ATTACHMENT 3 TO NL-04-095

ERRATA PAGES FOR WCAP-16157-NP

INDIAN POINT NUCLEAR GENERATING UNIT 2
STRETCH POWER UPRATE NSSS AND BOP LICENSING REPORT

Refer to Entergy letter NL-04-005, dated January 29, 2004 (Attachment III)

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2
DOCKET NO. 50-247

SUMMARY TABLE FOR WCAP-16157-NP ERRATA PAGES

Section	Page	Location of Revision	Revision
4.2	4.2-8	Last paragraph in subsection 4.2.4.1	Revise value for CST volume to 295,150 gallons (twice)
4.2	4.2-10	Subsection 4.2.6 under "Auxiliary Feedwater System"	Revise value for CST volume to 295,150 gallons (twice)
5.4	5.4-9	Subsection 5.4.3.4 under "Steam Generator and RCP Frames," first bullet under fourth paragraph	Change $F_t = 0.6\ F_y$ to $F_t = 0.6\ S_y$
5.9	5.9-1	First sentence of last paragraph of 5.9.1	Change "...a Model D Series 84 pressurizer...." to "...a Model 44 pressurizer...."
5.9	5.9-5	Last paragraph in subsection 5.9.3.3	Change RT_{PRS} to RT_{PTS}
6.5	6.5-42	Table 6.5-7, footnotes	Consistent with text on page 6.5-8, change footnotes to read: 3. M&E exiting the steam generator side of the break 4. M&E exiting the pump side of the break
6.5	6.5-47	Table 6.5-8, footnotes	Consistent with text on page 6.5-8, change footnotes to read: 3. M&E exiting the steam generator side of the break 4. M&E exiting the pump side of the break
6.5	6.5-51	Table 6.5-10, footnotes	Consistent with text on page 6.5-8, change footnotes to read: 3. M&E exiting the steam generator side of the break 4. M&E exiting the pump side of the break
6.6	6.6-19	Subsection 6.6.5, first paragraph	Correct typo by changing time period in last sentence from "...8- to 40-hour..." to "...8- to 30-hour..."
6.6	6.6-39	Table 6.6-19	Correct typo by changing time period in table from 8- to 40-Hour to 8- to 30-Hour
6.8	6.8-1	Last sentence of fourth paragraph of subsection 6.8.1	Change to read: For IP2, WCAP-10858P-A, *AMSAC Logic 1, AMSAC Actuation on Low Steam Generator Water Level,* was used.
9.12	9.12-3	Last sentence of subsection 9.12.2	Revise value for CST volume to 295,150 gallons

the FRVs fully open (Reference 1). The current Feedwater Pump Speed Control Program is set to provide an FRV pressure drop of approximately 166 psi at full-load, and this pressure drop results in an FRV lift of about 81 percent.

The hydraulic evaluation of the C&FS for the range of design parameters approved for the SPU indicates the lift of the FRVs at full power will increase by as much as 5.1 percent (from 81 to 86.1 percent at T_{avg} of 549°F) with the present Feedwater Pump Speed Control Program.

The hydraulic evaluation of the C&FS (refer to Section 9.4) also concluded that the C&FS could maintain adequate feedwater pump suction pressure, assuming 1 drain tank pump remains in service following a large load rejection.

To provide effective control of flow during normal operation, the FRVs are required to stroke open or closed in 20 seconds over the anticipated inlet pressure control range (approximately 0 to 1600 psig). Additionally, rapid closure of the FRVs is required after receiving a trip close signal in order to mitigate certain transients and accidents. These requirements are not affected by the SPU.

4.2.4 Auxiliary Feedwater System

The AFWS supplies feedwater to the secondary side of the steam generators at times when the normal feedwater system is not available, thereby maintaining the steam generator heat sink. The system provides feedwater to the steam generators during normal unit startup, hot standby, and cooldown operations and also functions as an engineered safety feature (ESF). In the latter function, the AFWS is required to prevent core damage and system overpressurization during transients and accidents, such as a loss of normal feedwater or a secondary system pipe break. The minimum flow requirements of the AFWS are dictated by accident analyses, and since the SPU affects these analyses, evaluations of the limiting transients and accidents are performed to confirm that the AFWS performance is acceptable at the SPU conditions. These evaluations are described in Section 6 of this report and show acceptable results.

4.2.4.1 AFW Storage Requirements

The AFWS pumps are normally aligned to take suction from the condensate storage tank (CST). To fulfill the ESF design functions, sufficient feedwater must be available during transient or accident conditions to enable the plant to be placed in a safe shutdown condition.

The limiting transient with respect to CST inventory requirements is the LOOP transient. The IP2 licensing basis requires that, in the event of a LOOP, sufficient CST useable inventory must be available to bring the unit from full-power to hot-standby conditions, and maintain the plant at hot standby for 24 hours.

Since the required CST inventory is a function of plant-rated power and other NSSS design parameters, a new analysis was performed to determine the required inventory for the range of NSSS design parameters approved for SPU. This analysis is based on the following conservative assumptions:

- Reactor trip occurs from 102 percent of rated core power (3216 MWt), from a low-low water level in the steam generators. A 2-second delay is assumed before reactor trip following LOOP.
- Steam is released from the steam generators at the first safety valve setpoint plus setting tolerance for drift.
- The steam generators are filled back up to 52-percent narrow range water level.
- The CST operating fluid temperature is at the maximum allowable value (120°F).

The analysis concluded that a minimum required useable inventory of 295,150 gallons is required to meet the plant licensing bases for the range of NSSS design parameters approved for SPU. *The CST Technical Specification requirement of 360,000 gallons ensures a usable* volume of 295,150 gallons.

4.2.5 Steam Generator Blowdown System

The Steam Generator Blowdown System (SGBS) is used to control the chemical composition of the steam generator secondary side water within the specified limits. The SGBS also controls the buildup of solids in the steam generator secondary.

The blowdown flow rates required during plant operation are based on chemistry control and tube-sheet sweep requirements to control the buildup of solids. The blowdown flow rate required to control chemistry and the buildup of solids in the steam generators is based on allowable condenser in-leakage, total dissolved solids in the plant circulating water, and the allowable primary to secondary leakage. Since these variables are not affected by the SPU, the blowdown required to control secondary chemistry and steam generator solids will not be affected by the SPU.

The inlet pressure to the SGBS varies with steam generator operating pressure. Therefore, as steam generator full-load operating pressure decreases, the inlet pressure to the SGBS control valves decreases and the valves must open to maintain the required blowdown flow rate into the system flash tank. The 1.4-percent MUR NSSS design parameters (Table 2.1-1) permit a maximum decrease in steam pressure from no-load to full-load of 370 psi (that is, from 1020 to 650 psia). Based on the revised range of SPU NSSS design parameters (Table 2.1-2), the no-load steam pressure (1020 psia) remains the same, and the current minimum allowable full-load steam pressure (650 psia) due to steam generator tubesheet ΔP limits does not change. Therefore, the range of design parameters approved for the SPU will not affect blowdown flow capability.

4.2.6 Conclusions

The following is a brief summary of the NSSS/BOP interface evaluation conclusions for the IP2 SPU Program.

Main Steam System

- The capacity of the installed MSSVs is adequate to meet the original sizing bases for the approved range of NSSS design parameters.
- The capacity of the installed ARVs is adequate to meet the original sizing bases for the approved range of NSSS design parameters.
- SPU does not adversely affect the criteria for the MSIVs and MSIV bypass valves.

Steam Dump System

An evaluation of the Steam Dump System indicates that the minimum system capacity is approximately 34 percent of the SPU full-load steam flow at the current minimum allowable full-load steam pressure of 650 psia. At full-load steam pressures higher than 650 psia, steam dump capacity would increase. The control system's margin to trip analysis provides an evaluation of the adequacy of steam dump in conjunction with the control system setpoints (see Section 4.3 of this report).

Condensate and Feedwater System

- The lift of the FRVs at full power will increase by as much as 5.1 percent (from 81 to 86.1 percent at T_{avg} of 549°F) with the present Feedwater Pump Speed Control Program.
- Per Section 9.4, feedwater pump suction pressure is adequate, assuming 1 drain tank pump remains in service following a large load rejection.

Auxiliary Feedwater System

- The AFWS is capable of delivering the minimum flow requirements for the SPU (see Section 6).

- The CST minimum useable inventory of 295,150 gallons is required to meet the plant licensing bases for the range of NSSS design parameters approved for SPU. The Technical Specification value of 360,000 gallons ensures a usable volume of 295,150 gallons.

Steam Generator Blowdown System

- The blowdown flow required to control secondary chemistry and steam generator solids is not affected by the SPU.

- The NSSS design parameters approved for the SPU coupled with the current minimum allowable full-load steam pressure will not affect blowdown flow capability.

4.2.7 References

1. *Indian Point Nuclear Generating Unit No. 2 1.4-Percent Measurement Uncertainty Recapture Power Uprate License Amendment Request Package*, Entergy Nuclear Operations, Inc., November 2002.

2. *ASME Boiler and Pressure Vessel Code*, Section III, "Rules for Construction of Nuclear Vessels," 1965 Edition with Winter 1965 Addenda, The American Society of Mechanical Engineers, New York, NY.

3. *Indian Point Nuclear Generating Unit No. 2, Updated Final Safety Analysis Report*, Docket No. 50-247.

The load combinations were based on Table 1.11-2 of the UFSAR (Reference 1).

The loading combinations were applicable for all support components. However, the allowable stress and loads were different and were addressed separately for the steam generator and RCP frames, the tie rods, snubbers, equipment hold-down bolts, embedments, and the RPV support.

5.4.3.4 Acceptance Criteria

The acceptance criteria for the IP2 RCSES as indicated in the *Indian Point Nuclear Generating Unit No. 2 Updated Final Safety Analysis Report* (Reference 1) are based upon Table 1.11-2, in combination with the criteria discussed below.

Steam Generator and RCP Frames

Per Section 4.1.7 and Table 4.1-9 of the UFSAR, the original piping design code for IP2 supports is the 1955 Edition of USAS B31.1 (Reference 2). The USAS B31.1 Code does not provide detailed guidance for the evaluation of piping supports. Common practice is to use the American Institute of Steel Construction (AISC) Specification (Reference 10).

The Sixth Edition AISC Specification (Reference 10) was used for evaluating the piping supports.

For Load Case 1, the allowable stresses provided in the AISC code were used (allowables were based on the actual temperature of the steel). For Load Case 2, the allowables could be increased by 1/3, however, compressive buckling stresses were limited to 2/3 of critical buckling.

For Load Cases 3 and 4, well-defined criteria were not available in the AISC Specification. The criterion is that "Deflections and stresses of supports limited to maintain supported equipment within their stress limits." This correlates to limiting the deflection of the supports such that additional stresses do not occur in the supported piping/equipment. Acceptable means of satisfying the above criteria are to use the faulted increase factors provided in Appendix F of the 1974 ASME Section II Code for Supports, that is, F-1370(a) and F-1370(c) (Reference 11). These rules state that the increase factor for faulted-condition loads can be increased above the Level A (AISC allowables) by:

- Increase factor = minimum ($1.2 \times S_y / F_t$, $0.7 \times S_u / F_t$), since $F_t = 0.6\ S_y$ for the frame members being considered

- Increase factor = minimum {2, 0.7 x S_u / (0.6 x S_y)}

- Section F-1370(c) states that loads shall not exceed 2/3 of the critical buckling load

Steam Generator and RCP Tie Rods

The steam generator and RCP tie rods are strictly tension members. As such, the allowable loads were based on the minimum of the turnbuckle allowables, the tensile area of the tie rods, and the compressive area under the nuts.

Concrete Embedments

The embedment allowable loads were taken from design information used in previous analyses.

Snubbers

The maximum allowable load per snubber was taken from design information used in previous analyses.

RCP and Steam Generator Holddown Bolts

The RCP bolts evaluation was in accordance with AISC, Seventh Edition (Reference 12). For Load Cases 3 and 4, the guidance provided in ASME Code Case 1644-6 (Reference 13) was used.

The steam generator feet connections were evaluated by comparison with design information used in previous analyses.

RPV Supports

The reactor vessel support evaluations were based on WCAP-9117 (Reference 14). The letter from W. J. Cahill, Jr. to the Director of Nuclear Regulatory Regulation, June 15, 1978, addresses the applicability of WCAP-9117 to Indian Point Unit 2 (Reference 15).

5.4.3.5 RCSES Analysis and Results

The loads on the steam generator, RCP, and RPV supports meet the acceptance criteria provided in subsection 5.4.3.4 of this report.

A summary of the results is provided in Table 5.4-2.

5.9 NSSS Components Fracture Integrity

5.9.1 Introduction

The Indian Point Unit 2 (IP2) Stretch Power Uprating (SPU) Program involves changes that affect each of the primary NSSS components. This section addresses the effects of the SPU on the fracture integrity of the ferritic Class 1 components, specifically the reactor vessel, steam generators, and pressurizer. These are the components for which non-ductile failure must be considered, according to the requirements of the *American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code*, Section III (Reference 1).

The IP2 reactor vessel was designed to Section III of the 1965 ASME Code. The non-ductile failure requirements were not incorporated into the Code until Appendix G (Reference 2) was added to the 1972 Summer Addenda. The Appendix G analysis for IP2 was completed in November 1974 to comply with the requirements of 10CFR50. That 1974 analysis was used as the basis for the current reactor vessel Appendix G analysis for the SPU Program.

IP2 has the Model 44F steam generator and a Model 44 pressurizer. Generic analyses were used for Appendix G qualification of the steam generator and pressurizer, respectively. These generic analyses were used as the base analyses for these components to assess the effect of the SPU.

5.9.2 Input Parameters and Assumptions

The key input parameters are the stresses in the various components, and the fracture properties of the components. The fracture integrity evaluations for the SPU Program draw on the ASME Code design re-evaluations for the reactor vessel, steam generator components, and pressurizer in Sections 5.1, 5.6, and 5.7, respectively.

The stresses for the baseline reactor vessel analysis were taken from the original IP2 reactor vessel fracture analysis. The original design transients were considered in that reactor vessel fracture analysis, and have been updated to account for the transients discussed in Section 3. The IP2 reactor vessel was previously evaluated as part of the Replacement Steam Generator (RSG) Program. The structural evaluations that were performed are included in an addendum to the reactor vessel stress report and were used in the SPU Program.

The stresses for the baseline steam generator analysis were taken from a typical Model F steam generator stress report. The Model D Series 84 pressurizer analysis was used as the base analysis for the IP2 pressurizer. The stresses obtained from those analyses were adjusted using scale factors previously discussed in earlier sections of this report.

5.9.3 Description of Analyses and Evaluations

5.9.3.1 Methodology

The approach used in the evaluations is a direct application of ASME B&PV Appendix G of Section III (Reference 1). A flaw is postulated, and the crack driving force or stress intensity factor is calculated after adding a safety factor of 2 on the primary stresses. The applied stress intensity factor is then compared with the material fracture toughness, as characterized by the reference stress intensity factor (K_{IR}) toughness curve contained in Appendix G. The following sections detail each of these steps.

5.9.3.2 Stress Intensity Factor Calculations and Postulated Flaw Size

The maximum defect assumed in Appendix G (Reference 1) is a sharp surface defect normal to the direction of the maximum stress. The typical flaw is assumed to be semi-elliptical with an aspect ratio of 1:6 and a depth of one quarter of the vessel wall thickness.

Appendix G (Reference 1) recognizes that some regions cannot be expected to meet the requirements of a one-quarter thickness defect; it states that "smaller defect sizes may be used on an individual case basis if a smaller size of maximum postulated defect can be assured." Welding Research Bulletin 175, *PVRC Recommendations on Toughness Requirements for Ferritic Materials* (Reference 3), provides procedures for considering postulated defect sizes smaller than one quarter of the wall thickness.

The combination of examinations originally required by ASME B&PV Section III (Reference 1) (radiography and surface exams) and the volumetric examination required by Section XI (ultrasonic mapping) are capable of detecting flaws of the magnitude of those assumed for the discontinuity regions for the SPU analyses.

The stress intensity factor, K_I, was calculated for both primary and secondary stress for the limiting transients.

A K_{IR} upper shelf of 200 ksi $\sqrt{in.}$ has been adopted for unirradiated material, and a shelf of 170 ksi $\sqrt{in.}$ has been fixed for irradiated material provided the upper shelf Charpy energy exceeds 50 ft lb. This is a generally accepted industry practice, as shown for example in EPRI Report NP-7195R (Reference 4).

Neutron irradiation adversely affects the toughness properties of the reactor vessel steel. The neutron embrittlement of the steel has been found to be a function of the copper content of the steel for given fluences.

A consequence of a decrease in the toughness properties is a shift in the fracture toughness curve to a higher temperature. Quantitatively, this shift can be assessed by determining the shift to higher temperatures of the initial reference nil ductility temperature RT_{NDT}.

The Nuclear Regulatory Commission (NRC) has also developed copper trend curves for the prediction of RT_{NDT} versus fluence (Reference 5). These curves are presented in Regulatory Guide (RG) 1.99, Revision 2. RG 1.99 curves predict RT_{NDT} shift as a function of nickel content as well as copper content.

The fracture toughness curve, indexed to $T - RT_{NDT}$, therefore, will shift along the abscissa by a value equal to ΔRT_{NDT} for a given level of irradiation and copper content as indicated by the copper trend curves. The RT_{NDT} values at the end of life (EOL) differ sufficiently for the locations, so different reference fracture toughness curves are required.

The fluence drops drastically at a short longitudinal distance beyond the vicinity of the core assemblies as illustrated by Figure 5.9-3. For instance, the nozzles are located more than 30 inches above the top level of the core assembly. The curve in Figure 5.9-3 shows that the fluence is about 0.6 percent of the peak fluence value. This is a typical curve, and not meant to represent IP2 specifically. Thus, the irradiation effects at the nozzle areas become insignificant due to the nozzle locations relative to the core.

The upper head and lower head junctions are located still farther from the core ensuring that there will be no significant irradiation effect at those locations. Consequently, only the K_{IR} curve of the vessel beltline, which is exposed to the maximum irradiation, has been adjusted to account for the shift in RT_{NDT} resulting from irradiation.

The material properties of the reactor vessel are tabulated in Table 5.9-1 along with the initial RT_{NDT}, predicted EOL RT_{NDT}, EOL fluence at the 1/4t location, and cross section thickness of each critical location. For the beltline region, EOL fluence and RT_{PTS} values in Table 5.1-2 of Section 5.1 are used.

5.9.3.4 Acceptance Criteria

The K_I values calculated for the affected regions of the reactor vessel, steam generator and pressurizer were compared with the corresponding material fracture toughness, K_{IR}. Protection against non-ductile failure is then assured if the K_I values were below the K_{IR} values.

The expression used to calculate the stress intensity factor was derived for application to a flaw in a flat plate. An axisymmetrical body provides more constraint than a flat plate does. So, the stress intensities calculated by Appendix G (Reference 1) will be higher than the actual values in the reactor vessel and steam generators.

5.9.4 Analysis and Results

Reactor Vessel—The procedures of Appendix G (Reference 1) were applied to 4 critical locations in the reactor vessel: the bottom head to shell junction, the beltline region, the closure head to upper flange region, and the outlet nozzle to shell region.

The original reactor vessel fracture evaluation was used as the baseline for assessing the effects of the SPU Program. The secondary stresses were adjusted to incorporate the changes described in Section 5.1 for the affected design transients. Since the pressure does not change measurably, the primary stresses are identical to the original analysis results. The reference flaw size was one quarter of the section thickness in all cases, except for the outlet nozzle where a reduced defect size of 1/5t was utilized. The justification for a 1/5t defect for the nozzle is based on the availability of highly reliable non-destructive inspection techniques that assure capability of detecting such a flaw, because of the greater cross-section thickness at the nozzle-shell juncture, this flaw size is negligibly smaller than a ¼t defect in the other areas of interest.

The combined K_I values for each design transient in Table 5.9-2 are compared with the appropriate EOL K_{IR} curve for the critical locations. Exceptions to this are the plant heatup and cooldown, and ISLH test conditions, which are controlled to be in compliance with Appendix G (Reference 1) margins through the plant Technical Specifications. Table 5.9-2 also shows minimum temperature during each transient for the SPU that is conservatively used for the Appendix G calculation.

The results of the analysis are plotted in Figures 5.9-4 through 5.9-7 for the bottom head to shell junction, the beltline region, the closure head to upper flange region and the outlet nozzle to shell region, respectively. Each transient is represented as a point corresponding to the stress intensity factor and the corresponding minimum temperature during that transient.

Table 6.5-7 (Cont.) DEPS Break (minimum safeguards case) Blowdown M&E Releases for IP2 SPU				
	Break Path No. 1[1]		Break Path No. 2[2]	
Time	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
17.8	4088.8	3361.4	8059.9	3783.8
18.0	4003.3	3338.9	8074.0	3771.4
18.2	3911.2	3320.1	7968.5	3702.9
18.4	3816.0	3306.6	7735.9	3574.5
18.6	3711.0	3295.4	7557.1	3468.0
18.8	3599.3	3288.9	7489.2	3409.1
19.0	3476.9	3281.6	7426.3	3351.2
19.2	3339.5	3274.3	7291.6	3261.6
19.4	3112.8	3201.9	7025.2	3115.2
19.6	2872.0	3112.9	6504.2	2859.2
19.8	2656.5	3017.4	5888.0	2563.0
20.0	2468.0	2906.1	5468.7	2349.1
20.2	2308.0	2789.5	5535.0	2328.4
20.4	2123.5	2598.3	6023.0	2465.5
20.6	1956.7	2408.2	6647.5	2649.6
20.8	1812.0	2238.1	6252.1	2446.3
21.0	1685.8	2087.7	5729.5	2214.5
21.2	1569.8	1948.2	5441.1	2075.4
21.4	1456.6	1811.1	5220.4	1957.8
21.6	1354.4	1686.8	5013.5	1843.9
21.8	1255.2	1565.8	4785.1	1723.3
22.0	1167.3	1458.1	4545.6	1601.6
22.2	1080.0	1351.1	4308.3	1484.2
22.4	1007.1	1261.7	4080.5	1374.5
22.6	925.4	1160.7	3859.9	1271.9
22.8	867.4	1089.4	3646.6	1176.7
23.0	825.3	1037.2	3444.0	1089.3
23.2	789.4	992.7	3240.0	1005.3
23.4	746.7	939.5	3027.4	922.4
23.6	698.4	879.3	2818.3	844.1
23.8	652.2	821.7	2632.9	775.9
24.0	605.7	763.5	2411.8	700.2
24.2	558.3	704.1	2167.0	620.5
24.4	510.1	643.7	1892.7	535.3
24.6	461.8	583.1	1579.6	442.0
24.8	413.7	522.6	1221.0	338.7
25.0	364.5	460.6	822.5	226.8
25.2	313.9	396.9	427.8	117.6
25.4	261.3	330.6	101.7	28.0
25.6	207.1	262.2	.0	.0
25.8	156.1	197.8	.0	.0

Table 6.5-7 (Cont.) DEPS Break (minimum safeguards case) Blowdown M&E Releases for IP2 SPU				
	Break Path No. 1[1]		Break Path No. 2[2]	
Time	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
26.0	102.9	130.6	.0	.0
26.2	40.0	50.9	.0	.0
26.4	.0	.0	.0	.0

Notes:

1. M&E exiting the steam generator side of the break
2. M&E exiting the pump side of the break

Table 6.5-8 (Cont.) DEPS Break (minimum safeguards case) Reflood M&E Releases for IP2 SPU				
	Break Path No.1[1]		Break Path No.2[2]	
Time	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
227.5	117.6	138.7	148.9	67.2
229.5	117.9	138.9	149.0	67.4
231.5	118.1	139.2	149.1	67.5
233.5	118.3	139.5	149.1	67.6
235.5	118.5	139.7	149.2	67.7
237.5	118.8	140.0	149.3	67.8
239.5	119.0	140.3	149.4	67.9
239.7	119.0	140.3	149.4	67.9

Notes:

1. M&E exiting the steam generator side of the break
2. M&E exiting the pump side of the break

Table 6.5-9

DEPS Break
(minimum safeguards case)
Principle Parameters During Reflood for IP2 SPU

Time Seconds	Flooding Temp (°F)	Flooding Rate (in/sec)	Carryover Fraction (—)	Core Height (feet)	Downcomer Height (feet)	Flow Fraction (—)	Injection Total (pounds mass per second)	Injection Accumulator (pounds mass per second)	Injection Spill (pounds mass per second)	Enthalpy Btu/lbm
26.4	190.0	.000	.000	.00	.00	.250	.0	.0	.0	.00
27.1	188.6	20.878	.000	.50	1.17	.000	6713.2	6713.2	.0	99.50
27.4	187.2	24.643	.000	1.09	1.23	.000	6647.4	6647.4	.0	99.50
27.8	186.9	2.520	.126	1.34	2.09	.225	6528.7	6528.7	.0	99.50
28.1	187.0	2.568	.184	1.39	2.78	.288	6457.6	6457.6	.0	99.50
28.8	187.3	2.426	.300	1.50	4.37	.322	6325.9	6325.9	.0	99.50
29.4	187.5	2.365	.373	1.58	5.66	.333	6214.4	6214.4	.0	99.50
33.7	189.4	2.656	.616	2.00	14.86	.353	5534.7	5534.7	.0	99.50
35.5	190.2	4.041	.665	2.18	16.12	.552	4887.5	4887.5	.0	99.50
37.5	191.2	3.817	.693	2.39	16.12	.548	4666.9	4666.9	.0	99.50
38.7	191.8	3.708	.703	2.51	16.12	.545	4554.1	4554.1	.0	99.50
44.7	195.4	3.354	.726	3.00	16.12	.529	4072.9	4072.9	.0	99.50
45.5	195.9	3.319	.727	3.06	16.12	.527	4017.0	4017.0	.0	99.50
46.5	196.5	3.905	.732	3.14	16.05	.638	.0	.0	.0	.00
47.5	197.3	4.688	.732	3.24	15.59	.640	.0	.0	.0	.00
50.3	199.4	3.866	.735	3.51	14.51	.608	358.5	.0	.0	78.02
56.6	204.5	3.371	.738	4.00	13.08	.603	367.3	.0	.0	78.02
64.5	211.7	2.865	.737	4.54	11.70	.596	375.5	.0	.0	78.02
72.4	219.2	2.447	.735	5.00	10.70	.587	381.4	.0	.0	78.02
83.5	229.1	1.994	.731	5.54	9.81	.571	386.6	.0	.0	78.02
94.6	236.8	1.674	.727	6.00	9.37	.554	389.8	.0	.0	78.02
109.5	244.9	1.407	.723	6.52	9.23	.532	392.0	.0	.0	78.02

Table 6.5-10 (Cont.) DEPS Break (minimum safeguards case) Post-Reflood M&E Releases for IP2 SPU				
	Break Path No. 1(1)		Break Path No. 2(2)	
Time	Flow	Energy	Flow	Energy
Seconds	lbm/Sec	Thousands Btu/Sec	lbm/Sec	Thousands Btu/Sec
429.8	205.4	256.1	199.9	126.6
434.8	205.1	255.7	199.5	126.1
439.8	204.6	255.1	199.1	125.7
444.8	204.0	254.3	198.8	125.3
449.8	203.4	253.6	198.4	124.8
454.8	85.7	106.9	310.3	154.0
627.6	85.7	106.9	310.3	154.0
627.7	87.5	108.4	308.4	147.8
629.8	87.5	108.3	308.5	147.6
1262.4	87.5	108.3	308.5	147.6
1262.5	74.8	86.1	321.1	31.0
1500.5	71.4	82.2	324.5	31.6
1500.6	71.4	82.2	167.3	63.8
2334.0	64.4	74.1	174.3	65.0
2334.1	64.4	74.1	174.3	65.0
3600.0	57.2	65.8	181.5	66.3
3600.1	54.3	62.5	184.3	48.7
10000.0	39.5	45.5	199.2	52.6
23400.0	31.9	36.7	206.8	54.6
23400.1	31.9	36.7	73.3	19.4
100000.0	21.1	24.3	84.1	22.2
1000000.0	9.1	10.4	96.1	25.4
10000000.0	2.8	3.3	102.4	27.0

Notes:

1. M&E exiting the steam generator side of the break
2. M&E exiting the pump side of the break

Table 6.5-11 DEPS Break Mass Balance (minimum safeguards case) for IP2 SPU								
		Mass Balance						
Time (seconds)		.00	26.40	26.40+ δ	239.71	627.68	1262.40	3600.00
		Mass (thousand lbm)						
Initial	In RCS and ACC	714.25	714.25	714.25	714.25	714.25	714.25	714.25
Added Mass	Pumped injection	.00	.00	.00	74.24	227.83	479.16	1074.54
	Total added	.00	.00	.00	74.24	227.83	479.16	1074.54
*** TOTAL AVAILABLE ***		714.25	714.25	714.25	788.50	942.08	1193.41	1788.79
Distribution	Reactor coolant	524.25	58.26	84.53	144.26	144.26	144.26	144.26
	Accumulator	190.00	126.74	100.47	.00	.00	.00	.00
	Total contents	714.25	185.00	185.00	144.26	144.26	144.26	144.26
Effluent	Break flow	.00	529.24	529.24	644.22	801.15	1052.40	1647.79
	ECCS spill	.00	.00	.00	.00	.00	.00	.00
	Total effluent	.00	529.24	529.24	644.22	801.15	1052.40	1647.79
*** TOTAL ACCOUNTABLE ***		714.25	714.24	714.24	788.48	945.41	1196.65	1792.04

Composite temperature and pressure profiles for the 600-second operator action time are provided in Figures 6.6-2 and 6.6-3 for the header and loop breaks for winter and summer conditions. Section 10.9.3 uses this information and the individual case profiles to address the qualification of the equipment for IP2 at the SPU conditions.

6.6.5 Steam Releases for Radiological Dose Analysis

The vented steam releases have been calculated for the locked rotor and steamline break events. Table 6.6-19 summarizes the vented steam releases from the operable steam generators as well as auxiliary feedwater flows for the 0- to 2-hour time period, the 2- to 8-hour time period, and the 8- to 30-hour time period for each of these events.

No explicit assumption is considered in these analyses regarding Steam Generator Blowdown System isolation. The implied assumption is that the entire inventory of the steam generators is released to the environment and no loss of inventory through the blowdown line is considered. This provides a conservative calculation of the quantity of steam vented during the noted time periods.

The steam releases discussed in this section have been provided as inputs to the radiological dose analyses (See subsection 6.11.9) in support of the IP2 SPU Program.

6.6.6 References

1. *Indian Point Nuclear Generating Unit No. 2, Updated Final Safety Analysis Report*, Docket No. 50-247.

2. *Indian Point Unit 2 Technical Specifications, Amendment 235*, February 6, 2003.

3. ANSI/ANS-5.1-1979, *American National Standard for Decay Heat Power in Light Water Reactors*, The American Nuclear Society Standards Institute, Inc., LaGrange Park, Illinois, August 1979.

4. WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Non-Proprietary), *LOFTRAN Code Description*, T. W. T. Burnett, et al., April 1984.

5. WCAP-8822 (Proprietary) and WCAP-8860 (Nonproprietary), *Mass and Energy Releases Following a Steam Line Rupture*, September 1976; WCAP-8822-S1-P-A (Proprietary) and WCAP-8860-S1-A (Non-Proprietary), *Supplement 1 – Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture*, September 1986; WCAP-8822-S2-P-A (Proprietary) and WCAP-8860-S2-A (Nonproprietary), *Supplement 2 – Impact of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture for Dry and Subatmospheric Containment Designs*, September 1986.

6. WCAP-8327 (Proprietary), WCAP-8326 (Non-Proprietary), *Containment Pressure Analysis Code (COCO), July 1974.*

7. Indian Point Power Station Unit 2, *Updated Final Safety Analysis Report*, Rev. 18, June 2003.

8. 10CFR50 Appendix A, *General Design Criteria for Nuclear Power Plants.*

9. WCAP-10961 *(Proprietary), Steamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment, Report to the Westinghouse Owners Group High Energy Line Break/Superheated Blowdowns Outside Containment Subgroup*, Rev. 1, October 1985.

10. 10CFR50.49, *Environmental Qualification Of Electric Equipment Important To Safety For Nuclear Power Plants*, 66 FR 64738, December 14, 2001.

11. NRC IE Information Notice 84-90, *Main Steam Line Break Effect on Environmental Qualification of Equipment*, December 07, 1984.

12. NAI 8907-02, *GOTHIC Containment Analysis Package User Manual*, Version 7.0, Rev. 13, July 2001.

Table 6.6-19 Vented Steam Releases from Operable Steam Generators and Auxiliary Feedwater Flows for the 0 – 2, 2 – 8, and 8 – 30 Hr Time Periods						
Event	Vented Steam Release			Auxiliary Feedwater Injection		
	0-2 hours	2-8 hours	8-30 hours	0-2 hours	2-8 hours	8-30 hours
Locked Rotor	384,000 lbm	860,000 lbm	1,488,000 lbm	568,000 lbm	943,000 lbm	1,488,000 lbm
Steamline Break	381,000 lbm	830,000 lbm	1,488,000 lbm	519,000 lbm	892,000 lbm	1,488,000 lbm

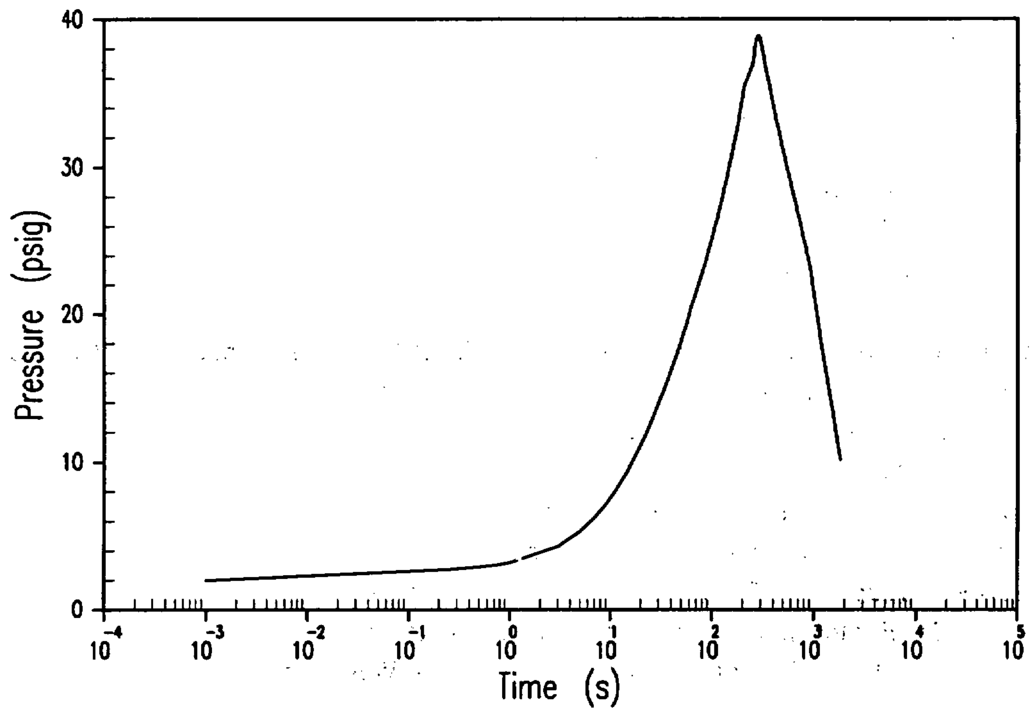

Figure 6.6-1

Containment Pressure Curve for Steamline Break for IP2

6.8 Anticipated Transients Without Scram

6.8.1 Introduction

For Westinghouse-designed pressurized water reactors (PWRs), the licensing requirements related to anticipated transients without scram (ATWS) are specified in the Final ATWS Rule, 10CFR50.62(c) (Reference 1). The requirement set forth in 10CFR50.62(c) is that all Westinghouse-designed PWRs must install AMSAC (ATWS Mitigation System Actuation Circuitry), and in compliance with this, AMSAC has been installed and implemented at Indian Point Unit 2 (IP2).

As documented in SECY-83-293 (Reference 2), the analytical bases for the Final ATWS Rule are the generic ATWS analyses for Westinghouse PWRs generated by Westinghouse in 1979. These generic ATWS analyses were formally transmitted to the Nuclear Regulatory Commission (NRC) via letter NS-TMA-2182 (Reference 3) and were performed based on the guidelines provided in NUREG-0460 (Reference 4).

In the generic ATWS analyses documented in NS-TMA-2182 (Reference 3), ATWS analyses were performed with the LOFTRAN computer code for the various American Nuclear Society (ANS) Condition II events (that is, anticipated transients), considering various Westinghouse PWR configurations applicable at that time. These analyses addressed two-, three-, and four-loop PWRs with various steam generator models. For IP2, the generic ATWS analyses applicable at that time were those for a four-loop PWR with Model 44 steam generators and a core power of 3025 MWt. These conditions are summarized in Table 3-1-d of NS-TMA-2182 (Reference 3). For this plant configuration, the peak Reactor Coolant System (RCS) pressure reported in NS-TMA-2182 for the limiting loss-of-load ATWS event is 2979 psia.

The generic ATWS analyses documented in NS-TMA-2182 (Reference 3) also support the analytical basis for the NRC-approved generic AMSAC designs generated for the Westinghouse Owners Group (WOG), as documented in WCAP-10858P-A, Revision 1 (Reference 5). For the purpose of these AMSAC designs, the generic ATWS analyses for the four-loop PWR configuration with Model 51 steam generators were used to conservatively represent all of the various Westinghouse PWR configurations contained in NS-TMA-2182. For IP2, WCAP-10858P-A, *AMSAC Logic 1, AMSAC Actuation on Low Steam Generator Water Level,* was used.

For the subject power uprating, an increase from a Nuclear Steam Supply System (NSSS) power of 3127 MWt to an NSSS power of 3230 MWt is proposed. This reflects a power increase of 6.8 percent above that considered in the generic ATWS analysis for the four-loop PWRs with Model 44 steam generators. As documented in NS-TMA-2182 (Reference 3), an increase in core thermal power adversely affects the results of the ATWS analyses. As reported for the generic four-loop PWR with Model 51 steam generators, an increase in power of 2 percent increases peak RCS pressure by 44 psi in the limiting loss-of-load ATWS. As demonstrated in NS-TMA-2182, the peak RCS pressure with the 2-percent increase in power remains below 3200 psig. This ATWS sensitivity analysis was performed assuming a 2-percent variation in power consistent with the typical calorimetric measurement uncertainty on power at the time of these analyses. Based on this sensitivity, the proposed increase in power of 6.8 percent would increase the RCS pressure in the limiting loss of load ATWS event by 150 psia.

As prescribed by NUREG-0460 (Reference 4), the 1979 generic ATWS analyses for Westinghouse PWRs documented in NS-TMA-2182 (Reference 3) assumed a full-power moderator temperature coefficient (MTC) of -8 pcm/°F. A sensitivity analysis including the use of an MTC of -7 pcm/°F was also provided as prescribed by NUREG-0460. In 1979, the MTC values of -8 pcm/°F and -7 pcm/°F represented MTCs that Westinghouse PWRs would be more negative than for 95 and 99 percent of the cycle, respectively. The base case of 95 percent represents a 95-percent confidence limit on favorable MTC for the fuel cycle. For IP2, the Technical Specification requirement on MTC is limited to < 0 pcm/°F at all power levels. The current MTC Technical Specification for IP2 remains the same as that which was applicable for most Westinghouse PWRs in 1979. Therefore, the reactivity feedback for IP2 remains sufficiently negative to be comparable to the generic Westinghouse ATWS analyses presented in NS-TMA-2182.

Relative to the other conditions important to the ATWS analyses, the pressurizer power-operated relief valve (PORV) relief capacity, safety valve relief capacity, and auxiliary feedwater (AFW) capacity is unaffected by the proposed stretch power uprate (SPU). The design capacity of each IP2 pressurizer PORV (179,000 lbm/hr) and pressurizer safety relief valve (408,000 lbm/hr) are consistent with the relief capacities assumed in the 1979 generic ATWS analysis for this plant configuration.

The design capacities of the IP2 AFW pumps are as follows.

- Motor-driven AFW pump - 400 gpm
- Turbine-driven AFW pump - 800 gpm

The IP2 Auxiliary Feedwater System (AFWS) has two motor-driven AFW pumps (each pump aligned to 2 steam generators) and a turbine-driven AFW pump that requires operator action to initiate flow to all 4 steam generators. Therefore, the total design capacity of the IP2 AFWS, originally designed for 1600-gpm flow, can only be credited for a total flow of 800 gpm. The reduced flow results in an overall peak pressure penalty when compared to the total AFWS

- AFW flow from the TDAFWP
- Maximum AFW temperature (that is, 120.0°F).

The limiting transient with respect to CST inventory is the LOAC to station auxiliaries transient. IP2 licensing basis dictates that in the event of a LOAC as described in subsection 6.3.8 of this report, sufficient CST inventory must be available to bring the unit from full power to hot standby conditions, and maintain the plant in hot standby for 24 hours. The SPU CST minimum useable volume requirement is 295,150 gallons.

9.12.3 Description of Analysis and Evaluation

Evaluation of the AFWS consists of documenting the current system functional requirements for transients/accidents and the extent to which SPU impacts these AFWS functions.

This evaluation compared AFWS component and equipment pressure and temperature design with the SPU pressure/temperature associated with AFWS operating conditions (that is, AFWS functions associated with normal plant startup and shutdown).

The evaluation also considered the extent to which sufficient AFW flow is provided to the steam generators following a design basis accident (DBA), and the extent to which adequate water inventory is available in the CST to satisfy AFWS functional requirements. The limiting transient (that is, design basis) with respect to CST inventory is LOAC as described in subsection 6.3.8 of this document.

9.12.4 Acceptance Criteria

The AFWS is considered acceptable under SPU conditions provided the following conditions are met:

- AFWS piping and component pressure and temperature design bounds pressure and temperature conditions under SPU off-normal operation (see subsection 9.12.1).

- Sufficient AFW flow is provided to the steam generators following a DBA and AFW pump operation is within acceptable margin of pump design parameters (for example, flow and total discharge head [TDH]).

- Based on the limiting transient (that is, LOAC as described in subsection 6.3.8 of this report) design basis, sufficient 120°F AFW inventory is available to maintain the IP2 plant in hot standby for 24 hours following a reactor trip from full power.

9.12.5 Results and Conclusions

The volume of water contained in the IP2 CST is adequate to support SPU.

AFWS component and equipment pressure and temperature design bounds maximum pressure and temperature conditions expected under SPU operation. AFWS components and equipment are considered acceptable for SPU operation.

The requirement to remove heat from steam generators under transient and accident conditions is the basis for AFWS minimum flow requirement. Currently, in the event of a loss of normal feedwater or a LOAC to station auxiliaries, a minimum flow of 380 gpm is assumed to 2 of the steam generators as a result of automatic actuation of 1 motor-driven AFW pump due to a low-low steam generator water level trip signal. Under SPU conditions an additional minimum AFW flow of 380 gpm, split evenly between the 2 remaining steam generators is required, and can be provided as a result of operator action taken to start the remaining motor-driven AFW pump or steam-driven AFW pump.

The AFWS is acceptable for operation under SPU conditions. No system modifications are required.