ATTACHMENT II TO IPN-93-[illegible]

IP3 CALCULATION - IP3-CALC-RCS-00873
PREDICTED EOL USE FOR BELTLINE PLATE B2803-1

New York Power Authority
Indian Point 3 Nuclear Power Plant
Docket No. 50-286

CALCULATION CONTROL SHEET

CALC. NO. IP3-CALC-RCS-00873 REV. ______ IP3 [X] JAF [X] (JB 11/1/93)

MOD/TASK NO. Predicted EOL USE for Beltline Plate

QA CATEGORY OF CALCULATION: 1

CALCULATION TYPE: PRELIMINARY: ______ FINAL: X

PROJECT/TASK: Reactor Vessel Surveillance Program

SYSTEM NO./NAME: Reactor Coolant System

TITLE: Predicted EOL USE for Beltline Plate B2803-1

	NAME	SIGNATURE	DATE
DESIGN ENG:	J. Lafferty	Joe Lafferty	10/29/93
PREPARER:	J. Lafferty	[signature]	10/28/93
CHECKER:	F. Gumble	Floyd Gumble	11/1/93
VERIFIED: N/A [ ]			
APPROVED:	R. Lauman	[signature]	11/1/93

PROBLEM/OBJECTIVE/METHOD

Perform a calculation which predicts the Upper Shelf Energy (USE) for Plate B2803-1.

DESIGN BASIS/ASSUMPTIONS

- IP3 Surveillance Program
- Reg. Guide 1.99, R/2, Methodology
- Fluence measurements based on reported values from the PTS report [4].

SUMMARY/CONCLUSIONS

IP3 R.V. Beltline Plate B2803-1 has an end of life upper shelf energy of 54 ft-lbs. This is above the 50 ft-lbs threshold value required by 10 CFR, Part 50, Appendix G.

REFERENCES

See enclosed reference section.

AFFECTED SYSTEMS/COMPONENTS/DOCUMENTS

- Reactor Vessel Beltline Material
- Core Design

☐ VOIDED OR

☐ SUPERSEDED BY: ______________________ (CALC. NO.)

New York Power Authority

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Preliminary	Prepared by J. Lafferty Date 10/29/93
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1.0 PURPOSE

The purpose of this calculation is to determine the predicted end of life (EOL) upper shelf energy (USE) for Beltline Plate B2803-1.

This request was made by the NRC in GL-92-01 Request for Additional Information (RAI) 1.

2.0 ASSUMPTIONS

1) License expiration December 2015 [10]
2) End of Life ¼ T fluence is calculated from a surface fluence of 1.04 E19 N/Cm^2 (E > 1.0 Mev.) [4]
3) EOL Fluence at the ¼ T location is calculated using Reg. 1.99 R/2 methods
4) EOL Fluence ≈ 26 EFPY = 6.20 E18 N/Cm^2, (E > 1.0 Mev.)
5) Initial USE 72 ft-lbs [5]
6) Copper Content 0.19% [5]
7) Vessel wall is 8 5/8" thick

3.0 REFERENCES

1. WCAP-11815 "Analysis of Capsule Z from the NYPA IP3 Reactor Vessel Radiation Program", Westinghouse Electric Corp., March 1988.

2. Report SIR-88-016 R/0, "Evaluation of Capsule Z from the IP3 Reactor Pressure Vessel Surveillance Program", Structural Integrity Associates, San Jose, CA, January 1989.

3. WCAP 11045, R/1 - "Indian Point Unit 3 Reactor Vessel Fluence and RT_{PTS} Evaluations", Westinghouse Electric Corporation, 6/1989.

4. WCAP 11057, R/1 - "Indian Point Unit 3 Reactor Vessel Fluence & RT_{PTS} Evaluations for considerations for Life Extension", Westinghouse Electric Corporation, June 1989.

5. WCAP 13587, R/1, "Reactor Vessel Upper Shelf Energy Bounding Evaluation for Westinghouse Pressurized Water Reactors", Westinghouse Electric Corporation, September 1993.

6. NYPA Letter IPN-92-031/JPN-92-037, Response to Generic Letter 92-01, Rev. 1, July 9, 1992.

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3.0 REFERENCES (continued)

7. Letter report FDRT-SRPLO-191/93, "Indian Point Unit 3 Upper Shelf Energy Data", Westinghouse Electric Corporation, 10/93.

8. ASTM-E185, "Standard Practice for Conducting Surveillance Test for Light Water Cooled Nuclear Power Plant Vessels", 1982.

9. USNRC Regulatory Guide 1.99 R/2, "Radiation Embrittlement of Reactor Vessel Materials", April 1988.

10. NRC letter to Ralph Beedle from Nicola Conicella, "Issuance of Amendment for Indian Point Generating Unit No. 3", Amendment No. 124, License No. DPR-64, dated July 15, 1992.

4.0 CALCULATION

Initial USE for Plate B2803-1 = 72 ft-lbs [7].

An excerpt from ASTM-E185, [8] follows which explains what was used to conclude that 72 ft-lbs is the unirradiated upper shelf energy for Plate # B2803-1.

ASTM-E185 [8] defines the initial upper shelf energy as the average energy value for three Charpy specimens whose temperature is above the upper end of the Charpy V-notch curve transition region. For specimens tested in sets of three, the set having the highest average may be regarded as defining the material's upper shelf energy.

Data from Table 7 [7] is provided to show the Charpy V-notch test results which were used to calculate the unirradiated upper shelf energy of Plate B2803-1.

Temperature (°F)	Energy (ft-lbs)	% Shear
210	64.5	100%
210	74	100%
210	77	100%
	Avg. = 72 ft-lbs	

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4.0 CALCULATION (continued)

The following section documents the calculation of fluence at the ¼ T location:

The equation for attenuation, which calculates reactor through-wall fluence values, is:

$f = f_{surf}\ (e^{-0.24x})$ [9]

f_{surf} = *is the peak measured fluence at the vessel inside surface to clad interface* [4]

$f_{surf} = 1.04\ E19$

$f = 1.04\ E19\ (e^{-.24 \times 2.156})$

$f = 6.20\ E18$

EOL USE = Initial USE - % decrease (Initial USE)

EOL USE = 72 - .255 (72)

EOL USE = 54 ft-lbs

For an EOL fluence of 6.20 E18 N/Cm^2, material copper content = .19%

The predicted % decrease in USE = 25%; EOL USE = 54 ft-lbs

Figure I graphs the % decrease in USE for the reported EOL fluence value of 54 ft-lbs.

The end of life fluence at ¼ T is 6.20 E18 N/Cm^2 as compared to the 6.44 x 10^{18} N/Cm^2 reported in NYPA's response to GL-92-01 [6], Table VIII.

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5.0 SUMMARY/DISCUSSION

A summary of fluence values for ¼ T location are presented below for comparison. The differences in ¼ T fluence values reported come from the change in the methods used in the calculation of ID vessel fluence and the implementation of low leakage core patterns. This comparison provides the EOL USE as a function of fluence values.

TABLE I

	EOL ID Fluence (N/Cm^2)	EFPY	EOL ¼ T Fluence (N/Cm^2)
Capsule Z [1]	1.08 E19	22.5	5.69 E18
RT_{PTS} Report [4]	8.81 E18	21.88	5.25 E18
GL-92-01 Response [3]	1.08 E19	21.88	6.44 E18
PT_{PTS} Report for 40 years [2]	1.04 E19	26.61	6.20 E18

For the purpose of this calculation, end of life fluence is calculated for a plant license expiration date of 2015. As of December 1993, we expect to have approximately 9 EFPY of plant operation. Through December of 2015, twenty two more calendar years of operation are available, assuming an 80% capacity factor provides 17.6 EFPY through end of license. A total of 26.6 EFPY's is an appropriate assumption for the expected end of license fluence. The PTS report [4] used actual measured dosimetry results though Cycle 7. The report also considers implementation of low core leakage patterns. This report provides a 26.6 EFPY fluence at the vessel wall to clad interface surface of 1.04 E19 N/Cm^2. This 26.6 EFPY fluence meets the GL-92-01, 80% capacity factor rule and is acceptable.

For the purpose of comparison, the upper shelf energies are shown below using the fluence values presented in Table I.

TABLE II

Initial USE (ft-lbs)	Fluence (N/Cm^2)	% Decrease	EOL USE (ft-lbs)
72	6.44 E18 [2]	25.5	54
72	5.25 E18 [7]	24	55
72	6.20 E18	25	54

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5.0 SUMMARY/DISCUSSION (continued)

This calculation shows the sensitivity to a fairly significant change in end of life fluence causes a relatively small change in the predicted end of life upper shelf energy for beltline material.

The "Indian Point 3 Reactor Vessel Fluence and RT_{PTS} Evaluation" [4] reports an accurate fluence value since it considers low core leakage patterns and a bias factor of 1.086. The reason for the difference in fluence at the ¼ T location is that the more recent fluence calculations consider low leakage core patterns and a bias factor of 1.086. This bias factor accounts for differences observed between cycle specific calculations and the results of neutron dosimetry for the first three capsules removed from the IP3 reactor.

This calculation also reflects the current license expiration date of 2015. The license expiration date changed from 2009 to 2015 after the response to GL-92-01 was submitted.

Therefore, 26.6 EFPY instead of 21.8 EFPY was used to calculate end of life fluence. It should be further noted that the WOG has performed generic bounding evaluations as per the proposed ASME Section XI, Appendix X, which demonstrate that IP3 reactor vessel beltline material has a margin of safety equivalent to that required by Appendix G of the ASME Code. The lowest upper shelf energy that satisfies the Appendix X requirements is 43 ft-lbs for IP3.

6.0 ATTACHMENTS

Figure 1

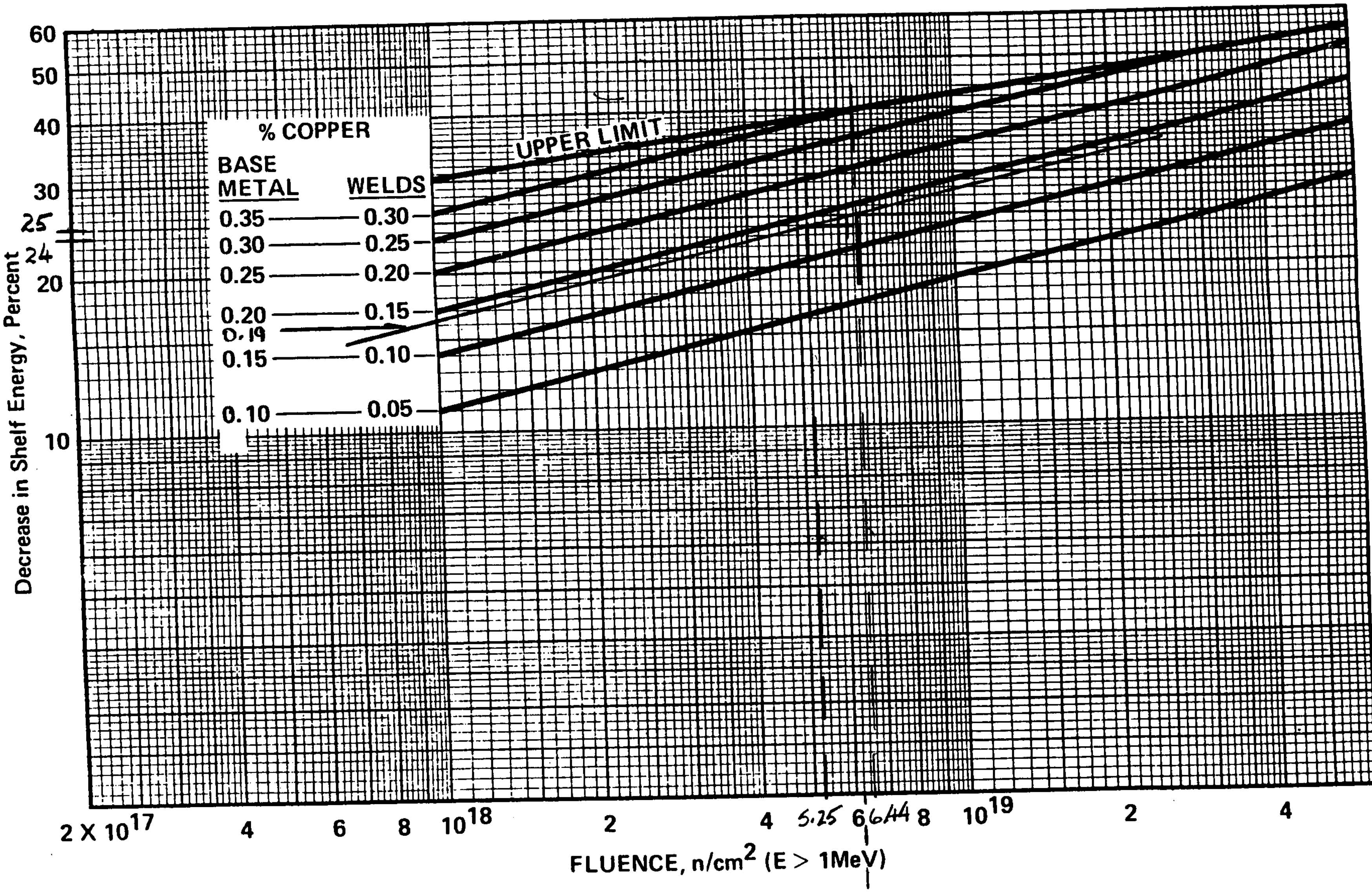

FIGURE 8 Predicted Decrease in Shelf Energy as a Function of Copper Content and Fluence

FOR PLATE B2803-1

Capsule Z WCAP 11815

TABLE 6-12

SUMMARY OF FAST NEUTRON EXPOSURE PROJECTIONS FOR THE INDIAN POINT UNIT 3 REACTOR VESSEL

	5.55 EFPY		22.5 EFPY	
	Φ (E > 1.0 MeV) (n/cm^2)	(dpa)	Φ (E > 1.0 MeV) (n/cm^2)	(dpa)
Vessel IR	3.13×10^{18}	5.10×10^{-3}	1.08×10^{19}	1.75×10^{-2}
Vessel 1/4T	1.65×10^{18}	3.26×10^{-3}	5.69×10^{18}	1.11×10^{-2}
Vessel 1/2T	7.51×10^{17}	1.97×10^{-3}	2.60×10^{18}	6.75×10^{-3}
Vessel 3/4T	3.29×10^{17}	1.13×10^{-3}	1.14×10^{18}	3.87×10^{-3}
Vessel OR	1.32×10^{17}	5.41×10^{-4}	4.95×10^{17}	1.85×10^{-3}

Note: Date are based on the extrapolation of Capsule Z dosimetry results to vessel locations.

PB Report

TABLE II.2-4

INDIAN POINT UNIT 3
FAST NEUTRON (E > 1.0 Mev) EXPOSURE AT THE
REACTOR VESSEL INNER RADIUS - 45 DEGREE AZIMUTHAL ANGLE (a)

IRRADIATION INTERVAL	ELAPSED IRRADIATION TIME (EFPY)	AVG. FLUX (b) (n/cm2-sec)	BELTLINE REGION CUMULATIVE FLUENCE (n/cm2) PLANT SPECIFIC (b)	REFERENCE (c)
CY-1	1.37	1.94E+10	8.40E+17	1.00E+18
CY-2	2.23	2.53E+10	1.52E+18	1.63E+18
CY-3	3.29	2.09E+10	2.22E+18	2.41E+18
CY-4	4.41	1.67E+10	2.81E+18	3.23E+18
CY-5	5.55	1.38E+10	3.31E+18	4.06E+18
CY-6 (d)	6.73	1.30E+10	3.79E+18	4.93E+18
CY7-EOL (e)	21.88	1.05E+10	8.81E+18	1.60E+19
CY7-40yrs (e)	26.61	1.05E+10	1.04E+19	1.95E+19
CY7-60yrs (e)	41.61	1.05E+10	1.53E+19	3.05E+19

2009

(a) Applicable to longitudinal weld 3-042C in the lower shell, the intermediate to lower shell circumferential weld 9-042, and all shell plates

(b) Includes an analytical bias factor of 1.086

(c) Reference fast neutron flux = 2.32E+10 n/cm2-sec at 3025 MWt

(d) Current neutron fluences are defined as of the end of cycle 6

(e) Fuel cycle projections are based on the average neutron flux for cycle 7 and an assumed capacity factor of 0.75

- The adjusted reference temperature for weld metal would be close to the 200°F limit,

- Upper shelf energy for the limiting plate would always be precariously close to the 50 ft-lb limit, and

- Projections of USE for weld metal also indicated an end of life value very near the 50 ft-lb limit.

Several actions were taken as a result. The Power Authority decided to test the WOL specimens removed with capsules T and Y to provide a quantitative determination of the fracture toughness of the Indian Point-3 reactor vessel materials. These results are reported in WCAP-10300-3, Volume 3 [13] and reviewed in [14]. The conclusion from these measurements is that the vessel material continues to retain sufficient fracture toughness even after irradiation to the 8.05 X 10^{18} n/cm^2 level. This corresponds to 1/4T at longer than the design life. The Power Authority also embarked upon a program to reduce the neutron exposure of the reactor vessel by a modified core loading pattern resulting in a low leakage core. This activity was initiated following Cycle 5 and continues to be used. A design end of life fluence of 6.44 X 10^{18} n/cm^2 at 1/4T is expected based on this current fuel loading practice.

Capsule Z Results

Capsule Z, a Type I capsule, was removed from the Indian Point-3 vessel during May, 1987, following cycle 5 (5.55 effective full power years). Like the previous capsules, mechanical properties test specimens and dosimetry materials were evaluated by Westinghouse [15]. This high lead factor capsule had accumulated 1.07 X 10^{19} n/cm^2 (E>1MeV), the equivalent of 0.0177 displacements per atom. This exposure is essentially identical to the projected end of life (22.5 EFPY) fluence at the vessel

R-34-15

Designation: E 185 – 82$^{\epsilon2}$

AMERICAN SOCIETY FOR TESTING AND MATERIALS
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If not listed in the current combined index, will appear in the next edition.

Standard Practice for CONDUCTING SURVEILLANCE TESTS FOR LIGHT-WATER COOLED NUCLEAR POWER REACTOR VESSELS, E 706 (IF)[1]

This standard is issued under the fixed designation E 185; the number immediately following the designation indicates the year of original adoption or, in the case of revision, the year of last revision. A number in parentheses indicates the year of last reapproval. A superscript epsilon (ε) indicates an editorial change since the last revision or reapproval.

$^{\epsilon1}$ NOTE—Section 9.2.3 was corrected editorially and the designation date was changed July 1, 1982.
$^{\epsilon2}$ NOTE—The title was changed editorially in July 1985.

1. Scope

1.1 This practice covers procedures for monitoring the radiation-induced changes in the mechanical properties of ferritic materials in the beltline of light-water cooled nuclear power reactor vessels. This practice includes guidelines for designing a minimum surveillance program, selecting materials, and evaluating test results.

1.2 This practice was developed for all light-water cooled nuclear power reactor vessels for which the predicted maximum neutron fluence ($E > 1$ MeV) at the end of the design lifetime exceeds 1×10^{21} n/m^2 (1×10^{17} n/cm^2) at the inside surface of the reactor vessel.

2. Applicable Documents

2.1 *ASTM Standards:*

A 370 Methods and Definitions for Mechanical Testing of Steel Products[2]

E 8 Methods of Tension Testing of Metallic Materials[3]

E 21 Recommended Practice for Elevated Temperature Tension Tests of Metallic Materials[3]

E 23 Methods for Notched Bar Impact Testing of Metallic Materials[3]

E 208 Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels[3]

E 482 Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance[4]

E 560 Recommended Practice for Extrapolating Reactor Vessel Surveillance Dosimetry Results[4]

2.2 *American Society of Mechanical Engineers Standard*: Boiler and Pressure Vessel Code, Sections III and XI[5]

3. Significance and Use

3.1 Predictions of neutron radiation effects on pressure vessel steels are considered in the design of light-water cooled nuclear power reactors. Changes in system operating parameters are made throughout the service life of the reactor vessel to account for radiation effects. Because of the variability in the behavior of reactor vessel steels, a surveillance program is warranted to monitor changes in the properties of actual vessel materials caused by long-term exposure to the neutron radiation and temperature environment of the given reactor vessel. This practice describes the criteria that should be considered in planning and implementing surveillance test programs and points out precautions that should be taken to ensure that: (*1*) capsule exposures can be related to beltline exposures, (*2*) materials selected for the surveillance program are samples of those materials most likely to limit the operation of the reactor vessel, and (*3*) the tests yield results useful for the evaluation of radiation effects on the reactor vessel.

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[1] This practice is under the jurisdiction of ASTM Committee E-10 on Nuclear Technology and Applications.
Current edition approved July 1, 1982. Published September 1982. Originally published as E 185 – 61 T. Last previous edition E 185 – 79.

[2] *Annual Book of ASTM Standards*, Vol 01.04.

[3] *Annual Book of ASTM Standards*, Vol 03.01.

[4] *Annual Book of ASTM Standards*, Vol 12.02.

[5] Available from the American Society of Automotive Engineers, 345 E. 47th St., New York, N. Y. 10017.

3.2 The design of a surveillance program for a given reactor vessel must consider the existing body of data on similar materials in addition to the specific materials used for that reactor vessel. The amount of such data and the similarity of exposure conditions and material characteristics will determine their applicability for predicting the radiation effects. As a large amount of pertinent data becomes available it may be possible to reduce the surveillance effort for selected reactors by integrating their surveillance programs.

4. Definitions

4.1 *adjusted reference temperature*—the reference temperature adjusted for irradiation effects by adding to RT_{NDT} the transition temperature shift (see 4.15).

4.2 *base metal (parent material)*—as-fabricated plate material or forging material other than a weldment or its corresponding heat-affected-zone (HAZ).

4.3 *beltline*—the irradiated region of the reactor vessel (shell material including weld regions and plates or forgings) that directly surrounds the effective height of the active core, and adjacent regions that are predicted to experience sufficient neutron damage to warrant consideration in the selection of surveillance material.

4.4 *EOL*—end-of-life; the design lifetime in terms of years; effective full power years; or neutron fluence.

4.5 *index temperature*—that temperature corresponding to a predetermined level of absorbed energy, lateral expansion, or fracture appearance obtained from the average (best fit) Charpy transition curve.

4.6 *fraction strength*—in a tensile test, the load at fracture divided by the initial cross-sectional area of the test specimen.

4.7 *fracture stress*—in a tensile test, the load at fracture divided by the cross-sectional area of the test specimen at time of fracture.

4.8 *heat-affected-zone* (HAZ)—plate material or forging material extending outward from, but not including, the weld fusion zone in which the microstructure of the base metal has been altered by the heat of the welding process.

4.9 *lead factor*—the ratio of the neutron flux density at the location of the specimens in a surveillance capsule to the neutron flux density at the reactor pressure vessel inside surface at the peak fluence location.

4.10 *neutron fluence*—the time integrated neutron flux density, expressed in neutrons per square metre or neutrons per square centimetre.

4.11 *neutron flux density*—a measure of the intensity of neutron radiation within a given range of neutron energies; the product of the neutron density and velocity, measured in neutrons per square metre-second or neutrons per square centimetre-second.

4.12 *neutron spectrum*—the distribution of neutrons by energy levels impinging on a surface, which can be calculated based on analysis of multiple neutron dosimeter measurements, on the assumption of a fission spectrum, or from a calculation of the neutron energy distribution.

4.13 *nil-ductility transition temperature* (T_{NDT})—the maximum temperature at which a standard drop weight specimen breaks when tested in accordance with Method E 208.

4.14 *reference temperature* (RT_{NDT})—See subarticle NB-2300 of the ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components."

4.15 *transition temperature shift* (ΔRT_{NDT}) *or adjustment of reference temperature*—the difference in the 41-J (30-ft·lbf) index temperatures from the average Charpy curves measured before and after irradiation.

4.16 *transition region*—the region on the transition temperature curve in which toughness increases rapidly with rising temperature. In terms of fracture appearance, it is characterized by a rapid change from a primarily cleavage (crystalline) fracture mode to primarily shear (fibrous) fracture mode.

4.17 *Charpy transition curve*—a graphic presentation of Charpy data, including absorbed energy, lateral expansion, and fracture appearance, extending over a range including the lower shelf energy (< 5 % shear), transition region, and the upper shelf energy (> 95 % shear).

4.18 *upper shelf energy level*—the average energy value for all Charpy specimens (normally three) whose test temperature is above the upper end of the transition region. For specimens tested in sets of three at each test temperature, the set having the highest average may be regarded as defining the upper shelf energy.

IP3 ☒

~~JAF ☒~~ FG

INDEPENDENT DESIGN VERIFICATION CONTROL SHEET

VERIFICATION OF: IP3-CALC-RCS-00873
Document Title/Number

SUBJECT: Predicted EOL USE for Beltline Plate B2803-1

MOD/TASK NUMBER (If Applicable): N/A

QA CATEGORY: I

DISCIPLINE:	ELEC	MECH	C/S	I&C	FIRE PROTECT	OTHERS (SPECIFY)
Check as required	☐	☐	☐	☐	☐	☑ Reactor Eng.
METHOD USED (1)						AC
VERIFIER'S NAME:						Floyd Gumble
VERIFIER'S INITIALS/DATE:						FWG 11/1/93

APPROVED BY: [signature] Date: 11/1/93

REMARKS/SCOPE OF VERIFICATION:

Verified calculation by performing independent determination of EOL conditions and applying NRC guidelines to the result.

(1) Methods of Verification: Design Review (DR), Alternate Calculations (AC), Qualification Test (QT)

DESIGN VERIFICATION CHECKLIST

DESIGN REVIEW METHOD

IP3 [ ]
JAF [X]

VERIFICATION OF: IP3-CALC-RCS-00873

Document/Title/Number

SUBJECT: Predicted EOL USE for Beltline Plate B2803-1

MOD/TASK NO: (If Applicable)

DESIGN VERIFIER: Floyd W. Humble - Senior Reactor Eng - 11/1/93
Signature/Title/Date

	ELEC	MECH	C/S	I&C	FIRE PROTECT	OTHER (SPECIFY)
Check as Required	[ ]	[ ]	[ ]	[ ]	[ ]	[✓] Reactor Eng.

Yes/Not Applicable

1. Were the inputs correctly selected and incorporated into the design? — Yes/NA
2. Are the physical and functional characteristics of the proposed design within the approved design basis of the system(s) structure(s) or component(s)? — Yes/NA
3. Does the proposed design incorporate license commitments? — Yes/NA
4. Are assumptions necessary to perform the design activity adequately described and reasonable: Where necessary, are the assumptions identified for subsequent reverifications when the detailed design activities are completed? — Yes/NA
5. Are the appropriate quality and quality assurance requirements specified? e.g., safety classification. — Yes/NA
6. Are the applicable codes, standards and regulatory requirements including issue and addenda properly identified and are their requirements for design met? — Yes/NA
7. Have applicable construction and operating experience been considered? — Yes/NA

		Yes/Not Applicable
8.	Have the design interface requirements been satisfied?	Yes/NA
9.	Was an appropriate design method used?	Yes/NA
10.	Is the output reasonable compared to inputs?	Yes/NA
11.	Are the specified parts, equipment and processes suitable for the required application?	Yes/NA
12.	Are the specified materials compatible with each other and the design environmental conditions to which the materials will be exposed?	Yes/NA
13.	Have adequate maintenance features and requirements been satisfied?	Yes/NA
14.	Are accessibility and other design provisions adequate for performance of needed maintenance and repair?	Yes/NA
15.	Has adequate accessibility been provided to perform the in-service inspection expected to be required during the plant life?	Yes/NA
16.	Has the design properly considered radiation exposure to the public and plant personnel? (ALARA/cobalt reduction)	Yes/NA
17.	Are the acceptance criteria incorporated in the design documents sufficient to allow verification that design requirements have been satisfactorily accomplished?	Yes/NA
18.	Have adequate pre-operational and subsequent periodic test requirements been appropriately specified?	Yes/NA
19.	Are adequate handling, storage, cleaning and shipping requirements specified?	Yes/NA

DESIGN VERIFICATION CHECKLIST
DESIGN REVIEW METHOD

	Yes/Not Applicable
20. Are adequate identification requirements specified?	Yes/NA
21. Are the conclusions drawn in the Safety Evaluation fully supported by adequate discussion in the test or Safety Evaluation itself?	Yes/NA
22. Are necessary procedural changes specified, and are responsibilities for such changes clearly delineated?	Yes/NA
23. Are requirements for record preparation, review, approval, retention, etc., adequately specified?	Yes/NA
24. Have supplemental reviews by other engineering disciplines (seismic, electrical, etc.) been performed on the integrated design package?	Yes/NA
25. Have the drawings, sketches, calculations, etc., included in the integrated design package been reviewed?	Yes/NA
26. Have reviews been performed to identify any effect on the Check Valve Maintenance Program?	Yes/NA
27. Does the design for check valves meet the intents of INPO SOER 86-03?	Yes/NA
28. Is the plant reference simulator physical and functional fidelity affected and its design change been factored into the cost?	Yes/NA
29. References used as part of the design review which are not listed as part of the design calculation/analysis.	Yes/NA

____

____

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ADDITIONAL INFORMATION PROVIDED BY NSSS VENDOR
TABLES I THROUGH VII

New York Power Authority
Indian Point 3 Nuclear Power Plant
Docket No. 50-286

TABLE I

Intermediate Shell, B2802-1 (Ht. A5394-2)

Temperature (°F)	Energy (ft-lbs)	% Shear	Upper Shelf (ft-lbs)
-100	2	5	
-100	3	9	
-100	4	5	
- 50	5.5	9	
- 50	6	9	
- 50	8	14	
10	33.5	34	
10	34	29	
10	37	29	
60	48	50	
60	60	47	
60	63	50	
110	71	75	
110	85	75	
110	94	80	
210	97	100	102
210	98	100	
210	110	100	
550	104	100	
550	104	100	

TABLE II

Intermediate Shell, B2802-2 (Ht. AO516-2)

Temperature (°F)	Energy (ft-lbs)	% Shear	Upper Shelf (ft-lbs)
-100	4.5	5	
-100	4.5	5	
-100	5.5	5	
- 50	11	18	
- 50	12.5	18	
- 50	14	18	
10	30	33	
10	35	38	
10	47	40	
60	52	57	
60	64	58	
60	66	55	
110	70	100	
110	86	84	
110	100	100	
210	86	100	
210	102	100	97
210	104	100	
550	92	100	
550	110	100	

TABLE III

Intermediate Shell, B2802-3 (Ht. B5391-2)

Temperature (°F)	Energy (ft-lbs)	% Shear	Upper Shelf (ft-lbs)
-100	2.5	5	
-100	4	9	
-100	4	5	
- 50	9	13	
- 50	9	14	
- 50	8	16	
10	30.5	29	
10	31	34	
10	32	38	
60	40	42	
60	47	47	
60	60	55	
110	72	74	
110	77	81	
110	92	87	
210	85	100	
210	97	100	95
210	102	100	
550	95.5	100	
550	100	100	

TABLE IV

Lower Shell, B2803-1 (Ht. AO495-2)

Temperature (°F)	Energy (ft-lbs)	% Shear	Upper Shelf (ft-lbs)
40	30	37	
40	22	37	
75	34	54	
75	37.5	46	
160	68.5	100	
160	71.5	100	
160	66	100	72
210	64.5	100	
210	74	100	
210	77	100	

TABLE V

Lower Shell, B2803-2 (Ht. C1397-3)

Temperature (°F)	Energy (ft-lbs)	% Shear	Upper Shelf (ft-lbs)
-100	3	5	
-100	4	5	
-100	3	5	
- 20	43.5	35	
- 20	51.5	40	
- 20	34	40	
40	40	45	
40	48	60	
40	35	45	
75	77	30	
75	71	30	
75	69	30	
100	82	80	
100	84	80	
100	78	80	
210	96	100	94
210	89	100	
210	97	100	

TABLE VI

Lower Shell, B2803-3 (Ht. A0512-2)

Temperature (°F)	Energy (ft-lbs)	% Shear	Upper Shelf (ft-lbs)
-20	9	5	
-20	7	5	
-20	11	9	
40	29.5	32	
40	24	33	
40	17.5	21	
75	34	41	
75	41	47	
75	33.5	42	
125	54	47	
125	59	51	
125	46	41	
160	65	100	
160	66	100	
160	59.5	100	
210	62	100	68
210	70	100	
210	65	100	
210	70.5	100	
210	68	100	
210	70	100	

TABLE VII

Surveillance Weld Metal

Temperature (°F)	Energy (ft-lbs)	% Shear	Upper Shelf (ft-lbs)
-150	5	5	
-150	2	5	
-150	4.5	9	
-100	29	20	
-100	18	18	
-100	25.5	23	
- 50	35	40	
- 50	33	47	
- 50	32.5	40	
- 35	78	64	
- 35	69.5	67	
- 35	54.5	40	
- 20	87	77	
- 20	82	77	
- 20	89	81	
10	100	81	
10	105	82	
10	113.5	100	
60	115	100	
60	119	100	
60	121.5	100	
160	124	100	120
160	125	100	
160	112	100	

RECORDS MANAGEMENT SERVICES

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555
July 15, 1992

Docket No. 50-286

Mr. Ralph E. Beedle
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Beedle:

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 (TAC NO. M76970)

The Commission has issued the enclosed Amendment No. 124 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3 in response to your application transmitted by letter dated June 11, 1990, as supplemented June 18, 1991, February 11, 1992, and May 13, 1992.

The amendment extends the expiration date of the facility operating license from August 13, 2009, to December 12, 2015.

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

Sincerely,

N. F. Conicella

Nicola F. Conicella, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.124 to DPR-64
2. Safety Evaluation

cc w/enclosures:
See next page

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