4.0 TIME-LIMITED AGING ANALYSES

4.1 IDENTIFICATION OF TIME-LIMITED AGING ANALYSES

Time-limited aging analyses are defined in 10 CFR 54.3.

> *Time-limited aging analyses*, for the purposes of this part, are those licensee calculations and analyses that:
>
> (1) Involve systems, structures, and components within the scope of license renewal, as delineated in §54.4(a);
>
> (2) Consider the effects of aging;
>
> (3) Involve time-limited assumptions defined by the current operating term, for example, 40 years;
>
> (4) Were determined to be relevant by the licensee in making a safety determination;
>
> (5) Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in §54.4(b); and
>
> (6) Are contained or incorporated by reference in the CLB.

Section 10 CFR 54.21(c) requires a list of time-limited aging analyses (TLAA) as part of the application for a renewed license. Section 10 CFR 54.21(c)(2) requires a list of current exemptions to 10 CFR 50 based on TLAA as part of the application for a renewed license.

> §54.21 Contents of application — technical information.
>
> (c) An evaluation of time-limited aging analyses.
>
> (1) A list of time-limited aging analyses, as defined in §54.3, must be provided. The applicant shall demonstrate that—
>
> (i) The analyses remain valid for the period of extended operation;
>
> (ii) The analyses have been projected to the end of the period of extended operation; or
>
> (iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.
>
> (2) A list must be provided of plant-specific exemptions granted pursuant to 10 CFR 50.12 and in effect that are based on time-limited aging analyses as defined in §54.3. The applicant shall provide an evaluation that justifies the continuation of these exemptions for the period of extended operation.

4.1.1 Identification of TLAA

The process used to identify the time-limited aging analyses is consistent with the guidance provided in NEI 95-10, *Industry Guidelines for Implementing the Requirements of 10 CFR 54 – The License Renewal Rule*, Revision 6, June 2005. Calculations and analyses that potentially meet the definition of 10 CFR 54.3 were identified by searching CLB documents including the following.

- Technical Specifications and Bases
- UFSAR
- Technical Requirements Manual
- Westinghouse Commercial Atomic Power (WCAP) topical reports referenced in the UFSAR and in licensing correspondence with the NRC
- docketed licensing correspondence
- Fire Protection Program documents
- NRC safety evaluation reports
- ASME Section XI Inservice Inspection program

Industry documents that list generic time-limited aging analyses were also reviewed to provide additional assurance of the completeness of the plant-specific list. These documents included NEI 95-10; NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants*, Revision 1, September 2005; NUREG-1801, *Generic Aging Lessons Learned (GALL) Report*, Revision 1, September 2005; and NRC safety evaluation reports related to license renewal applications by other PWR licensees.

Table 4.1-1 and Table 4.1-2 provide a summary listing of the TLAAs.

4.1.2 Identification of Exemptions

IPEC exemptions were identified by searching CLB documents including the following.

- Technical Specifications
- UFSAR (IP2 and IP3)
- Fire Protection Program Documents
- NRC Correspondence

No IPEC exemptions are based on time-limited aging analyses.

Table 4.1-1
List of IP2 TLAA and Resolution

TLAA Description	Resolution Option	Section
Reactor Vessel Neutron Embrittlement Analyses		
Charpy upper-shelf energy	Analyses projected 10 CFR 54.21(c)(1)(ii)	4.2.2
Pressure/temperature limits	P-T limit curves managed 10 CFR 54.21(c)(1)(iii)	4.2.3
Low temperature overpressure protection (LTOP)	LTOP limits managed 10CFR54.21(c)(1)(iii)	4.2.4
Pressurized thermal shock	Analysis projected 10 CFR 54.21(c)(1)(ii)	4.2.5
Metal Fatigue Analyses		
Reactor vessel	Analyses remain valid 10 CFR 54.21(c)(1)(i)	4.3.1.1
Reactor vessel internals	Analyses remain valid 10 CFR 54.21(c)(1)(i)	4.3.1.2
Pressurizer	Analyses remain valid 10 CFR 54.21(c)(1)(i)	4.3.1.3
Pressurizer insurge/outsurge transients	Analyses remain valid 10 CFR 54.21(c)(1)(i)	4.3.1.3
Steam generator	Analyses remain valid 10 CFR 54.21(c)(1)(i)	4.3.1.4
Reactor coolant pump	Analyses remain valid 10 CFR 54.21(c)(1)(i)	4.3.1.5
Control rod drive mechanisms	Analyses remain valid 10 CFR 54.21(c)(1)(i)	4.3.1.6
Regenerative letdown heat exchanger	Analyses remain valid 10 CFR 54.21(c)(1)(i)	4.3.1.7
Class 1 piping and in-line components—ANSI B31.1 piping	Analyses remain valid 10 CFR 54.21(c)(1)(i)	4.3.1.8
Class 1 piping and in-line components—pressurizer surge line	Analyses remain valid 10 CFR 54.21(c)(1)(i)	4.3.1.8

Table 4.1-1
List of IP2 TLAA and Resolution (Continued)

TLAA Description	Resolution Option	Section
Class 1 piping and in-line components—thermowells	Analyses remain valid 10 CFR 54.21(c)(1)(i)	4.3.1.8
Class 1 piping and in-line components—charging system	Analysis will be updated as part of environmental fatigue evaluation. See Section 4.3.3.	4.3.1.8
Class 1 piping and in-line components—loop 3 accumulator nozzle	Analyses remain valid 10 CFR 54.21(c)(1)(i)	4.3.1.8
Non-Class 1 piping and in-line components	Analyses remain valid 10 CFR 54.21(c)(1)(i)	4.3.2
Non-Class 1, non-piping components - residual heat removal heat exchanger	Analyses remain valid 10 CFR 54.21(c)(1)(i)	4.3.2
Effects of reactor water environment on fatigue life	Analyses remain valid 10 CFR 54.21(c)(1)(i) OR Aging effect managed 10 CFR 54.21(c)(1)(iii)	4.3.3
Environmental Qualification Analyses Of Electrical Equipment	Aging effect managed 10 CFR 54.21(c)(1)(iii)	4.4
Concrete Containment Tendon Prestress Analyses	IPEC does not have pre-stressed tendons in the containment structures.	4.5
Containment Liner Plate and Penetrations Fatigue Analyses		
Containment penetration (feedwater line #22) fatigue analysis	Analyses remain valid 10 CFR 54.21(c)(1)(i)	4.6
Other TLAA		
Leak before break	Analysis remains valid 10 CFR 54.21(c)(1)(i)	4.7.2
Steam generator flow-induced vibration (tube wear)	Analyses remain valid 10 CFR 54.21(c)(1)(i)	4.7.3

Table 4.1-2
List of IP3 TLAA and Resolution

TLAA Description	Resolution Option	Section
Reactor Vessel Neutron Embrittlement Analyses		
Charpy upper-shelf energy	Analyses projected 10 CFR 54.21(c)(1)(ii)	4.2.2
Pressure/temperature limits	P-T limit curves managed 10 CFR 54.21(c)(1)(iii)	4.2.3
Low temperature overpressure protection (LTOP)	LTOP limits managed 10CFR54.21(c)(1)(iii)	4.2.4
Pressurized thermal shock	Aging effects managed 10 CFR 54.21(c)(1)(iii)	4.2.5
Metal Fatigue Analyses		
Reactor vessel	Analysis remains valid 10 CFR 54.21(c)(1)(i)	4.3.1.1
Reactor vessel internals	Analysis remains valid 10 CFR 54.21(c)(1)(i)	4.3.1.2
Pressurizer	Analysis remains valid 10 CFR 54.21(c)(1)(i)	4.3.1.3
Pressurizer insurge/outsurge transients	Analysis remains valid 10 CFR 54.21(c)(1)(i)	4.3.1.3
Steam generator	Analysis remains valid 10 CFR 54.21(c)(1)(i)	4.3.1.4
Reactor coolant pump	Analyses remain valid 10 CFR 54.21(c)(1)(i)	4.3.1.5
Control rod drive mechanisms	Analysis remains valid 10 CFR 54.21(c)(1)(i)	4.3.1.6
Regenerative letdown heat exchangers	Analyses remain valid 10 CFR 54.21(c)(1)(i)	4.3.1.7
Class 1 piping and in-line components—ANSI B31.1 piping	Analysis remains valid 10 CFR 54.21(c)(1)(i)	4.3.1.8
Class 1 piping and in-line components—pressurizer surge line	Analysis remains valid 10 CFR 54.21(c)(1)(i)	4.3.1.8

Table 4.1-2
List of IP3 TLAA and Resolution (Continued)

TLAA Description	Resolution Option	Section
Class 1 piping and in-line components —thermowells	Analysis remains valid 10 CFR 54.21(c)(1)(i)	4.3.1.8
Class 1 piping and in-line components —charging system	Analysis will be updated as part of environmental fatigue evaluation. See Section 4.3.3.	4.3.1.8
Non-Class 1 piping and in-line components	Analyses remain valid 10 CFR 54.21(c)(1)(i)	4.3.2
Non-Class 1, non-piping components—residual heat removal heat exchanger	Analyses remain valid 10 CFR 54.21(c)(1)(i)	4.3.2
Effects of reactor water environment on fatigue life	Analyses remain valid 10 CFR 54.21(c)(1)(i) OR Aging effect managed 10 CFR 54.21(c)(1)(iii)	4.3.3
Environmental Qualification Analyses of Electrical Equipment	Aging effect managed 10 CFR 54.21(c)(1)(iii)	4.4
Concrete Containment Tendon Prestress Analyses	IPEC does not have pre-stressed tendons in the containment structures.	4.5
Containment Liner Plate and Penetrations Fatigue Analyses	No TLAA for these components.	4.6
Other TLAA		
Leak before break	Analysis remains valid 10 CFR 54.21(c)(1)(i)	4.7.2
Steam generator flow-induced vibration (tube wear)	Analyses projected 10 CFR 54.21(c)(1)(ii)	4.7.3

4.2 REACTOR VESSEL NEUTRON EMBRITTLEMENT

The regulations governing reactor vessel integrity are in 10 CFR 50. Section 50.60 requires that all light-water reactors meet the fracture toughness, pressure-temperature limits, and material surveillance program requirements for the reactor coolant pressure boundary as set forth in 10 CFR 50 Appendices G and H.

The IPEC current licensing basis analyses evaluating reduction of fracture toughness of the reactor vessel for 40 years are TLAA. The reactor vessel neutron embrittlement TLAA for each unit is summarized below. Forty-eight effective full-power years (EFPY) are projected for the end of the period of extended operation (60 years) based on actual capacity factors from the start of commercial operation until 2005 and an average capacity factor of 95% from 2005 till the end of the period of extended operation.

4.2.1 Reactor Vessel Fluence

The neutron exposure levels for the reactor pressure vessels have been projected for an operating period extending to 48 EFPY. These calculations utilized discrete ordinates $\mathbf{S}_n$ transport analysis to determine the neutron radiation environment within the reactor pressure vessel and surveillance capsules.

Unit 2

In the 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 sixteen reactor operating cycles (1973–2004). The fuel cycle designs analyzed in these calculations have been implemented. Also included in the calculation are analyses for three other cycle designs that were created as a part of the 2003 stretch power uprate study. Therefore, the 48 EFPY projections include the effects of stretch power uprate.

The projected 48 EFPY peak beltline fluence level of 1.906E+19 n/cm^2 (at the 45 degree azimuth position) is used for all beltline materials except axial welds. The beltline axial welds are located at 0, 15 and 30 degree azimuth positions. The maximum projected 48 EFPY peak fluence level for the beltline axial welds is 1.295E+19 n/cm^2 at the 30 degree azimuth position.

The ¼t fluence level is determined by applying Equation (3) of Regulatory Guide 1.99, based on a vessel thickness of 8.625" which yields a fluence of 7.72E+18 n/cm^2 for beltline axial welds and 1.136E+19 n/cm^2 for remaining beltline materials.

Unit 3

In the 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 thirteen reactor operating cycles (1976–2005). The fuel cycle designs analyzed in these calculations have been implemented. Also included in the calculation are analyses for three other cycle designs that were created as a part of the 2003 stretch power uprate study. Therefore, the 48 EFPY projections include the effects of stretch power uprate.

The projected 48 EFPY peak beltline fluence level of 1.560E +19 n/cm^2 (at the 45 degree azimuth position) is used for all beltline materials including axial welds.

The ¼t fluence level is determined by applying Equation (3) of Regulatory Guide 1.99, based on a vessel thickness of 8.625" which yields a fluence of 9.298E+18 n/cm^2.

4.2.2 Charpy Upper-Shelf Energy

The pressure-retaining components of the reactor coolant pressure boundary that are made of ferritic materials must meet the requirements of the ASME Code, supplemented by the additional requirements defined in Appendix G of 10 CFR 50, for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences. For the reactor vessel beltline materials, the values of the reference temperature (RT_{NDT}) and Charpy upper-shelf energy must account for the effects of neutron radiation. The effects of neutron radiation must consider the fluence at the deepest point on the crack front of the flaw assumed in the analysis. Reactor vessel beltline materials must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb.

Regulatory Guide 1.99 provides two methods (positions) for determining Charpy upper-shelf energy (C_VUSE). Position 1 applies for material that does not have surveillance data available and Position 2 applies for material that does have surveillance data. For Position 1, the percent drop in C_VUSE, for a stated copper content and neutron fluence, is determined by reference to Figure 2 of Regulatory Guide 1.99 in accordance with Regulatory Guide 1.99 Section 1.2. This percentage drop is applied to the initial C_VUSE to obtain the adjusted C_VUSE. For Position 2, the percent drop in C_VUSE is determined by plotting the available data on Figure 2 and fitting the data with a line drawn parallel to the existing lines that bound all the plotted points in accordance with Regulatory Guide 1.99 Section 2.2.

Unit 2

The upper shelf energy (USE) values have been determined based on the maximum projected 48 EFPY beltline fluence shown in Section 4.2.1. The beltline region chemistry and surveillance data, including the un-irradiated C_VUSE information, is from the RVID2 database and clarified in WCAP-15629, Revision 1. The projected 48 EFPY peak beltline fluence level at the clad/base metal interface of 1.906E+19 n/cm^2 was applied to all beltline materials except axial welds where the expected peak fluence is 1.295E+19 n/cm^2. The resulting projected 48 EFPY C_VUSE drop and resulting ¼t C_VUSE are shown in Table 4.2-1. One intermediate shell plate (B2002-3) and one lower shell plate (B2003-1) have projected upper shelf energy levels that fall below

50 ft-lb during the period of extended operation. All remaining plate and weld beltline materials exceed 50 ft-lb at 48 EFPY.

10 CFR Part 50, Appendix G, Section IV.A.1 requires licensees to take further corrective actions for cases where the 50 ft-lbs end-of-life USE criterion cannot be met (e.g., when the EOL USE falls below the USE value criterion specified in a previously NRC-approved EMA). As noted in Table 4.2-1, the lowest projected USE level for the IP2 beltline plate material through the period of extended operation is 47.4 ft-lb for intermediate shell plate B2002-3. An equivalent margins analysis performed in WCAP-13587, Rev. 1 demonstrated that the minimum acceptable USE for reactor vessel plate material in 4 loop plants such as IP2 is 43 ft-lbs. In the safety assessment of WCAP-13587, the NRC concluded the report demonstrated margins of safety equivalent to those of the ASME code for beltline plate and forging materials. The IP2 USE values are therefore acceptable since the IP2 lowest projected USE level for the IP2 beltline plate material through the period of extended operation of 47.4 ft-lb for intermediate shell plate B2002-3 is above the 43 ft-lbs minimum acceptable USE for 4 loop plants determined in WCAP-13587 Rev. 1. This determination is consistent with NUREG-1800, Section 4.2.2.1.1.2, and with the NRC Safety Evaluation Report of acceptable USE for H. B Robinson Unit 2 as documented in NUREG-1785. The TLAA for USE is projected through the period of extended operation in accordance with 10CFR54.21(c)(1)(ii).

<u>Unit 3</u>

The IPEC Unit 3 upper shelf energy values have been determined based on the maximum projected 48 EFPY beltline fluence and the beltline region chemistry and surveillance data including the un-irradiated C_VUSE information as summarized in the RVID2 database. The projected 48 EFPY peak beltline fluence level at the clad/base metal interface of 1.560E+19 n/cm^2 was conservatively applied to all beltline materials. The 48 EFPY ¼t fluence level of 9.298E+18 n/cm^2 was calculated in accordance with Regulatory Guide 1.99, Equation (3) based on a vessel thickness of 8.625". The resulting projected 48 EFPY C_VUSE drop and resulting ¼t C_VUSE are displayed in Table 4.2-2. All plate and weld beltline materials exceed 50 ft-lb at 48 EFPY and an equivalent margins analysis is not required.

The TLAA for USE is projected through the period of extended operation in accordance with 10CFR54.21(c)(1)(ii)

Table 4.2-1
IP2 Charpy Upper-Shelf Energy Data for 48 Effective Full-Power Years (EFPY)

Reactor Vessel Location (Beltline Identification)	Material Ident	Material Type	Heat #	Fluence Vessel Clad/BM 48 EFPY	Fluence 1/4T 48 EFPY	%Cu	Un-irradiated USE	%Drop in USE	48 EFPY USE at 1/4T	RG 1.99 Position
Intermediate shell	B2002-1	A302BM	B-4688-2	1.906E+19	1.136E+19	0.190	70	21.1%	55.2	2.2
Intermediate shell	B2002-2	A302BM	B-4701-2	1.906E+19	1.136E+19	0.170	73	22.8%	56.4	2.2
Intermediate shell	B2002-3	A302BM	B-4922-1	1.906E+19	1.136E+19	0.250	74	36.0%	47.4	2.2
Lower shell	B2003-1	A302BM	B-4791-1	1.906E+19	1.136E+19	0.200	71	29.9%	49.8	1.2
Lower shell	B2003-2	A302BM	B-4782-1	1.906E+19	1.136E+19	0.190	88	28.9%	62.6	1.2
Intermediate shell axial welds	2-042 A/B/C	Linde 1092	W5214	1.295E+19	7.72E+18	0.213	121	42.2%	69.9	2.2
Lower shell axial welds	3-042 A/B	Linde 1092	W5214	1.295E+19	7.72E+18	0.213	121	42.2%	69.9	2.2
Intermediate to lower shell circumferential weld	9-042	Linde 1092	34B009	1.906E+19	1.136E+19	0.192	82	34.2%	53.9	1.2

Table 4.2-2
IP3 Charpy Upper-Shelf Energy Data for 48 Effective Full-Power Years (EFPY)

Reactor Vessel Location (Beltline Identification)	Material Ident	Material Type	Heat #	Fluence Vessel Clad/BM 48 EFPY	Fluence 1/4T 48 EFPY	%Cu	Un-irradiated USE	%Drop in USE	48 EFPY USE at 1/4T	RG 1.99 Position
Intermediate shell	B2802-1	A302BM	B-5394-2	1.560E+19	9.298E+18	0.200	102	28.5%	72.9	1.2
Intermediate shell	B2802-2	A302BM	A-0516-2	1.560E+19	9.298E+18	0.220	97	30.5%	67.4	1.2
Intermediate shell	B2802-3	A302BM	B-5391-2	1.560E+19	9.298E+18	0.200	95	28.5%	67.9	1.2
Lower shell	B2803-1	A302BM	A-0495-2	1.560E+19	9.298E+18	0.190	72	27.5%	52.2	1.2
Lower shell	B2803-2	A302BM	C-1397-3	1.560E+19	9.298E+18	0.220	94	30.5%	65.4	1.2
Lower shell	B2803-3	A302BM	A-0512-2	1.560E+19	9.298E+18	0.240	68	21.3%	53.5	2.2
Intermediate shell axial welds	2-042	Linde 1092	34B009	1.560E+19	9.298E+18	0.192	112	32.6%	75.5	1.2
Lower shell axial welds	3-042	Linde 1092	34B009	1.560E+19	9.298E+18	0.192	112	32.6%	75.5	1.2
Intermediate to lower shell circumferential weld	9-042	Linde 1092	13253	1.560E+19	9.298E+18	0.221	111	35.5%	71.6	1.2

4.2.3 Pressure-Temperature Limits

Appendix G of 10 CFR 50 requires operation of the reactor pressure vessel within established pressure-temperature (P-T) limits. These limits are established by calculations that utilize the materials and fluence data obtained through the unit specific reactor surveillance capsule program. Normally, the pressure-temperature limits are calculated for several years into the future and remain valid for an established period of time.

Unit 2

Technical Specifications contain pressure/temperature limits valid through 25 EFPY including the effects of the stretch power uprate.

The P-T limit curves will continue to be updated, as required by Appendix G of 10 CFR Part 50 or as operational needs dictate. This updating will assure that the operational limits remain valid through the period of extended operation. Additional P-T limit analysis is not required at this time. Maintaining the P-T limit curves in accordance with Appendix G of 10 CFR 50 assures that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation consistent with 10 CFR 54.21(c)(1)(iii).

Unit 3

Technical Specifications contain pressure/temperature limits valid through 34 EFPY including the effects of the stretch power uprate. At present, plate B2803-3 with an initial RT_{NDT} of 74°F restricts operation (P-T) in the 150-250°F range. Resolution to the P-T operating window is a current operating term issue and will be resolved three years prior to reaching the RT_{PTS} screening criterion per 10 CFR 50.61 requirements.

The P-T limit curves will continue to be updated, as required by Appendix G of 10 CFR Part 50 or as operational needs dictate. This updating will assure that the operational limits remain valid through the period of extended operation. Additional P-T limit analysis is not required at this time. Maintaining the P-T limit curves in accordance with Appendix G of 10 CFR 50 assures that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation consistent with 10 CFR 54.21(c)(1)(iii).

4.2.4 Low Temperature Overpressure Protection (LTOP) PORV Setpoints

Each time the P-T limit curves are revised, LTOP must be re-evaluated to ensure its functional requirements can be met. Therefore, low temperature overpressure protection limits are considered part of the calculation of pressure/temperature curves. See Section 4.2.3.

4.2.5 Pressurized Thermal Shock

10 CFR 50.61(b)(1) provides rules for protection against pressurized thermal shock events for pressurized water reactors. Licensees are required to perform an assessment of the projected

values of the reference temperature for pressurized thermal shock (RT_{PTS}) whenever a significant change occurs in the parameters affecting RT_{PTS}, such as a change in the expiration date for the operation of the facility.

Section 10 CFR 50.61(b)(2) establishes screening criteria for RT_{PTS} of 270° F for plates, forgings, and axial welds and 300° F for circumferential welds.

Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials" provides two methods for determining RT_{PTS}. Position 1 applies for material that does not have surveillance data available and Position 2 applies for material that does have surveillance data. Adjusted reference temperatures are calculated for both Positions 1 and 2 by following the guidance in Regulatory Guide 1.99, Sections 1.1 and 2.1, respectively, using copper and nickel content of beltline materials and end-of-life (EOL) best estimate fluence projections.

Unit 2

The projected 48 EFPY peak beltline fluence level at the clad/base metal interface of 1.906E+19 n/cm^2 was applied to all beltline materials, except axial welds where the expected peak fluence is 1.295E+19 n/cm^2. The resulting projected 48 EFPY RT_{PTS} are shown in Table 4.2-3. All projected RT_{PTS} values are within the established screening criteria at 48 EFPY. Values of RT_{NDT} for the IP2 beltline materials at ¼ T and ¾ T are summarized in Table 4.2-5.

The TLAA for RTPTS is projected through the period of extended operation in accordance with 10CFR54.21(c)(1)(ii).

Unit 3

The projected 48 EFPY peak beltline fluence level at the clad/base metal interface of 1.560E+19 n/cm^2 was applied to all beltline materials. The resulting projected 48 EFPY RT_{PTS} are shown in Table 4.2-4. All projected RT_{PTS} values are within the established screening criteria for 48 EFPY with the exception of plate B2803-3, which exceeds the screening criterion by 9.9 F. Values of RT_{NDT} for the IP3 beltline materials at ¼ T and ¾ T are summarized in Table 4.2-6.

As required by 10 CFR 50.61(b)(4), a plant-specific safety analysis for plate B2803-3 will be submitted to the NRC three years prior to reaching the RT_{PTS} screening criterion. Alternatively, IP3 may choose to implement the revised PTS (10 CFR 50.61) rule if approved, which would permit use of Regulatory Guide 1.99, Revision 3. Application of Regulatory Guide 1.99, Revision 3 to plate B2803-3 is expected to result in an acceptable through-wall crack frequency at 48 EFPY.

Therefore, the RT_{PTS} TLAA will be adequately managed for the period of extended operation in accordance with 10CFR54.21(c)(1)(iii).

Table 4.2-3
IP2 Pressurized Thermal Shock Data for 48 Effective Full-Power Years (EFPY)

Reactor Vessel Location (Beltline Identification)	Material Ident	Material Type	Heat Number	%Cu	%Ni	Fluence Vessel Clad/BM (10^{19} n/cm^2)	Fluence Factor	Chemistry Factor WCAP-15629 Rev. 1	Un-irradiated RT_{NDT} (°F)	ΔRT_{NDT} (°F)	Margin (°F)	48 EFPY RT_{PTS} (°F)	Method RG 1.99
Intermediate shell	B2002-1	A302BM	B-4688-2	0.190	0.650	1.906	1.176	114.0	34.0	134.1	17.0	185.1	2.1
Intermediate shell	B2002-2	A302BM	B-4701-2	0.170	0.460	1.906	1.176	118.2	21.0	139.1	34.0	194.1	2.1
Intermediate shell	B2002-3	A302BM	B-4922-1	0.250	0.600	1.906	1.176	181.9	21.0	214.0	17.0	252.0	2.1
Lower shell	B2003-1	A302BM	B-4791-1	0.200	0.660	1.906	1.176	152.00	20.0	178.8	34.0	232.8	1.1
Lower shell	B2003-2	A302BM	B-4782-1	0.190	0.480	1.906	1.176	128.80	-20.0	151.5	34.0	165.5	1.1
Intermediate shell axial welds	2-042 A/B/C	Linde 1092	W5214	0.213	1.007	1.295	1.072	254.7	-56	273.0	44.0	261.0	2.1
Lower shell axial welds	3-042 A/B	Linde 1092	W5214	0.213	1.007	1.295	1.072	254.7	-56	273.0	44.0	261.0	2.1
Intermediate to lower shell circumferential weld	9-042	Linde 1092	34B009	0.192	1.007	1.906	1.176	220.9	-56	259.9	65.5	269.4	1.1

Table 4.2-4
IP3 Pressurized Thermal Shock Data for 48 Effective Full-Power Years (EFPY)

Reactor Vessel Location (Beltline Identification)	Material Ident	Material Type	Heat Number	%Cu	%Ni	Fluence Vessel Clad/BM (10^{19} n/cm^2)	Fluence Factor	Chemistry Factor GL 92-01	Un-irradiated RT_{NDT} (°F)	ΔRT_{NDT} (°F)	Margin (°F)	48 EFPY RT_{PTS} (°F)	Method RG 1.99
Intermediate shell	B2802-1	A302BM	B-5394-2	0.200	0.500	1.560	1.123	137.0	5.0	153.8	34.0	192.8	1.1
Intermediate shell	B2802-2	A302BM	A-0516-2	0.220	0.530	1.560	1.123	151.6	-4.0	170.2	34.0	200.2	1.1
Intermediate shell	B2802-3	A302BM	B-5391-2	0.200	0.490	1.560	1.123	135.8	17.0	152.5	34.0	203.5	1.1
Lower shell	B2803-1	A302BM	A-0495-2	0.190	0.470	1.560	1.123	127.7	49.0	143.4	34.0	226.4	1.1
Lower shell	B2803-2	A302BM	C-1397-3	0.220	0.520	1.560	1.123	150.2	-5.0	168.7	34.0	197.7	1.1
Lower shell	B2803-3	A302BM	A-0512-2	0.240	0.520	1.560	1.123	168.2	74.0	188.9	17.0	279.9	2.1
Intermediate shell axial welds	2-042	Linde 1092	34B009	0.192	1.007	1.560	1.123	221.3	-56	248.5	65.5	258.0	1.1
Lower shell axial welds	3-042	Linde 1092	34B009	0.192	1.007	1.560	1.123	221.3	-56	248.5	65.5	258.0	1.1
Intermediate to lower shell circumferential weld	9-042	Linde 1092	13253	0.221	0.732	1.560	1.123	189.1	-54	212.3	56.0	214.3	1.1

Table 4.2-5
IP2 Adjusted Reference Temperature at 48 Effective Full-Power Years (EFPY)

Reactor Vessel Location (Beltline Identification)	Material Ident	Heat Number	Chemistry Factor WCAP-15629 Rev. 1	Un-irradiated RT_{NDT} (° F)	1/4 T Neutron Fluence (10^{19} n/cm^2)	1/4 T Fluence Factor	1/4 T ΔRT_{NDT} (°F)	3/4 T Neutron Fluence (10^{19} n/cm^2)	3/4 T Fluence Factor	3/4 T ΔRT_{NDT} (°F)	48 EFPY 1/4 T RT_{NDT} (°F)	48 EFPY 3/4 T RT_{NDT} (°F)
Intermediate shell	B2002-1	B-4688-2	114.0	34.0	1.136	1.036	118.1	0.404	0.748	85.3	169.1	136.3
Intermediate shell	B2002-2	B-4701-2	118.2	21.0	1.136	1.036	122.4	0.404	0.748	88.5	177.4	143.5
Intermediate shell	B2002-3	B-4922-1	181.9	21.0	1.136	1.036	188.4	0.404	0.748	136.1	226.4	174.1
Lower shell	B2003-1	B-4791-1	152.00	20.0	1.136	1.036	157.4	0.404	0.748	113.8	211.4	167.8
Lower shell	B2003-2	B-4782-1	128.80	-20.0	1.136	1.036	133.4	0.404	0.748	96.4	147.4	110.4
Intermediate shell axial welds	2-042 A/B/C	W5214	254.7	-56	0.772	0.927	236.2	0.274	0.647	164.9	224.2	152.9
Lower shell axial welds	3-042 A/B	W5214	254.7	-56	0.772	0.927	236.2	0.274	0.647	164.9	224.2	152.9
Intermediate to lower shell circumferential weld	9-042	34B009	220.9	-56	1.136	1.036	228.8	0.404	0.748	165.3	238.3	174.8

Table 4.2-6
IP3 Adjusted Reference Temperature at 48 Effective Full-Power Years (EFPY)

Reactor Vessel Location (Beltline Identification)	Material Ident	Heat Number	Chemistry Factor RVID2	Un-irradiated RT_{NDT} (° F)	1/4 T Neutron Fluence (10^{19} n/cm^2)	1/4 T Fluence Factor	1/4 T ΔRT_{NDT} (° F)	3/4 T Neutron Fluence (10^{19} n/cm^2)	3/4 T Fluence Factor	3/4 T ΔRT_{NDT} (° F)	48 EFPY 1/4 T RT_{NDT} (° F)	48 EFPY 3/4 T RT_{NDT} (° F)
Intermediate shell	B2802-1	B-5394-2	137.0	5.0	0.930	0.980	134.2	0.330	0.695	95.2	173.2	134.2
Intermediate shell	B2802-2	A-0516-2	151.6	-4.0	0.930	0.980	148.5	0.330	0.695	105.4	178.5	135.4
Intermediate shell	B2802-3	B-5391-2	135.8	17.0	0.930	0.980	133.0	0.330	0.695	94.4	184.0	145.4
Lower shell	B2803-1	A-0495-2	127.7	49.0	0.930	0.980	125.1	0.330	0.695	88.8	208.1	171.8
Lower shell	B2803-2	C-1397-3	150.2	-5.0	0.930	0.980	147.1	0.330	0.695	104.4	176.1	133.4
Lower shell	B2803-3	A-0512-2	168.2	74.0	0.930	0.980	164.8	0.330	0.695	116.9	255.8	207.9
Intermediate shell axial welds	2-042	34B009	221.3	-56	0.930	0.980	216.7	0.330	0.695	153.8	226.2	163.3
Lower shell axial welds	3-042	34B009	221.3	-56	0.930	0.980	216.7	0.330	0.695	153.8	226.2	163.3
Intermediate to lower shell circumferential weld	9-042	13253	189.1	-54	0.930	0.980	185.2	0.330	0.695	131.4	187.2	133.4

4.2.6 References

4.2-1 Improved Technical Specifications, Appendix A to Facility Operating License No. DPR-64 (Indian Point 3), Amendment 231.

4.2-2 NL-04-069, Letter from F. Dacimo to NRC, Proposed Changes to Technical Specifications: Stretch Power Uprate (4.85%) and Adoption of TSTF-339, June 3, 2004.

4.2-3 NL-05-020, Letter from F. Dacimo to NRC, Reply to RAI Regarding Indian Point 3 License Amendment Requests for Stretch Power Uprate, February 11, 2005.

4.2-4 NL-02-006, Letter from F. Dacimo to NRC, Response to Request for Additional Information Indian Point 2 License Amendment Request for Reactor Coolant System Heatup and Cooldown Limitation Curves (TAC No.: MB2419), January 11, 2002.

4.2-5 US NRC, Reactor Vessel Integrity Database (RVID), Version 2.0.1, July 2000.

4.3 METAL FATIGUE

Fatigue analyses are potential TLAA for Class 1 and selected non-Class 1 mechanical components. Fatigue is an age-related degradation mechanism caused by cyclic stressing of a component by either mechanical or thermal stresses. Fatigue analyses are TLAA if they meet all six elements of the definition in 10 CFR 54.3(a). If the analyses are based on a number of cycles estimated for the current license term, they may be considered to meet criteria 54.3(a)(3) of being based on the current operating term. Evaluation of the TLAA, per 10 CFR 54.21 (c)(1), determines whether:

(i) The analyses remain valid for the period of extended operation,

(ii) The analyses have been projected to the end of the period of extend operation, or

(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

The aging management reviews conducted as part of the integrated plant assessment (IPA) (summarized in Section 3) identified all components that are susceptible to fatigue damage. If a component has a fatigue TLAA that remains valid (10 CFR 54.21 (c)(1)(i)) or is projected to cover the period of extended operation (10 CFR 54.21 (c)(1)(ii)), then cracking due to fatigue is not an aging effect requiring management for that component. If the TLAA does not remain valid for the period of extended operation, then cracking due to fatigue is an aging effect requiring management for the analyzed component. Cracking due to fatigue can be managed by a variety of plant programs in accordance with 10 CFR 54.21(c)(1)(iii).

Fracture mechanics analyses of flaws discovered during in-service inspection may be TLAA for those analyses based on time-limited assumptions defined by the current operating term. When a flaw is detected during in-service inspections, either the flaw must be repaired or the component that contains the flaw can be evaluated for continued service in accordance with ASME Section XI. These evaluations may show that the component is acceptable to the end of the license term based on projected in-service flaw growth. Flaw growth is typically predicted based on the design thermal and mechanical loading cycles.

4.3.1 Class 1 Fatigue

Components designed in accordance with ASME Section III have fatigue analyses. Current design basis fatigue evaluations calculate cumulative usage factors (CUFs) for components or sub-components based on design transient cycles. The design transients are listed in Table 4.3-1 for IP2 and Table 4.3-2 for IP3. The resulting CUFs are listed in their respective subcomponent section, starting with the reactor vessel in Section 4.3.1.1.

The current design basis fatigue evaluations do not consider the effects of reactor water environment on fatigue life. This is consistent with SECY-95-245, in which the NRC indicated that no immediate staff or licensee action is necessary to deal with the environmentally assisted fatigue issue prior to the period of extended operation for license renewal.

The numbers of cycles accrued to date have been projected to determine the numbers of cycles expected at the end of 60 years of operation. Tables 4.3-1 and 4.3-2 also show the projected values for the period of extended operation. With the limited exceptions discussed below, the projected numbers of cycles for 60 years of operation do not exceed the analyzed numbers of cycles.

The Fatigue Monitoring Program tracks and evaluates the design transients and requires corrective actions if the numbers of analyzed transients are approached. The Fatigue Monitoring Program ensures that the numbers of transient cycles experienced by the plant remain within the analyzed numbers of cycles, and hence the component CUFs remain below the values calculated in the design basis fatigue evaluations. Further details on the Fatigue Monitoring Program are provided in Appendix B.

Unit 2

The cycle counts are divided into normal conditions, test conditions, abnormal (upset) conditions, pressurizer spray actuations, and other events. A rate per day was calculated for each event and that rate was multiplied by the days remaining to the end of the period of extended operation to project the cycles. The rates for most transients were based on the cycles accrued to date and the time from initial operation.

Some transients, such as reactor trips, were projected based on more recent operating history, 1999 to 2005. This is because plant operating practices have changed and some of the transients occur more or less often now than they did early in plant life. There were substantially more reactor trips in the early years of operation at IPEC, and the rate of reactor trips experienced in the last six years is more representative of the rate of trips expected through the remainder of plant life.

The 60-year projections for IP2 show the following.

> The only normal condition projecting above the analyzed number of cycles is steady state fluctuations. The projection is 1.5×10^6 while the analyzed number is 1×10^6. However, the value shown in Table 4.3-1 is not based on actual cycles. The value shown in Table 4.3-1 for cycles as of 10/31/1999 is a calculated value based on the assumption that the transients occur at a constant rate that results in a number of transients over 40 years of operation equal to the analyzed number of transients. Hence the projection to 60 years based on this calculated value is 1.5 times the analyzed number of transients. In accordance with the Fatigue Monitoring Program, prior to the period of extended operation, corrective actions will be taken to confirm that monitoring is not required or to establish appropriate monitoring.

Feedwater cycling, a replacement steam generator design transient limited to 18,300 cycles, does not appear on Table 4.3-1. The value of 18,300 is the projected value for 40 years of steam generator operation. Since the IP2 replacement steam generators will not be in service for 40 years at the end of the period of extended operation, feedwater cycling is not expected to exceed the analyzed number of cycles.

The only abnormal condition projected to exceed its monitored limit is loss of power. Refer to the Fatigue Monitoring Program, Section B.1.12, for enhancements related to "loss of power" cycling.

Several of the "Other Events" will exceed their analyzed numbers prior to the end of the period of extended operation. These transients apply to the charging system piping, which is evaluated as described in Section 4.3.3.

As indicated above, for certain events that affect fatigue usage, linear projections of the actual data to the end of the period of extended operation will exceed the analyzed number of design basis transients. However, because of the conservative nature of the CUF estimates, implicit margin exists. For those locations where additional fatigue analysis is required to take advantage of the implicit margin, actions will be taken in accordance with the Fatigue Monitoring Program prior to exceeding the analyzed numbers of transients.

IP2 will continue to monitor analyzed cycles under the Fatigue Monitoring Program. Enhancements to the Fatigue Monitoring Program discussed in Appendix B will address the 60-year projections discussed above.

Unit 3

Transients associated with the reactor vessel, safety injection actuations, and residual heat removal cycles are tracked. A rate per day was calculated for each transient and that rate was multiplied by the days remaining to 60 years to project the number of future cycles. The rates were based on the cycles accrued to date and the time from initial operation.

The numbers of plant heatups and cooldowns were taken from the IP3 Shutdown History and Shutdown Summary which contains the shutdown count through 1995. The rate from 1973 to 1995 was used to project shutdowns and startups, and this should be a conservative projection as improved operations have resulted in less frequent shutdowns/startups in recent years.

The 60-year projections for IP3 show that no transient will exceed the number of analyzed cycles prior to the end of the period of extended operation.

The Fatigue Monitoring Program will assure that the analyzed numbers of transients are not exceeded during the period of extended operation. Enhancements to the Fatigue Monitoring Program discussed in Appendix B will add additional transients to the Unit 3 list of transients monitored, similar to the Unit 2 list.

Table 4.3-1
IP2 Analyzed and Projected Number of Thermal Cycles

Transient Condition	Analyzed Numbers of Cycles	Cycles as of 5/24/2005	60-year Projection 9/28/2033[1]
Normal Conditions			
Plant heatup at 100°F per hr	200	103	196
Plant cooldown at 100°F per hr	200	103	196
Refueling	80	16	31
Plant loading at 5 percent of full power per min	14,500	1,500	2,844
Plant unloading at 5 percent of full power per min	14,500	1,092	2,071
Step load increase of 10 percent of full power (but not to exceed full power)	2,000	44	84
Step load decrease of 10 percent of full power	2,000	315	598
Step load decrease of 50 percent of full power	150	36	69
Boron concentration equalization	36,600	3,004	5,695
Feedwater cycling	2,000	416	789
Loop out of service	80	4	8
Reactor coolant pump start/stop	10,000	2018	3,826
Steady state fluctuations	1,000,000	781,209	1,480,919
RCS depressurization from 2250 psig to 2000 psig	50	2	4
Test Conditions			
Turbine roll test	20	1	2
Hydrostatic test at 3110 psig	1	1	1[2]
Hydrostatic test at 2485 psig, 400°F	50	43	43[2]
Primary to secondary hydrotest, 1356 psi	5	1	2

Table 4.3-1
IP2 Analyzed and Projected Number of Thermal Cycles (Continued)

Transient Condition		Analyzed Numbers of Cycles	Cycles as of 5/24/2005	60-year Projection 9/28/2033[1]
Secondary Hydro	SG-21	10	1	2
	SG-22	10	1	2
	SG-23	10	1	2
	SG-24	10	1	2
Secondary Pressure Test	SG-21	120	1	2
(Primary at 0 psig,	SG-22	120	1	2
Secondary at 1085 psig)	SG-23	120	1	2
	SG-24	120	1	2
Abnormal Conditions				
Reactor trip		400	239	292[3]
- No excessive cooldown		230	88	124
- Excessive cooldown		160	148	159
- Excessive cooldown with safety injection		10	3	9
Loss of load, without immediate turbine trip or reactor trip		80	31	59
Loss of power		10	6	12
Control rod drop (with reactor trip)		80	2	4
Loss of secondary pressure		6	0	0
Partial loss of flow, one pump only		80	13	25
Excessive feedwater flow		30	0	0
Inadvertent safety injection actuation		60	0	0
Inadvertent startup of an RCP		10	0	0
Inadvertent RCS depressurization		10	0	0
Abnormal condition		270	44	84

Table 4.3-1
IP2 Analyzed and Projected Number of Thermal Cycles (Continued)

Transient Condition	Analyzed Numbers of Cycles	Cycles as of 5/24/2005	60-year Projection 9/28/2033[1]
Turbine roll test	20	1	2
Other Events			
Pressurizer safety valve cycles	40	0	0
Power operated relief valve cycles	100	4	8
Charging and letdown flow shutoff and return to service	171	129	267
Letdown flow shutoff with prompt return to service	507	348	583
Letdown flow shutoff with delayed return to service	3	2	2
Charging flow shutoff with prompt return to service	101	60	60
Charging flow shutoff with delayed return to service	0	0	0
Charging flow step decrease and return to normal	4118	2467	2543
Charging flow step increase and return to normal	4116	2466	2542
Letdown flow step decrease and return to normal	1642	1028	1283
Letdown flow step increase and return to normal	8688	6927	15860

1. Projection is the number of cycles as of 5/24/2005 plus the rate per day times the number of days from 5/24/2005 to the end of the period of extended operation.
2. Hydro tests are no longer required or performed. Therefore hydro tests are projected to remain at the current value for the remainder of plant life. Section 3.0 of WCAP-16169 states the vessel is currently analyzed for 200 hydrotests.
3. Total reactor trips were projected by summing the three sub-categories of trips below this entry, not by projecting the totals. This gives a conservative result due to the round up on each of the three parts.

Table 4.3-2
IP3 Analyzed and Projected Number of Thermal Cycles

	Transient Condition	Analyzed Numbers of Cycles	Cycles as of 3/31/2006	60-year Projection[1] 12/12/2035
1	Plant heatup at 100°F per hr	200	Note 2	120[2]
2	Plant cooldown at 100°F per hr	200	Note 2	120[2]
3	Plant loading at 5 percent of full power per minute	14500	24	48
4	Plant unloading at 5 percent of full power per minute	14500	24	48
5	Step load increase of 10 percent of full power (but not to exceed full power)	2000	53	105
6	Step load decrease of 10 percent of full power	2000	53	105
7	Step load decrease of 50 percent of full power	200	26	52
8	Reactor trip	400	83	165
9	Hydrostatic test at 3110 psig pressure	5	NA[3]	NA[3]
10	Hydrostatic test at 2485 psig pressure and 400°F temperature	200	NA[3]	NA[3]
11	Steady state fluctuations	Infinite	NA[4]	NA[4]
12	Loss of load, without immediate turbine trip or reactor trip	80	NA	0
13	Partial loss of flow, one pump only	80	5	10
14	Operating basis earthquake (OBE)	5	0	0
15	Design basis earthquake (DBE)	1	0	0
	SI actuations	40	11	22
	RHR cycles	200	55	109

1. Cycle projection based on rate of occurrence of cycles between 1975 and 2006, unless otherwise indicated. Projection is the number of cycles as of 3/31/2006 plus the rate per day times the number of days from 3/31/2006 to the end of the period of extended operation, unless otherwise indicated.
2. Cycle projection based on rate of occurrence of cycles between 1975 and 1995. Projection is the number of cycles as of 12/31/1995 plus the rate per day times the number of days from 12/31/1995 to the end of the period of extended operation.
3. Hydro tests are no longer required or performed. Current values are zero and projections are zero.
4. As an infinite number of steady state fluctuations are allowed, these fluctuations are not counted.

4.3.1.1 Reactor Vessel

The reactor pressure vessel (and appurtenances) fatigue analyses were performed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition, 1966 and 1967 addenda. (A complete listing of applicable codes is given in Tables 4.1-9 of the IP2 and IP3 UFSARs.) The existing fatigue analyses of the reactor vessel are considered TLAA because they are based on numbers of cycles expected in 40 years of operation. The CUFs for the reactor pressure vessel are given in Table 4.3-3 for IP2 and Table 4.3-4 for IP3.

Design cyclic loadings and thermal conditions for the reactor pressure vessel were originally defined in the design specifications and analyzed in the original vessel stress reports. These analyses have been occasionally revised, most recently for the extended power uprate. These latest analyses are reflected in the current UFSAR tables. As described in Section 4.3.1, the projected numbers of transient cycles used for reactor vessel fatigue analyses remain within analyzed values. Consequently, the TLAA (reactor vessel fatigue analyses) based on those transients will remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i) for both IP2 and IP3.

Table 4.3-3
Current Cumulative Usage Factors for the IP2 Reactor Vessel

Location	UFSAR Table 4.3-2
Control rod housing	0.01
Head flange	0.0107
Vessel flange	0.0229
Closure studs	0.944[1]
Primary nozzles - inlet	0.050
Primary nozzles – outlet	0.281
Core support pad (lateral)	0.904
Bottom head to shell	0.004
Bottom instrument penetrations	0.201
Nozzle belt to shell	0.0029
Head (CRDM) adapter plugs	0.0036

1. The CUF of the reactor vessel studs was revised based on the optimization of the stud tensioning procedures and a UFSAR change is in process to reflect this revision. These CUFs were determined from the previous CUFs based on the change in peak stresses seen during the revised tensioning procedure; hence, these CUFs are also based on the design number of cycles.

Table 4.3-4
Current Cumulative Usage Factors for the IP3 Reactor Vessel

Location	UFSAR Table 4.3-2
Control rod housing	0.124
Head flange	0.024
Vessel flange	0.023
Closure studs	0.944[1]
Primary nozzles - inlet	0.049
Primary nozzles – outlet	0.259
Core support pad (lateral)	0.052
Bottom head to shell	0.02
Bottom instrument penetrations	0.206
Nozzle belt to shell	0.002
Head (CRDM) adapter plugs	0.0036

1. The CUF of the reactor vessel studs was revised based on the optimization of the stud tensioning procedures and a UFSAR change is in process to reflect this revision. These CUFs were determined from the previous CUFs based on the change in peak stresses seen during the revised tensioning procedure; hence, these CUFs are also based on the design number of cycles.

4.3.1.2 Reactor Vessel Internals

The IPEC reactor vessel internals were designed to meet the intent of Subsection NG of the ASME Boiler and Pressure Vessel Code, Section III. A plant-specific stress report on the reactor internals was not required. The structural integrity of the reactor internals design has been ensured by analyses performed on both generic and plant-specific bases. These analyses were used as the basis for evaluating critical reactor internal components with CUFs provided in Tables 4.3-5 and 4.3-6.

Table 4.3-5
CUFs for the IP2 Reactor Vessel Internals

Location	CUF
Upper support plate assembly	0.173
Upper support plate flange	0.065
Upper core plate	0.026
Mid core barrel	0.0197
Upper core barrel	0.0193
Core barrel nozzle	0.389
Core barrel flange	0.0193
Lower radial key plate	0.001
Lower radial key 45° plane	0.144
Lower core plate	0.420
Lower core support plate	0.521
Lower support columns	0.245

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Table 4.3-6
CUFs for the IP3 Reactor Vessel Internals

Location	CUF
Upper support plate assembly	0.81
Upper core plate	0.062
Core barrel to LSP junction	0.51
Thermal shield	0.348
Lower core plate	0.237
Instrumentation columns	0.22
Lower support columns	0.49

These CUFs are TLAA as they are based on numbers of cycles that were expected during 40 years of operation. Although not required by Code, these analyses were used to justify these components for service. The CUFs were based on the same transients as the reactor vessel, and those transients will not be exceeded in 60 years, therefore these TLAA remain valid for the period of extended operation per 10 CFR 54.21(c)(1)(i).

4.3.1.3 Pressurizer

In the original stress report the pressurizer shells were not analyzed for fatigue usage factors as they met the requirements of the ASME code, Section N-415.1, "Vessels Not Requiring Analysis for Cyclic Operation." The stress report contains a bounding analysis for the 1800 cubic foot pressurizer. This analysis bounds both IP2 and IP3 as well as other Westinghouse pressurizer designs. The design transients used in evaluating Sections N-415.1(a) through (f) of ASME Section III are given in the stress report. Each of eleven transients identified in the stress report was analyzed for a number of occurrences at or above the IPEC analyzed numbers given in Table 4.3-1 and Table 4.3-2.

Section 4.3.1 projected the numbers of cycles of the all transients used in the pressurizer fatigue determination, except steady state oscillations, would remain below the numbers analyzed by the stress report through the period of extended operation. The stress report analyzed the 106 steady state oscillations only for condition N-415.1(b), where these oscillations were determined to be "Not Significant." The projection of steady state oscillations therefore does not affect the results of the stress report evaluation of N-415.1.

Therefore the number of significant cycles will remain below that analyzed by the stress report. Thus the TLAA for determining that detailed fatigue analyses are not required remains valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

The original stress report did analyze the surge nozzle and spray nozzle.

The IPEC pressurizers were evaluated for the stretch power uprates and cumulative usage factors were updated. Usage factors are given in Tables 4.3-7 and 4.3-8.

Table 4.3-7
Cumulative Usage Factors for the IP2 Pressurizer

Location	CUF
Safety and relief nozzle	0.2047
Spray nozzle	0.996
Upper shell	0.4161
Surge nozzle	0.264

Table 4.3-8
Cumulative Usage Factors for the IP3 Pressurizer

Location	CUF
Safety and relief nozzle	0.1981
Spray nozzle	0.974
Upper shell	0.4161
Surge nozzle	0.9612

None of the design transients used in the analysis of the pressurizer will be exceeded as discussed in Section 4.3.1. The pressurizer fatigue analyses will thus remain valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

Insurge/Outsurge Transients

The impact of pressurizer insurge/outsurge transients was not considered in original design basis calculations for the pressurizer. The IP2 CUF of record for the pressurizer surge nozzle remains the original design stress report number of 0.264. IP3 re-evaluated the CUF of the pressurizer surge line nozzle considering insurge/outsurge during the 200 design heatups and cooldowns. The revised CUF for IP3 is 0.9612. The CUFs are reflected in Tables 4.3-7 and 4.3-8. As the cycles on which these analyses are based will not be exceeded through the period of extended operation, these TLAA remain valid through the period of extended operation per 10CFR54.21(c)(1)(i). Nonetheless, as the surge nozzles require environmental fatigue considerations, they will be reanalyzed for license renewal as discussed in Section 4.3.3.

4.3.1.4 Steam Generators

Summary Description

IP2 replaced steam generators during an outage completed in January 2001. IP3 replaced steam generators during an outage completed in June 1989. The IPEC replacement steam generators were analyzed for fatigue in their component stress reports. The replacement steam generators were re-evaluated with respect to fatigue for the power increase.

Cumulative usage factors for critical components, shown in Table 4.3-9 and 4.3-10, are from the power uprate analyses. Usage factors for additional, non-critical, components are available in the stress reports. The usage factor calculations are considered TLAA.

Table 4.3-9
Cumulative Usage Factors for the IP2 Steam Generators

Location	CUF
Primary Side	
Divider plate	0.683
Tubesheet/shell junction	0.451
Tube/tubesheet weld	0.809
Tubes	0.484
Secondary Side	
Main feedwater nozzle	0.898
Secondary manway stud[1]	0.438
Steam nozzle	0.212
Steam nozzle support ring	0.220
Steam nozzle insert	0.212

1. The IP2 replacement steam generators use studs and nuts. There are no longer any secondary manway bolts in use at IPEC.

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Table 4.3-10
Cumulative Usage Factors for the IP3 Steam Generators

Location	CUF
Primary Side	
Divider plate	0.789
Tubesheet/shell junction	0.416
Tube/tubesheet weld	0.082
Tubes	0.161
Secondary Side	
Main feedwater nozzle	1.00
Secondary manway stud[1]	0.920
Steam nozzle	0.023
Steam nozzle support ring	0.894
Steam nozzle insert	0.208

1. The IP3 replacement steam generators use studs and nuts. There are no longer any secondary manway bolts in use at IPEC.

Evaluation

Section 4.3.1 projects that none of the design transients used for steam generator fatigue analysis will exceed their analyzed numbers during the period of extended operation. These usage factor calculations are based on the design transients discussed in Section 4.3.1 and will remain valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

4.3.1.5 Reactor Coolant Pump Fatigue Analysis

The reactor coolant pumps were evaluated with respect to fatigue for the stretch power uprate. Stresses in the reactor coolant pumps were reviewed and shown to remain within the ASME Code allowable stresses. These stress calculations have no time dependent assumptions and therefore are not TLAA.

Detailed fatigue analyses of RCP casings were not required because the conditions specified in the 1965 edition of the ASME code Sections N-415.1(a) through (f), “Vessels Not Requiring Analysis for Cyclic Operation,” were met. These fatigue waiver evaluations may be considered TLAA if they used the numbers of design cycles in the evaluation of items N-415.1(a) through (f). IPEC has chosen to conservatively call the evaluations TLAA. These determinations were based on the numbers of design cycles. The projections in Tables 4.3-1 and 4.3-2 show that the numbers of significant cycles in 60 years will remain below the numbers of cycles used in these determinations. Thus the TLAAs for determining that detailed fatigue analyses are not required remain valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

Unit 2

From stretch power uprate analyses, the CUF for the RCP main flange bolts is 0.44. As this CUF is based on the design transients, and the design transients will not be exceeded, the calculation of CUF for the main flange bolts remains valid for the period of extended operation in accordance with 10CFR54.21(1)(c)(i).

Unit 3

From stretch power uprate analyses, the CUF for the RCP main flange bolts is 0.32. As this CUF is based on the design transients, and the design transients will not be exceeded, the calculation of CUF for the main flange bolts remains valid for the period of extended operation in accordance with 10CFR54.21(1)(c)(i).

4.3.1.6 Control Rod Drive Mechanisms

The IPEC control rod drive mechanisms were originally analyzed for fatigue in the generic component stress report. The IPEC control rod drive mechanisms were evaluated with respect to fatigue for the power uprate and cumulative usage factors, provided in Tables 4.3-11 and 4.3-12, were updated. These usage factor calculations are considered TLAA as they are based on design transients intended to allow at least 40 years of operation.

Table 4.3-11
Cumulative Usage Factors for the IP2 CRDMs

Location	CUF
Upper joint canopy	0.858
Upper joint weld canopy	0.5045
Upper joint threaded area	0.36025
Middle joint weld canopy	0.5235
Lower joint weld canopy	0.02422

Table 4.3-12
Cumulative Usage Factors for the IP3 CRDMs

Location	CUF
Upper joint canopy	0.763
Middle joint weld canopy	0.425
Lower joint weld canopy	0.003
Capped latch housing	0.093

As discussed in Section 4.3.1, the numbers of analyzed design transients used in this fatigue analysis will not be exceeded in 60 years of operation and thus this TLAA will remain valid through the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

4.3.1.7 Class-1 Heat Exchangers

The original manufacturing equipment specification for the regenerative letdown heat exchangers and the excess letdown heat exchangers says these heat exchangers are to be qualified for various transients. The E-spec suggests that the manufacturer should verify in writing that all conditions of Paragraph N-415.1 of Section III are satisfied for the transient conditions; otherwise, a fatigue analysis is required. The IPEC UFSARs say the regenerative letdown heat exchangers and the excess letdown heat exchangers are qualified to 2000 temperature cycles.

Westinghouse determined that the regenerative heat exchanger was the controlling heat exchanger with regards to fatigue, and therefore only that heat exchanger was analyzed. The associated report concludes that by 10/31/1999, Unit 2 had accumulated 466 of the analyzed 2000 cycles (23.3%) on the regenerative heat exchanger. Further, since the analyzed CUF was only 0.235, the CUF as of 10/31/1999 was equal to 0.235 x 23.3% = 0.05. For license renewal, the thermal cycles seen by the regenerative heat exchanger can be projected through the period of extended operation to show that only 1072 cycles (54%) are expected in 60 years, corresponding to a projected CUF of 0.235 x 54% = 0.13. The IP3 auxiliary heat exchangers have no plant-specific evaluation. However, the similarity in design and operation between the two units indicates the results would be similar. As the projected IP2 CUF is 0.13, it follows that the IP3 CUF would also be well below the limit of 1.0, such that a plant-specific analysis is not required. Thus the TLAA for the heat exchanger fatigue remains valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

IPEC design documents indicate that the auxiliary heat exchangers are not the limiting components in the CVCS system. The charging nozzles are more limiting. Therefore, monitoring of the charging nozzles will assure acceptability of the auxiliary heat exchangers.

Because the charging nozzle is one of the locations identified by NUREG-6260 as requiring environmental adjustments to the fatigue analysis, this nozzle will be evaluated with the other NUREG-6260 locations as discussed in Section 4.3.3.

4.3.1.8 Class 1 Piping and Components

ANSI B31.1 Piping

The IPEC Class 1 boundary corresponds to all reactor coolant system (RCS) pressure boundary components within the ASME Section XI, IWB inspection boundary.

USAS B31.1 was used in the design of the primary coolant piping. A thermal expansion flexibility stress analysis was performed on the main primary coolant piping in accordance with the criteria set forth in USAS B31.1 to ensure that the stress range is within the prescribed limits. As per the requirements of USAS B31.1, no fatigue analysis is required and no fatigue analysis of the reactor coolant loop piping is performed. Rather stress range reduction factors are used to account for anticipated transients (normally, a stress range reduction factor of 1.0 is acceptable in the stress analyses for up to 7000 cycles).

IPEC evaluated the projected thermal cycles for 60 years of plant operation at IP2. For IP2 Class 1 piping, thermal cycling is limited to the transients identified in Table 4.3-1 and through the period of extended operation will remain well below the 7000 cycles allowed by B31.1 for a stress range reduction factor of 1.0.

IPEC evaluated the projected thermal cycles for 60 years of plant operation at IP3. For IP3 Class 1 piping, thermal cycling is limited to the transients identified in Table 4.3-2. Counting every expected transient, the total cycles in 60 years of operation is well below the 7000 cycles allowed by B31.1 for a stress range reduction factor of 1.0.

The results of this evaluation indicate that 7000 thermal cycles will not be exceeded for 60 years of operation. Therefore, the Class 1 pipe stress calculations are valid for the period of extended operation. Thus, the TLAA remains valid for the extended period of operation in accordance with 10 CFR 54.21(c)(1)(i).

Pressurizer Surge Line Piping

NRC Bulletin 88-11 addresses potential thermal stresses associated with thermal stratification experienced by the pressurizer surge line. Thermal stratification in the pressurizer surge line can cause unexpected piping movement and potential plastic deformation. The original design analyses for IP2 and IP3 did not consider thermal stratification of the surge line. Per NRC Bulletin 88-11, utilities were required to establish and implement a program to confirm pressurizer surge line integrity in view of the occurrence of thermal stratification as a result of the plant heatup/cooldown cycles. IPEC participated in a program to assess the impact of pressurizer insurge/outsurge transients on the structural integrity of the pressurizer. Operating procedures were modified to decrease the severity of transients resulting from pressurizer surges during heatup and cooldown. New pressurizer lower head transients were developed based on the modified operating procedures and usage factors of the limiting pressurizer items in the lower head were reevaluated and shown to be less than 1.0 for 40 years. These analyses were performed as a Westinghouse Owners Group task, supplemented by additional unit-specific inspections and activities. The NRC review of the Westinghouse Owners Group task concluded that plant-specific analyses would be required for 28 Westinghouse plants, including IP2 and IP3. Plant-specific analyses for IP2/IP3 were performed in 1991 and the NRC reviewed the analyses and concluded that IPEC had addressed the actions required by Bulletin 88-11. The maximum CUF for the IP2 and IP3 surge line piping occurred at the pipe side of the pressurizer nozzle safe end with a value of 0.60.

The site-specific evaluations of the pressurizer surge line are considered TLAA since the evaluations use time-limited assumptions such as thermal and pressure transients, and operating cycles. The dominant cycles in the surge line analysis are the 200 heatups and cooldowns, including the stratification and striping associated with those transients. As discussed in Section 4.3.1, the number of analyzed heatups/cooldowns, as well as the other design transients presented in Tables 4.3-1 and 4.3-2, will not be exceeded in 60 years of operation. Thus this TLAA remains valid through the end of the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

<u>Thermowells</u>

Westinghouse identified cumulative usage factors for various thermowells associated with the IPEC pressurizers based on 200 heatups and cooldowns with a maximum CUF of 0.021. Since Table 4.3-1 and Table 4.3-2 project that 200 heatups and cooldowns will not be exceeded, this TLAA remains valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

<u>Charging System Piping</u>

<u>Unit 2</u>

The IP2 charging system piping fatigue analysis determined the limiting CUF for the charging nozzle as 0.99 for the numbers of analyzed transients shown in the last nine entries in Table 4.3-1. As shown in Table 4.3-1, three of these nine transients are expected to exceed their analyzed number by the end of 60 years of operation. Because the charging nozzle is one of the locations identified by NUREG-6260 as requiring environmental adjustments to the fatigue analysis, this nozzle will be evaluated with the other NUREG-6260 locations as discussed in Section 4.3.3.

<u>Unit 3</u>

No specific analysis of the IP3 charging system piping was identified. However, because the charging nozzle is one of the locations identified by NUREG-6260 as requiring environmental adjustments to the fatigue analysis, this nozzle will be evaluated with the other NUREG-6260 locations as discussed in Section 4.3.3 below.

IP2 Loop 3 Accumulator Nozzle

The IP2 loop 3 accumulator nozzle does not have a thermal sleeve. Although this piping was built to B31.1 and no fatigue analysis of the piping was originally performed, a fatigue analysis was performed to justify continued operation without the thermal sleeve. An analysis of the nozzle determined the CUF to be 0.95. This analysis was based on the same design cycles as the reactor vessel, and those analyzed numbers of cycles will not be exceeded for 60 years of operation. Therefore, this TLAA for the IP2 loop 3 accumulator nozzle remains valid for the period of extended operation per 10CFR54.21(c)(1)(i).

4.3.2 Non-Class 1 Fatigue

The design of ASME III Code Class 2 and 3 piping systems incorporates the Code stress reduction factor for determining acceptability of piping design with respect to thermal stresses. In general, 7000 thermal cycles are assumed, allowing a stress reduction factor of 1.0 in the stress analyses. IPEC evaluated the validity of this assumption for 60 years of plant operation. The results of this evaluation indicate that the 7000 thermal cycle assumption is valid and bounding for 60 years of operation. Therefore, the pipe stress calculations are valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Review of potential TLAAs for IPEC non-Class 1 components identified a fatigue analysis only for the residual heat removal (RHR) heat exchanger.

Residual Heat Removal Heat Exchanger

The original manufacturing equipment specification states the RHR heat exchanger is to be qualified for 200 cycles that would occur during plant shutdowns. The IP2 UFSAR, Table 6.2-8 and the IP3 UFSAR, Table 6.2-6 state the RHR heat exchangers are qualified to 200 cycles from 85 °F to 350 °F.

No fatigue analyses for these heat exchangers have been identified. It is believed that the manufacturers showed the requirements of Paragraph N-415.1 of ASME Section III were met; but no written statement from the manufacturer has been found. Nonetheless, IPEC is conservatively considering that determination a TLAA. This TLAA is considered based on the specified 200 design cycles, corresponding to the 200 design heatups/cooldowns for the reactor coolant system. The system will not exceed 200 heatups and cooldowns in 60 years as projected in Tables 4.3-1 and 4.3-2. Thus this TLAA remains valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

4.3.3 Effects of Reactor Water Environment on Fatigue Life

Industry test data indicate that certain environmental effects (such as temperature, dissolved oxygen content, and strain rate) in the primary systems of light water reactors could result in greater susceptibility to fatigue than would be predicted by fatigue analyses based on the ASME Section III design fatigue curves. The ASME design fatigue curves were based on laboratory tests in air and at low temperatures. Although the failure curves derived from laboratory tests were adjusted to account for effects such as data scatter, size effect, and surface finish, these adjustments may not be sufficient to account for actual plant operating environments.

As reported in SECY-95-245, the NRC believes that no immediate staff or licensee action is necessary to deal with the environmentally assisted fatigue issue. In addition, the staff concluded that it could not justify requiring a back fit of the environmental fatigue data to operating plants. However, the NRC concluded that, because metal fatigue effects increase with service life, environmentally assisted fatigue should be evaluated for any proposed extended period of operation for license renewal.

NUREG/CR-6260 applied the fatigue design curves that incorporated environmental effects to several plants and identified locations of interest for consideration of environmental effects. Section 5.5 of NUREG/CR-6260 identified the following component locations to be most sensitive to environmental effects for IPEC vintage Westinghouse plants. These locations and the subsequent calculations are directly relevant to IPEC.

1. Reactor vessel shell and lower head
2. Reactor vessel inlet and outlet nozzles
3. Pressurizer surge line (including hot leg and pressurizer nozzles)
4. RCS piping charging system nozzle
5. RCS piping safety injection nozzle
6. RHR Class 1 piping

IPEC evaluated the limiting locations using the guidance provided in NUREG-1801, Volume 2, Section X.M1. NUREG-1801 calls for using the guidance (formulas) provided in NUREG/CR-5704 and NUREG/CR-6583 to calculate environmentally assisted fatigue correction factors (F_{en}).

The environmentally adjusted CUFs for IPEC are shown in Table 4.3-13 (Unit 2) and Table 4.3-14 (Unit 3). All locations within the reactor vessel, shell, heads and nozzles have environmentally adjusted CUFs less than 1.0. All surge line piping and IP2 charging system nozzle have environmentally adjusted CUFs greater than 1.0.

IPEC has no plant-specific CUFs for some of the piping locations identified in NUREG-6260. At IPEC these locations are designed to ASME piping code B31.1, and no specific fatigue analysis resulting in CUFs is required by the current licensing basis.

The IP2 pressurizer surge nozzle has an environmentally adjusted CUF less than 1.0, while the IP3 pressurizer surge nozzle has an environmentally adjusted CUF of greater than 1.0. This is because the IP3 surge nozzle calculation includes the effects of the insurges/outsurges seen by these nozzles, while the IP2 analysis does not include these effects. IPEC will re-analyze the pressurizer surge line nozzle for both units, including insurge/outsurge and environmental effects. The pressurizer surge line nozzle re-analysis will review other components in the lower portion of the pressurizer to assure that the limiting IPEC location has been analyzed.

For those locations with CUFs less than 1.0, the TLAA has been projected through the period of extended operation per 10CFR54.21(c)(1)(ii).

Due to the factor of safety included in the ASME code, a CUF of greater than 1.0 does not indicate that fatigue cracking is expected; rather, it indicates that there is a higher potential for fatigue cracking at locations having CUFs exceeding 1.0. Tables 4.3-13 and 4.3-14 do not indicate that 40 year CUFs will exceed 1.0 because the EAF adjustment is not applied during the initial 40 years of operation. However, some of the CUFs will exceed 1.0 at the beginning of the period of extended operation when the EAF adjustment is added to the CUF calculation.

At least 2 years prior to entering the period of extended operation, for the locations identified in NUREG/CR-6260 for Westinghouse PWRs of the IPEC vintage, IPEC will implement one or more of the following:

(1) Refine the fatigue analyses to determine valid CUFs less than 1 when accounting for the effects of reactor water environment. This includes applying the appropriate Fen factors to valid CUFs determined in accordance with one of the following.

    For locations, including NUREG/CR-6260 locations, with existing fatigue analysis valid for the period of extended operation, use the existing CUF to determine the environmentally adjusted CUF.

    More limiting IPEC-specific locations with a valid CUF may be added in addition to the NUREG/CR-6260 locations. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component.

    Representative CUF values from other plants, adjusted to or enveloping the IPEC plant-specific external loads may be used if demonstrated applicable to IPEC.

    An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF.

(2) Manage the effects of aging due to fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC).

(3) Repair or replace the affected locations before exceeding a CUF of 1.0.

Should IPEC select the option to manage the aging effects due to environmental-assisted fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be submitted to the NRC at least 2 years prior to the period of extended operation.

Depending on the option chosen, which may vary by component, this TLAA will be projected through the period of extended operation per 10CFR54.21(c)(1)(ii) or the effects of environmentally assisted fatigue will be managed per 10CFR54.21(c)(1)(iii).

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Table 4.3-13
IP2 Cumulative Usage Factors for NUREG/CR-6260 Limiting Locations

NUREG-6260 Generic Location		IP2 Plant-Specific Location	Material Type	CUF of Record	Per NUREG/CR-6583 or NUREG/CR-5704	
					F_{en}	Environmentally Adjusted CUF
1	Vessel shell and lower head	Bottom head to shell	LAS	0.004	2.45	0.01
2	Vessel inlet and outlet nozzles	Reactor vessel inlet nozzle	LAS	0.05	2.45	0.12
2	Vessel inlet and outlet nozzles	Reactor vessel outlet nozzle	LAS	0.281	2.45	0.69
3	Pressurizer surge line nozzles	Pressurizer surge nozzle[1]	LAS	0.264	2.45	0.646
3	Pressurizer surge line piping	Surge line piping to safe end weld	SS	0.6	15.35	9.21
4	RCS piping charging system nozzle	Charging system nozzle	SS	0.99	15.35	15.20
5	RCS piping safety injection nozzle	NA[2]	SS	NA[2]	15.35	NA[2]
6	RHR Class 1 piping	NA[2]	SS	NA[2]	15.35	NA[2]

1. The surge line nozzle in the RCS piping is bounded by the surge line piping to safe end weld at the pressurizer nozzle.
2. The RCS piping at IP2 is designed to ANSI B31.1 and as such no fatigue analysis was performed and no CUFs were calculated.

Table 4.3-14
IP3 Cumulative Usage Factors for NUREG/CR-6260 Limiting Locations

	NUREG-6260 Location	IP3 Plant-Specific Location	Material Type	CUF of Record	Per NUREG/CR-6583 or NUREG/CR-5704	
					F_{en}	Environmentally Adjusted CUF
1	Vessel shell and lower head	Bottom head to shell	LAS	0.02	2.45	0.05
2	Vessel inlet and outlet nozzles	Reactor vessel inlet nozzle	LAS	0.049	2.45	0.12
2	Vessel inlet and outlet nozzles	Reactor vessel outlet nozzle	LAS	0.259	2.45	0.64
3	Pressurizer surge line nozzles	Pressurizer surge line nozzle[1]	LAS	0.9612	2.45	2.35
3	Pressurizer surge line piping	Surge line piping to safe end weld	SS	0.6	15.35	9.21
4	RCS piping charging system nozzle	NA[2]	SS	NA[2]	15.35	NA[2]
5	RCS piping safety injection nozzle	NA[2]	SS	NA[2]	15.35	NA[2]
6	RHR Class 1 piping	NA[2]	SS	NA[2]	15.35	NA[2]

1. The surge line nozzle in the RCS piping is bounded by the surge line piping to safe end weld at the pressurizer nozzle.
2. The RCS piping at IP3 is designed to ANSI B31.1 and as such no fatigue analysis was performed and no CUFs were calculated.

4.4 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRIC EQUIPMENT

Summary Description

IPEC evaluates environmentally qualified (EQ) electrical components using 10 CFR 50.49(f) qualification methods. Equipment qualifications evaluations that specify a qualification duration of at least 40 years, but less than 60 years, are considered TLAA for license renewal.

Evaluation

These TLAA have not been projected for the period of extended operation; rather, the aging effects associated with these analyses are managed by the Environmental Qualification of Electric Components (EQ) Program in accordance with 10 CFR 54.21(c)(1)(iii).

The EQ Program is an existing program established to meet IPEC commitments for 10 CFR 50.49. It is consistent with NUREG-1801, Section X.E1, "Environmental Qualification (EQ) of Electric Components." Consistent with NRC guidance provided in RIS 2003-09, no additional information is required to address GSI 168, "EQ of Electrical Components."

4.5 CONCRETE CONTAINMENT TENDON PRESTRESS

This section is not applicable as IPEC does not have pre-stressed tendons in the containment structures.

4.6 CONTAINMENT LINER PLATE AND PENETRATION FATIGUE ANALYSES

Unit 2

Summary Description

In 1973, a feedwater line cracked circumferentially resulting in damage to the liner plate causing containment liner plate buckling at the penetration for feedwater line #22. No repair was required for this buckling of the liner plate.

Studies were performed to evaluate the effects of fatigue on the deformed area of the liner due to predicted high strain-limited cycle loading during its projected forty-year life. The evaluation used an AEC-approved maximum strain and concluded that the strain load endurance limit of the material was 450 cycles at 7.7% strain. The evaluation estimated that the containment liner was likely to see 50 LOCAs (concurrent with earthquakes) at 1% strain, and 8 cycles from containment testing (1 pre-startup full pressure test at 6.5% strain and 7 cycles at 3.25% strain). This combines to 58 cycles at assorted strain (6.5% maximum strain). The evaluation conservatively projected a worst case of 60 cycles at 6.5% strain. As this projection was so far below the allowed 450 cycles at 7.7% strain, no further analysis was performed.

Evaluation

IP2 will not experience 50 LOCAs/earthquakes in 60 years of operation. Containment pressure testing is scheduled only once every 10 years. Therefore, the number of cycles experienced will continue to be less than the 60 cycles originally assumed and well below the 450 cycle limit in 60 years of operation. Therefore, the TLAA associated with the IP2 liner adjacent to the feedwater line #22 penetration remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

There are no other TLAA associated with IP2 containment liner plate or penetrations.

Unit 3

There are no TLAA associated with the IP3 containment liner plate or penetrations

4.7 OTHER PLANT-SPECIFIC TLAA

4.7.1 Reactor Coolant Pump Flywheel Analysis

Summary Description

The reactor coolant pump motors are provided with flywheels to increase rotational inertia, thus prolonging pump coast-down and assuring a more gradual loss of primary coolant flow to the core in the event that pump power is lost. The aging effect of concern is fatigue crack initiation and growth in the flywheel bore keyway from stresses due to starting the motor. Regulatory Guide 1.14 recommends periodic volumetric inspection of flywheels.

IPEC inspects the RCP flywheels as required by Technical Specifications (IP2: Section 5.5.5, IP3: Section 5.5.6). These inspections are performed at least once every 20 years in accordance with the staff approved WCAP-15666-A. WCAP-15666-A is based on 6000 start/stop cycles of a reactor coolant pump, an order of magnitude beyond the number of cycles that are expected in 60 years.

As indicated in Tables 4.3-1 and 4.3-2, the allowable number of heatup and cooldown cycles for 60 years of operation is 200 for Units 2 and 3. The analyzed number of cycles is far greater than the expected number, even if multiple reactor coolant pump starts are assumed in each startup/shutdown cycle. Because the 6000 cycles assumed in the analysis far exceeds the expected cycles in 60 years, and because the analysis is based on 60 years rather than 40 years, this analysis does not meet the 10CFR54(3)(a)(3) criteria for a TLAA. It does not involve time-limited assumptions defined by the current operating term or by an operating term less than the current operating term plus the period of extended operation requested in the license renewal application.

Evaluation

Evaluation is not applicable since the flywheel analysis is not a time-limited aging analysis as defined by 10 CFR 54.3. The analysis does not meet Part (3) of the definition in 10 CFR 54.3, Definitions.

4.7.2 Leak before Break

Summary Description

Leak before break (LBB) analyses evaluate postulated flaw growth in piping. These analyses consider the thermal aging of the CASS piping and fatigue transients that drive flaw growth over the operating life of the plant. Because these two analysis considerations could be influenced by time, LBB analyses are identified as potential TLAA.

Unit 2

The structural design of IP2 considered and protected against the effect of postulated reactor coolant loop pipe ruptures. LBB analyses have been documented in WCAP-10977, WCAP-10977 Supplement 1, and WCAP-10931. The time-related assumptions in the analyses include the thermal aging of cast austenitic stainless steel and the fatigue crack growth analysis. These two assumptions are addressed below.

Unit 3

The structural design of IP3 considered and protected against the effect of postulated reactor coolant loop pipe ruptures. LBB analyses have been documented in Appendix A of WCAP-8228. The time-related assumptions in the analyses include the thermal aging of cast austenitic stainless steel and the fatigue crack growth analysis. These two assumptions are addressed below.

Thermal Aging of CASS

The first analysis consideration that could be influenced by time is the material properties of cast austenitic stainless steel (CASS) used in the pipe fittings. Thermal aging causes an elevation in the yield strength of CASS and a decrease in fracture toughness, the decrease being proportional to the level of ferrite in the material. Thermal aging in these stainless steels will continue until a saturation, or fully aged, point is reached. The analyses used fully aged toughness values. As the LBB evaluations for both units use saturated (fully aged) fracture toughness properties, these analyses do not have a material property time-dependency and are not considered TLAA.

Fatigue Crack Growth

The second analysis consideration that could be influenced by time is the accumulation of actual fatigue transient cycles. A fatigue crack growth analysis of the reactor vessel inlet nozzle to safe-end region was performed to determine its sensitivity to the presence of small cracks. The nozzle to safe-end connection was selected because crack growth calculated at this location is representative of the entire primary loop. The nozzle to safe-end connection configuration includes an SA-508 Class 2 or Class 3 stainless steel clad nozzle connected to a stainless steel safe end by a nickel-based alloy weld. The crack growth due to fatigue was evaluated assuming the reactor vessel experienced the total allowable numbers of normal, upset, and test transients.

Evaluation

The calculated fatigue crack growth for 40 years was very small (less than 50 mils) regardless of the material evaluated. As noted in Section 4.3.1, the projections for 60 years of operation indicate that the numbers of significant transients for IP2 or IP3 will not exceed the design analyzed values. Thus the IP2 and IP3 analyses will remain valid during the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.7.3 Steam Generator Flow Induced Vibration and Tube Wear

Unit 2

Summary Description

The IPEC Unit 2 steam generators were evaluated with respect to flow induced vibration (tube wear) for the power increase. The analysis of the effects of steam generator flow induced vibration on tube wear assumed 40 years of operation.

Evaluation

The IP2 replacement steam generators went into service in January 2000 and will thus have less than 40 years of service at the end of the period of extended operation (September 2033). Therefore the analysis of flow induced vibration effects on tube wear will remain valid through the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Unit 3

Summary Description

The IPEC Unit 3 steam generators were evaluated with respect to flow induced vibration (tube wear) for the power increase. The maximum pre-uprate predicted tube wear was 1.3 mils. As a result of the 4.8% uprate, the increase in tube wear is 87%. The post-uprate wear over 40 years is approximately 2.4 mils (~4.9% through-wall wear). This amount of wear will not significantly affect the tube integrity. As the IP3 replacement steam generators went into service in 1989, they will have reached 46.5 years of service at the end of the period of extended operation, 2035. Therefore these analyses are considered TLAA.

Evaluation

As the tube wear is a function of time in service, it is appropriate to project the additional wear for the period of extended operation as 46.5/40 times the 40-year wear. Projected wear is 2.8 mils (~5.7% through-wall) by the end of the period of extended operation. This is still well below the allowable 40% through-wall wear depth (20 mils). Hence the period of extended operation will not result in unacceptably high tube wear. Thus, the TLAA associated with Unit 3 tube wear has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.7.4 References

None