two- and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," August 1996.

4 - RSIC Data Library Collection DLC-185, "BUGLE-96, Coupled 47 Neutron, 20 Gamma-Ray Group Cross-Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," March 1996.

5 - P. J. Sipush, et. al., "Core Physics Characteristics of the Indian Point Nuclear Plant Unit III, Cycle 1," WCAP-8360, October 1974.

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11 - T. A. Downs, et. al., "Nuclear Parameters and Operations Package for Indian Point Unit 3, Cycle 7," WCAP-12169, May 1989.

12 - J. E. Pritchett, et. al., "Nuclear Parameters and Operations Package for Indian Point Unit 3, Cycle 8," WCAP-12750, February 1991.

13 - K. R. Robinson, et. al., "Nuclear Parameters and Operations Package for Indian Point Unit 3, Cycle 9," WCAP-13270 Rev. 1, January 1995.

14 - R. J. Fetterman, et. al., "Nuclear Parameters and Operations Package for Indian Point Unit 3, Cycle 10," WCAP-14889, June 1997.

15 - A. Schmittroth, "FERRET Data Analysis Code," HEDL-TME-79-40, Hanford Engineering Development Laboratory, Richland, WA, September 1979.

16 - RSIC Data Library Collection DLC-178, "SNLRML Recommended Dosimetry Cross-Section Compendium," July 1994.

Attachment V to IPN-00-033

**NYPA Calculation ID IP3-CALC-RV-03197,**
**"Revised Reactor Vessel Neutron Fluence Calculation," Revision 0**

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

Nuclear Engineering
CALCULATION CONTROL SHEET

Nuclear Plant: IP3 ☑ JAF ☐

CALC. NO. IP3-CALC-RV-03197 REVISION 0 PAGE 1 OF 2

CALCULATION IS: PRELIMINARY ___ FINAL x

| | NAME | SIGNATURE | DATE |
|---|---|---|---|
| PREPARER: | F. Gumble | | 4/25/00 |
| CHECKER: | G. Canavan | | 4/25/00 |
| (DESIGN) VERIFIED/NA | G. Canavan | | 4/25/00 |
| APPROVED: | G. Grochowski | | 4/25/00 |

ORIGINATOR: NYPA ☑ OR OTHER ☐

SYSTEM NO./NAME Reactor Vessel

TITLE: Revised Reactor Vessel Neutron Fluence Calculation --- April 2000

QA CATEGORY: I DISCIPLINE: Rx Eng STRUCTURE: Rx Vessel

MODIFICATION NO./TASK NO. N/A DBD REF. NO. N/A

PROBLEM/OBJECTIVE/METHOD
Based on Westinghouse's April 2000 re-evaluation of reactor vessel fluence, determine a revised expiration date for the heatup/cooldown/OPS curves in the IP3 Technical Specifications

DESIGN BASIS/ASSUMPTION
Calculations are based on flux and fluence evaluations using the ENDF/B-VI cross-section libraries.

SUMMARY/CONCLUSIONS
Curves may be extended to 16.2 Effective Full-Power Years

THIS CALC SUPERSEDES OR VOIDS CALC.# None

DISTRIBUTION: C= CONTROLLED I= INFO

| NAME | DEPT | LOC | C | I |
|---|---|---|---|---|
| F. Gumble | PRE | WPO | x | |
| K. Jacobs | PRAS | WPO | x | |
| W. Spataro | PEP | WPO | x | |
| G. Canavan | PRE | WPO | x | |
| M. Cochrane | IDSE | IP3 | x | |

CALCULATION SHEET

| | | |
|---|---|---|
| New York Power Authority | | |
| CALCULATION NO. IP3-CALC-RV-03197 | | REVISION 0 |
| Project N/A | | Page 1 of 10 |
| Title Revised Rx Vessel Fluence Calc | | Date April 17, 2000 |
| Preliminary ____ | Prepared by F. Gumble FMG | Date 4/17/00 |
| Final X | Checked by G. Canavan GC | Date 4/20/00 |

## 1.0 BACKGROUND AND PURPOSE

Technical Specification (Tech Spec) Amendment 179 revised the heatup, cooldown and OPS curves (HU/CD/OPS) for a reactor vessel maximum service lifetime of 13.3 Effective Full-Power Years (EFPY). The calculations associated with that amendment included a 15% penalty on neutron flux and fluence. This was at the NRC's request, since the reactor vessel surveillance capsules had been analyzed using the ENDF/B-IV and ENDF/B-V cross-section libraries, not the ENDF/B-VI library, which the Commission preferred.

Prior to Amendment 179, no credit for low-leakage loading patterns had ever been used in the development of the HU/CD/OPS curves. Therefore, in order to compensate somewhat for the 15% penalty on neutron flux, a partial allowance for reduced neutron leakage core designs was incorporated into the Tech Spec curves. Here is how it was done: for Amendment 179, the neutron flux that had been calculated by Westinghouse for the end of Cycle 5 was reduced by an amount comparable to the measured change in full-power excore detector current for each subsequent cycle. In order to allow for uncertainty, this reduced flux value was increased, by ten percent of the difference, in order to ensure a conservative result. In summary, the neutron flux values that led to Tech Spec Amendment 179 were determined by:

> Starting with the measured flux at the end of Cycle 5 (from the Capsule Z Report);
> Increasing all flux and fluence values by 15%, per the NRC's recommendation;
> Reducing the resultant flux to allow for low-leakage core design;
> Increasing the resultant flux slightly to ensure conservatism.

In April 2000, Westinghouse completed a formal reanalysis of the IP3 surveillance capsules and reactor vessel flux/fluence, using the ENDF/B-VI libraries. This reanalysis also formally incorporated the effects of low-leakage core designs through Cycle 10. This allows the Authority to eliminate the 15% penalty previously applied to flux and fluence, and to use a more accurate representation of low-leakage core designs.

Since the operators have become familiar with the curves as they now appear in the Tech Specs, Reactor Engineering recommends that the existing curves stay the same but with extended expiration dates. This will have a minimal effect on plant operations.

The revised analysis allows the curves expiration dates to be extended from 13.3 EFPY to 16.2 EFPY. Details on this conclusion are provided in the sections that follow.

## 2.0 ASSUMPTIONS

2.1 Neutron flux and fluence calculations are determined using the ENDF/B-VI cross-section library as described in Reference 6.1

2.2 The limiting location for neutron flux impingement remains the 45° position. This is assured by a review of the other positions on the Westinghouse report (Reference 6.1).

2.3 This calculation presumes that neutron flux and fluence remain below those of Cycle 10. For Cycle 11, this is shown explicitly in Section 3. For Cycles 12 and beyond, a follow item is opened to evaluate neutron flux for every cycle and revise the curves if the flux exceeds that of Cycle 10.

It should be noted that this should not be a problem, since the Cycle 11 flux was severely reduced due to the installation of hafnium flux suppressors, which are expected to be re-used (or replaced) every cycle. This will keep the neutron flux at the limiting location well below Westinghouse's assumptions for Cycle 10 (See Section 5.0).

## 3.0 CALCULATIONS

The current HU/CD curves in the IP3 Tech Specs are based on those provided by the reactor vessel vendor, ABB-Combustion Engineering (ABB-CE), in References 6.2 and 6.3. The OPS curve is itself generated from the isothermal CD curve. Its derivation can be found in Reference 6.4.

There are two major parameters that affect the generation of the HU/CD/OPS curves: these are the Adjusted Reference Temperature (ART) at the 1/4 and 3/4 reactor vessel thickness, and the total fluence on the vessel, which is itself a function of the average neutron flux.

Reference 6.2 defines the parameters upon which the limiting curves are based. Reference

6.3 provides the actual curves. These curves are based upon the following assumptions:

> Limiting neutron fluence at the inner wetted surface of 6.702E+18 n/cm$^2$ ;
> ART of 214°F at the 1/4 vessel thickness and 171.5°F at the 3/4 vessel thickness.

The ART is calculated in accordance with the procedures established in Reg Guide 1.99 Rev 2 (Reference 6.6). The ART is a function of the following factors

Initial $RT_{NDT}$
Chemistry Factor
Neutron Fluence
Margin

$RT_{NDT}$ is the the reference temperature for the unirradiated material as defined in Paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code (Reference 6.7). The limiting initial $RT_{NDT}$ for the reactor vessel is based on the manufacturing records for plate B-2803-3. Initial $RT_{NDT}$ does not change for this calculation.

Margin is an uncertainty factor defined in Reg Guide 1.99 Rev 2. ABB-CE defined the margin factor in Reference 6.3, based on the results of the surveillance capsule analyses. This margin factor was used in the derivation of the curves in Amendment 179. Since the capsule analysis has not changed, and since the neutron fluence at the capsule centers (as seen below) is essentially the same as previously calculated, the margin factor does not change for this calculation.

Chemistry Factor (CF) and neutron fluence (f) are both affected by Westinghouse's re-evaluation. They are used in combination to determine how the ART changes with reactor vessel service life, in accordance with the equation:

$$\Delta RT_{NDT} = (CF)\ f^{(0.28 - 0.10 \log f)}$$

In the above equation, $\Delta RT_{NDT}$ represents the change of ART in degrees F. The fluence factor f is in units of 1E+19 n/cm$^2$.

In the following sections, we will evaluate CF and f for the revised analysis.

### 3.1 Chemistry Factor

The Chemistry Factor CF can be determined in two ways, either by a table look-up or a statistical combination of measured surveillance capsule data. For IP3, the statistical method results in a higher (i.e., more limiting) CF. Therefore, the statistical method is always used.

The Westinghouse reanalysis in Reference 6.1 provides us with revised capsule neutron fluences. As noted in the table that follows, the reanalyzed fluences using ENDF/B-VI are quite similar to those analyzed using ENDF/B-V.

| Capsule | Fluence (Current Tech Spec) | Fluence (Revised April 2000) |
|---|---|---|
| T | 3.12E+18 n/cm$^2$ | 3.117E+18 n/cm$^2$ |
| Y | 7.24E+18 n/cm$^2$ | 7.305E+18 n/cm$^2$ |
| Z | 1.04E+19 n/cm$^2$ | 1.046E+19 n/cm$^2$ |

For Tech Spec Amendment 179, Reference 6.2 showed that the statistically combined results of the capsule fluences with the measured shift in reference temperature at the 30 ft-lb level for each surveillance capsule resulted in a CF of 168.12°F. The measured shifts do not change for this evaluation. When they are combined with the revised fluences, the results are as follows:

| A | B | C | D | E | F | G |
|---|---|---|---|---|---|---|
| **Capsule** | **Specimen Orientation** | **Capsule Fluence f ($10^{19}$ n/cm$^2$)** | $f^{(0.28 - 0.10 \log f)}$ | **Measured Shift* (°F)** | **"D" x "E" (°F)** | **(D)$^2$** |
| T | Transverse | 0.3117 | 0.680 | 118 | 80.24 | .4624 |
| T | Longitudinal | 0.3117 | 0.680 | 137 | 93.16 | .4624 |
| Y | Transverse | 0.7305 | 0.9119 | 150 | 136.79 | .8316 |
| Z | Transverse | 1.046 | 1.0126 | 155 | 156.95 | 1.0254 |
| Z | Longitudinal | 1.046 | 1.0126 | 170 | 172.14 | 1.0254 |

*Note: Column E is the measured shift in reference temperature at the 30 ft-lb level for each reactor vessel specimen (Reference 6.3)

The sum of column F is 639.28°F.
The sum of column G is 3.8072

Reg Guide 1.99 Rev 2 defines CF as $(\Sigma F)/(\Sigma G) = 639.28°F / 3.8072$
$= 167.91$ °F

This is slightly LESS than the current value of 168.12°F. Therefore, the use of the current CF is slightly more restrictive and therefore conservative. For ease of calculation, we shall RETAIN the bounding value of 168.12°F.

Note: It is expected that analysis of the next capsule (Capsule S, scheduled for retrieval in the 2001 refueling outage) will be consistent with the analysis shown here. However, since it is possible that the next capsule may result in a higher CF, a follow item is opened to re-evaluate ART subsequent to removal of Capsule S (See Section 5.0).

For purposes of comparison, note that CF determined by the table-lookup method is equal to 160.4°F, which is less than the statistical value of CF.

### 3.2 Flux and Fluence

The Westinghouse re-evaluation under ENDF/B-VI is the first cycle-specific calculation of vessel neutron exposure to be performed since 1989. At that time, Westinghouse issued WCAP-11057, Rev 1 (Reference 6.8), which estimated inner vessel wall exposure based on the analytical results of the Capsule T, Y and Z analyses. The WCAP made assumptions about the effects of low-leakage cores which have since been proven to be overly conservative (i.e., too high).

The revised analysis re-evaluates average neutron flux and fluence at various locations around the vessel wall (45°, 15°, 30° and 0° azimuthal angles). As expected, the limiting case is the 45° angle, where the reactor comes closest to the vessel wall.

The new analysis (Reference 6.1) incorporates the results of each individual core design. The individual fluxes and fluences are as follows:

| Cycle | Total EFPY at end of cycle | Average Flux ($n/cm^2$-s) | Maximum Accumu-latedFluence ($n/cm^2$) |
|---|---|---|---|
| 1 | 1.37 | 1.95E+10 | 8.43E+17 |
| 2 | 2.23 | 2.23E+10 | 1.45E+18 |
| 3 | 3.28 | 2.13E+10 | 2.15E+18 |
| 4 | 4.41 | 1.63E+10 | 2.73E+18 |
| 5 | 5.55 | 1.34E+10 | 3.21E+18 |
| 6 | 6.73 | 1.21E+10 | 3.67E+18 |
| 7 | 7.81 | 9.60E+09 | 3.99E+18 |
| 8 | 8.94 | 9.73E+09 | 4.34E+18 |
| 9 | 10.49 | 9.18E+09 | 4.79E+18 |
| 10 | 12.28 | 8.67E+09 | 5.28E+18 |

The fluxes and fluences here represent the irradiation profile at the clad/basemetal (C/BM) interface for the reactor vessel. The inner cladding of the reactor vessel attenuates the neutron flux by a factor of 0.949, according to Reference 6.2 (i.e., the flux at the C/BM interface is 5.1% lower than at the inner wetted surface).

References 6.2 and 6.3 identify the fluence assumed in the generation of the current HU/CD curves as 6.702E+18 n/cm$^2$. This supports an ART of 214° at the 1/4 thickness and 171.5°F at the 3/4 thickness (for the Tech Spec amendment, the 3/4 thickness ART will be rounded up to the nearest whole number, or 172°F). The 6.702E+18 value is at the **wetted inner surface.** Therefore, in order to be consistent with the new Westinghouse analysis, it will be reduced by 0.949, to 6.36E+18 n/cm$^2$. This is the fluence at the C/BM interface.

To summarize what we have so far, there is an analysis in place that identifies the fluence at the end of Cycle 10 as 5.28E+18 n/cm$^2$ at the limiting C/BM interface, with an average flux of 8.67E+09 n/cm$^2$-s. The next step is to determine at what burnup the C/BM fluence will reach the analytical limit of 6.36E+18 n/cm$^2$.

$$(6.36E{+}18\,n/cm^2 - 5.28E{+}18\,n/cm^2) \div 8.67E{+}09\,n/cm^2 s \div 3.1557E{+}07\,sec/yr$$
$$= 3.947 \text{ year} = 3.947 \text{ EFPY at } 100\% \text{ capacity factor}$$

The burnup at the end of Cycle 10 is 12.28 EFPY (Reference 6.1).
Therefore, the curves can be extended to a new upper limit of

12.28 EFPY + 3.947 EFPY = 16.227 EFPY, rounded down to 16.2 EFPY

> Note: The above calculation uses the Westinghouse value of 12.28 EFPY for reactor vessel service through the end of Cycle 10 (Reference 6.1). IP3 uses a slightly different methodology to calculate lifetime burnup, with a resulting burnup of 12.31 EFPY at the end of Cycle 10. Regardless of which value is used for Cycle 10 burnup, the new upper limit of 16.2 EFPY remains valid.

### 3.3 Assurance for Current and Future Cycles

The Westinghouse analysis takes us through the end of Cycle 10, but the current cycle is Cycle 11, and future cycles stretch out ahead. There needs to be some assurance in place that the assumptions for flux, upon which the curves are based, will remain valid.

During the Cycle 10/11 outage (R10), the fluence on the reactor vessel inner wall was forced down by the installation of hafnium flux suppressors. These devices take the place of the older pyrex glass burnable absorbers and have the advantage of being reusable from cycle to cycle. In order to support the policy of reactor vessel embrittlement protection, the Authority expects to re-use (or replace, as necessary) the hafnium suppressors every cycle.

The effect of the suppressors can be seen in a comparison of the fuel assembly powers in the corners of the core for Cycles 10 and 11. This can be seen in the design documents for the two cycles (References 6.9 and 6.10) in which assembly powers are shown throughout the core for a wide variety of cycle burnups. In all cases, the power in the corner assemblies is less for Cycle 11 than for Cycle 10, which thereby ensures that the Westinghouse Cycle 10 flux remains limiting for Cycle 11.

For future cycles, a routine re-evaluation will be performed each refueling to ensure that the Cycle 10 assumptions remain limiting (See Section 5.0).

### 3.4 Affected Curves

This calculation affects all curves in the IP3 Tech Specs that are based on either the heatup or the cooldown curves, as provided by ABB-CE. It is important to note that the OPS curves are all based directly on the isothermal cooldown curve (per Reference 6.4). Therefore, if the cooldown curve remains the same, so do all the OPS-related curves. A list of curves appears below.

## 4.0 SUMMARY AND CONCLUSION

The existing Tech Spec curves 3.1.A-1, 3.1.A-2, 3.1.A-5, 3.1.A-6, 3.1-1, 3.1-2 and 4.3-1 may all be revised to reflect an expiration date of 16.2 EFPY.

## 5.0 PROCEDURES AFFECTED BY THIS CALCULATION

None.

However, to ensure continuing validity of these calculations, an ACTS Follow Item is opened stating, "Evaluate neutron flux on the limiting reactor vessel location throughout Cycle 12, and ensure that average cycle flux is less than that for Cycle 10. If average flux exceeds that of Cycle 10, immediately prepare a revision of the heatup/cooldown/OPS Technical Specification curves. Upon completion of this item, open a similar item for the next cycle." (ACTS Item No. 00-49128)

In addition, if the next surveillance capsule (Capsule S, scheduled for removal in the 2001 refueling outage) is analyzed prior to a reactor vessel service life of 16.2 EFPY, then the ART and CF shall be re-evaluated and the expiration dates of the HU/CD/OPS adjusted if necessary (ACTS Item 00-49129).

## 6.0 REFERENCES

6.1 Westinghouse Letter INT-00-211, "Evaluation of Reactor Vessel Flux and Fluence Calculations"

6.2 PENG-98-014, "Revised ART Determination for Indian Point Unit 3," ABB-Combustion Engineering, January 23, 1998

6.3 MISC-MPS-ER-005, "Final Report on Pressure-Temperature Limits for Indian Point Unit 3 Nuclear Power Plant," ABB-Combustion Engineering, July 1991

6.4 IP3 Calculation IP3-CALC-RCS-02444, Rev 1, "Generation of All Curves Associated with the Pressure-Temperature Limit Report"

6.5 IP3 Technical Specifications Sections 3.1 and 4.3

6.6 Reg Guide 1.99 Rev 2

6.7 ASME Boiler and Pressure Vessel Code Section III, Appendix G, "Protection Against Nonductile Failure"

6.8 WCAP 11057, Rev 1, "Indian Point Unit 3 Reactor Vessel Fluence and RT-PTS Evaluations for Consideration of Life Extension" Westinghouse Electric Corp, June 1989

6.9 WCAP-14889, Rev 0, "Nuclear Parameters and Operations Package for Indian Point Unit 3, Cycle 10," Westinghouse Electric Corp, June 1997

6.19 WCAP-15303, Rev 0, "Nuclear Parameters and Operations Package for Indian PointUnit 3, Cycle 11," Westinghouse Electric Corp, October 1999