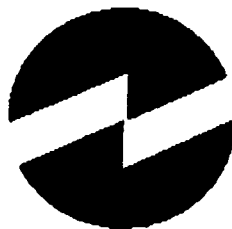


INDIAN POINT UNIT 3



**SAFETY EVALUATION
of
24-MONTH FUEL CYCLE PHASE 1 INSTRUMENT CHANNEL UNCERTAINTIES**

SECL-96-103

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WESTINGHOUSE

NUCLEAR SAFETY EVALUATION CHECKLIST

1. NUCLEAR PLANT: INDIAN POINT UNIT 3
2. SUBJECT: SAFETY EVALUATION of 24 MONTH FUEL CYCLE PHASE 1
INSTRUMENT UNCERTAINTIES
3. A written safety evaluation addressing the subject change or modification has been prepared based on the regulatory screening criteria of 10 CFR 50.59, and is attached. If the formal 10 CFR 50.59 safety evaluation was not required or is incomplete for any reason, an explanation is provided on Page 2.

Parts A and B of this Safety Evaluation Checklist are completed only on the basis of the attached written safety evaluation.

CHECKLIST - PART A - 10 CFR 50.59 (a) (1)

- (3.1) Yes No A change to the plant as described in the FSAR?
 (3.2) Yes No A change to procedures as described in the FSAR?
 (3.3) Yes No A test or experiment not described in the FSAR?
 (3.4) Yes No A change to the plant technical specifications
 (See **REMARKS** on Page 2)

4. CHECK LIST - PART B - 10 CFR 50.59 (a) (2) (Justification for Part B answers must be included on page 2.)

- (4.1) Yes No Will the probability of an accident previously evaluated in the FSAR be increased?
 (4.2) Yes No Will the consequences of an accident previously evaluated in the FSAR be increased?
 (4.3) Yes No May the possibility of an accident which is different than any already evaluated in the FSAR be created?
 (4.4) Yes No Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
 (4.5) Yes No Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
 (4.6) Yes No May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?
 (4.7) Yes No Will the margin of safety as defined in the bases to any technical specification be reduced?

If the answers to any of the questions on the previous page are unknown, explain below under REMARKS. If the answer to question 3.4 of Part A or any of the questions in Part B cannot be answered in the negative, based on written safety evaluation, the change review would require an application for license amendment as required by 10 CFR 50.59(c) and submitted to the NRC pursuant to the requirements of 10 CFR 50.90.

5. REMARKS

The answers given in Sections 3 and 4, Parts A and B of the Safety Evaluation Checklist are based on the attached safety evaluation.

The New York Power Authority is extending the Indian Point Unit 3 Technical Specification surveillance intervals for pressurizer pressure, accumulator pressure and level, and volume control tank level consistent with 24-month fuel cycles and the regulatory guidance of Nuclear Regulatory Commission Generic Letter 91-04⁽¹⁾. This change in surveillance intervals cannot be implemented without prior Nuclear Regulatory Commission approval since it represents a change to the Indian Point Unit 3 Technical Specifications, and 10CFR50.59 requires an application for amendment to the Indian Point Unit 3 license pursuant to the regulations of 10CFR50.90. Furthermore, a No Significant Hazards evaluation is required according the regulations of 10CFR50.92. This safety evaluation addresses the potential effects of changes to Indian Point Unit 3 safety analyses input assumptions that resulted from increased instrument channel uncertainties associated with extended surveillance (calibration) intervals. This report supports the requirement for a written safety evaluation, and explicitly addresses the regulatory screening criteria of both 10CFR50.59 (Section 4.0) and 10CFR50.92 (Section 5.0).

SAFETY EVALUATION APPROVALS:

Prepared by: R.R. Laubham Date: 6/18/96
 R.R. Laubham
 Operating Plant Licensing

Reviewed by: J.J. DeBlasio Date: 6/18/96
 J.J. DeBlasio
 Operating Plant Licensing

INDIAN POINT UNIT 3

SAFETY EVALUATION OF 24-MONTH FUEL CYCLE PHASE 1 INSTRUMENT CHANNEL UNCERTAINTIES

1.0 Background

Westinghouse is supporting the New York Power Authority (NYPA) in an effort to justify extending the Indian Point Unit 3 (IP3) Technical Specification surveillance intervals that had not been extended as part of a Power Authority program several years ago. This safety evaluation is part of a current effort to complete the IP3 surveillance interval extension program, and is essential to preclude a September 1996 surveillance outage.

The current surveillance extension effort is phased according to when IP3 surveillances come due. Phase 1 addresses the Power Authority's first priority to extend the earliest calibrations on pressurizer pressure, accumulator pressure and level, and Volume Control Tank (VCT) level which are currently due to be performed in September 1996.

The purpose of this safety evaluation is to address the effects of the extended technical specification surveillance intervals with the IP3 licensing basis accident analyses for Cycle 9. The Power Authority has defined the following specific changes to safety analyses input assumptions consistent with the IP3 extended surveillance interval program⁽²⁾:

- a) Uncertainty on initial condition pressurizer pressure of ± 40 psi.
 - All current Cycle 9 Safety Analyses Limits (SALs) based on pressurizer pressure uncertainties remain bounding for extended surveillance intervals (high and low pressure trips).
 - All current Cycle 9 Engineered Safety Feature Actuation System (ESFAS) trip settings based on pressurizer pressure uncertainty remain bounding (low pressure safety injection).
- b) Lower bound on initial accumulator pressure of 556 psia (541 psig).
- c) Range on accumulator volume from 762.5 ft³ to 827.5 ft³.
- d) The current VCT uncertainties remain bounding for extended surveillance intervals.
- e) The SAL K1 values for Cycle 9 remain applicable. K1 is a constant used in the Technical Specification OverTemperature ΔT trip setpoint.

Therefore, the only safety analyses input changes stemming from changes to instrument channel uncertainties (due to increased instrument drift associated with extended surveillance intervals) are to the initial condition pressurizer pressure, and accumulator pressure and level. As noted in item (d), VCT level uncertainty does not change

The evaluations and conclusions addressed by this safety evaluation represent the result of individual reviews performed by Westinghouse in several areas. The areas not affected by the subject safety analyses input changes include, Instrumentation and Control, Systems and Components, Steam Generator Tube Rupture (SGTR) Accident Analysis, Radiological Assessment, and Emergency Operating Procedures (EOPs). The LOCA, Non-LOCA, and Containment Integrity areas were determined to be potentially affected by the subject conditions, and the scope of this safety evaluation was limited to these areas. The Power Authority will identify the IP3 Technical Specification changes necessary to reflect the surveillance interval extensions for the subject parameters.

Westinghouse concludes that the IP3 licensing basis safety analyses remain limiting with the changes to the safety analyses inputs as described on the previous page and, therefore, the subject changes do not adversely affect safe plant operation.

2.0 Licensing Basis Acceptance Criteria

This evaluation was performed according to the regulations set forth in Title 10 of the Code of Federal Regulations, Part 50, (10 CFR 50.59). This regulation allows the holder of a license authorizing operation of a nuclear power facility the capacity to evaluate changes to the plant and/or procedures, and tests or experiments not described in the Final Safety Analysis Report (FSAR)⁽⁹⁾. Furthermore, prior NRC approval is not required to implement a change provided that it does not involve an unreviewed safety question or result in a change to plant technical specifications. However, Technical Specification surveillance interval extensions, such as those described herein, will necessitate certain Technical Specification changes, and cannot be implemented without prior NRC approval. 10CFR50.59 requires an application for amendment to the IP3 license pursuant to the regulations of 10CFR50.90. Furthermore, a No Significant Hazards evaluation is required according the regulations of 10CFR50.92. This report supports the regulatory requirement for a written safety evaluation, and explicitly addresses the regulatory screening criteria of both 10CFR50.59 (Section 4.0) and 10CFR50.92 (Section 5.0).

The determinations by this safety evaluation that the subject changes to the IP3 safety analyses input assumptions do not involve an unreviewed safety question, and do not involve significant hazards consideration were made based on individual evaluations performed according to pertinent licensing-basis acceptance criteria for IP3. This was accomplished as follows:

2-1 The Non-LOCA safety analyses evaluation (Section 3.1) demonstrates that:

- a. The minimum DNBR will not violate applicable minimum limit values.
- b. The maximum pressurizer water volume will not violate the current limit of 1849.4 ft³.
- c. The maximum reactor coolant system and main steam system peak pressures will not violate current limits values of 2733.5 psig and 1193.5 psig respectively.

2-2 The LOCA analyses safety evaluation (Section 3.2) demonstrates compliance with the Peak Clad Temperature (PCT) limit of 2200°F as specified in 10CFR50.46 b(1), and was performed consistent with the requirements of 10CFR50, Appendix K. The evaluation also ensures compliance with other 10CFR50.46 criteria paraphrased as follows:

- a. The total cladding oxidation must be less than 17% of the total cladding thickness prior to oxidation.
- b. The total hydrogen generated must be less than 1% of the hypothetical amount that would be generated if all the cladding were to react with water or steam.
- c. The core must remain amenable to cooling.
- d. The core temperature must be maintained acceptably low, and decay heat must be removed for the period of time required by the long-lived radioactivity remaining in the core.

2-3 The Containment Integrity accident analyses safety evaluation (Section 3.3) demonstrates that the peak calculated containment pressure will remain less than both the containment design value of 47 psig as identified in SECL-92-131⁽⁴⁾, and the ILRT pressure of 42.42 psig as specified in the IP3 Technical Specifications (Section 4.4-A.1.a). SECL-92-131⁽⁴⁾ documents the current licensing basis containment analysis of record. The 47 psig limit was most recently used as an acceptance criteria by the NRC in their SER addressing the Containment Margin Improvement and Ultimate Heat Sink Programs⁽⁵⁾. This safety evaluation also demonstrates that the peak containment temperature is unaffected and will remain less than the IP3 Equipment Qualification (EQ) envelope temperature of 290°F⁽⁶⁾.

3.0 Evaluations

3.1 Non-LOCA Transient Analyses Evaluations

The IP3 FSAR⁽⁹⁾ includes analyses and evaluations for the following pertinent Non-LOCA transient events:

<u>FSAR Section</u>	<u>Event Title</u>
14.1.1	Uncontrolled Control Rod Withdrawal from a Subcritical Condition
14.1.2	Uncontrolled Control Rod Assembly Withdrawal at Power
14.1.3	Rod Assembly Misalignment
14.1.4	Rod Cluster Control Assembly (RCCA) Drop
14.1.5	Chemical and Volume Control System Malfunction
14.1.6	Loss of Reactor Coolant Flow
14.1.7	Startup of an Inactive Reactor Coolant Loop
14.1.8	Loss of External Electrical Load
14.1.9	Loss of Normal Feedwater
14.1.10	Excessive Heat Removal Due to Feedwater System Malfunctions
14.1.11	Excessive Load Increase Incident
14.1.12	Loss of All AC Power to the Station Auxiliaries
14.2.5	Rupture of a Steam Pipe
14.2.6	Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)

The current analyses of record for these events already conservatively incorporate a ± 40 psi uncertainty on pressurizer pressure.

The only non-LOCA transient analyses that model the passive accumulators are the steamline break core response analysis and subsequent evaluations. These analyses assume a minimum initial accumulator gas pressure of 613.8 psia at 120°F. This is non-conservative with respect to extended surveillance interval value of 556 psia, however; the passive accumulators do not actuate for any of the IP3 ^{non-LOCA} analyses. Therefore, the proposed reduction in the minimum initial accumulator pressure to 556 psia is acceptable with respect to the IP3 non-LOCA transient analyses for Cycle 9.

Lastly, the same non-LOCA transient analyses incorporate a minimum available accumulator volume of 707.5 ft³. Again, the accumulators do not actuate for any of the IP3 Cycle 9 licensing basis non-LOCA transient analyses and, in any case, the proposed minimum available accumulator volume of 762.5 ft³ is conservatively bounded by the 707.5 ft³ used in the current Cycle 9 analyses.

3.2 LOCA Analyses Evaluation

This evaluation was performed to determine the effects of the identified changes to the initial condition pressurizer pressure, accumulator pressure and accumulator volume on the IP3 LOCA-related analyses. The IP3 LOCA analyses are discussed in FSAR⁽³⁾ Chapter 14.3.

3.2.1 Large Break LOCA - FSAR Chapter 14.3

The large break LOCA (LBLOCA) analysis which serves as the licensing basis for IP3 was performed using the 1981 Evaluation model with BASH. This analysis was performed assuming a core power level of 102% of 3025 MWt with a total peaking factor (F_Q) of 2.32, a hot channel enthalpy rise factor ($F_{\Delta H}$) of 1.62, a thermal design flow of 323,600 gpm, and an initial system pressure of 2265 psig (2280 psia). This analysis resulted in a peak cladding temperature (PCT) of 1893°F for the $C_D=0.4$ break case. This analysis has been supplemented by safety evaluations which have increased the resultant PCT to 1975°F⁽⁶⁾.

The LBLOCA analysis described above was based upon the following input assumptions:

Accumulator Pressure:	600 psia
Accumulator Volume:	787.5 ft ³ (nominal based upon 775 - 800 ft ³ range)
Initial RCS Pressure:	2280 psia (2250 psia plus 30 psi uncertainty)

An evaluation was previously performed for IP3 to increase the maximum accumulator volume from 800 ft³ to 815 ft³. That evaluation⁽⁷⁾ resulted in a 15°F increase in the LBLOCA PCT.

As a result of this current evaluation of changing the minimum accumulator pressure to 556 psia, the accumulator volume range to 762.5 ft³- 827.5 ft³, and the initial RCS pressure to 2290 psia, Westinghouse determined that the total PCT must increase by a conservative amount of 34°F. However, while this penalty increases the total PCT for the LBLOCA to 2009°F, it is still well below the regulatory criteria of 2200°F.

3.2.2 Small Break LOCA - FSAR Chapter 14.3

The current small break LOCA (SBLOCA) licensing basis analysis for IP3 was performed with the NRC-approved Evaluation Model using the NOTRUMP code. This analysis was performed assuming a core power level of 102% of 3025 MWt with an F_Q of 2.42 and an $F_{\Delta H}$ of 1.62, a thermal design flow of 323,600 gpm, and an initial primary system pressure of 2265 psig (2280 psia). This resulted in a PCT of 1407°F for the limiting break of three inches and has been supplemented by evaluations, changing the resultant PCT to 1321°F⁽⁶⁾.

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The SBLOCA analysis described above was based upon the following input assumptions:

Accumulator Pressure: 600 psia
Accumulator Volume: 795 ft³ (nominal based upon 775 - 815 ft³ range)
Initial RCS Pressure: 2280 psia (2250 psia plus 30 psi uncertainty)

As a result of this current evaluation of changing the minimum accumulator pressure to 556 psia, the accumulator volume range to 762.5 ft³- 827.5 ft³, and the initial RCS pressure to 2290 psia, Westinghouse determined that the total PCT must increase by a conservative amount of 5°F. However, while this penalty increases the total PCT for the LBLOCA to 1326°F, it is still well below the regulatory criteria of 2200°F.

3.2.3 Blowdown Reactor Vessel and Loop Forces - FSAR Chapter 14.3.4

LOCA hydraulic forcing function analyses performed by Westinghouse are dependent upon RCS geometry, RCS fluid density (defined by RCS pressure and temperatures), and break size. The primary system pressure supported by the current IP3 LOCA forces analysis is 2290 psia (2250 psia plus 40 psi uncertainty). Since this accounts exactly for the extended surveillance interval uncertainty specified by the Power Authority⁽⁹⁾, the licensing basis LOCA forcing function analysis is unaffected.

The accumulators are not modeled in the IP3 licensing basis LOCA forcing function analysis, so changes to the values of accumulator pressure and volume do not affect them.

3.2.4 Post LOCA Long Term Core Cooling - Westinghouse Licensing Position FSAR Chapter 14.3

The Westinghouse methodology for satisfying the requirements of 10 CFR Part 50 Section 50.46 Paragraph (b) Item (5) "Long Term Cooling" is comprised of a calculation of the post-LOCA recirculation sump boron concentration. Since credit for the control rods is not taken for LBLOCA, the borated ECCS water provided by the accumulators and the refueling water storage tank (RWST) must have a concentration that, when mixed with other sources of borated and non-borated water, will result in the reactor core remaining subcritical assuming all control rods out.

The inputs for this calculation are the boron concentrations of the reactor coolant system (RCS) and the ECCS, the associated water volumes, and the initial RCS operating soluble boron concentration. In the calculation performed for IP3, an accumulator volume of 775 ft³ is modeled. Westinghouse recalculated the sump boron concentration using the revised minimum accumulator volume of 762.5 ft³. Although the change in accumulator minimum volume changed the total system boron mass and solution mass, the effect on the sump boron concentration is negligible. This is because the 12.5 ft³ decrease in the minimum volume is inconsequential compared to the combined mass of the RCS, RWST, and other system components. As such, the change in accumulator volume does not adversely affect this calculation.

The accumulator pressure is not modeled in this calculation. Although the change in initial primary system pressure may change the density and thus the initial RCS mass, the effect on the sump boron concentration is negligible. As such, the change in accumulator pressure and initial primary system pressure does not adversely affect this calculation.

3.2.5 Hot Leg Switchover to Prevent Potential Boron Precipitation - FSAR Chapter 14.3.2.1

The post-LOCA hot leg switchover time is determined for inclusion in plant EOPs and is calculated to preclude boron precipitation in the reactor vessel following the initiation of boiling in the core. The time at which switchover occurs is dependent upon the core power history, as well as the RCS, RWST, and accumulator water volumes and boron concentrations. This input is similar to that used in the calculation of the sump boron concentration except that the boron concentrations are assumed to be at a maximum.

As with the sump boron calculation, Westinghouse recalculated the hot leg switchover time using the revised accumulator volume. For this calculation, the maximum volume of 827.5 ft³ was modeled. Westinghouse determined that this change from the 815 ft³ volume used in the current licensing basis calculation has a negligible effect on the hot leg switchover time. This is also due to the fact that the 12.5 ft³ increase in volume is inconsequential compared to the combined mass of the RCS, RWST, and other system components. Therefore, the increase in accumulator volume does not adversely affect this calculation.

Again, although the change in initial primary side pressure may change the density and thus the initial RCS mass, the effect on the system boron concentration is negligible. The accumulator pressure is also not modeled in this calculation. As such, the change in RCS initial pressure and accumulator pressure does not adversely affect this calculation.

3.2.6 LOCA Conclusion

Westinghouse evaluated the potential effects of identified changes to the initial pressurizer pressure, accumulator pressure, and accumulator volume on the licensing basis LOCA-related accident analyses, and determined that none were adversely affected and all continue to meet the pertinent criteria of 10 CFR 50.46.

3.3 Containment Integrity Analyses Evaluation

The current licensing basis subcompartment and containment integrity analyses are discussed in Section 14.3.6 of the IP3 FSAR⁽⁶⁾. The Containment Subcompartment Analysis is performed to demonstrate the integrity of containment internal structures when subjected to dynamic, localized pressurization effects that could occur during the very early time period following a design basis LOCA. The Containment Integrity Analyses is performed to demonstrate that the containment, containment structures, and containment cooling systems are adequate to mitigate the consequences of a hypothetical large-break LOCA such that resultant containment pressures do not exceed the containment design pressure of 47 psig, and less than the Integrated Leak Rate Test (ILRT) pressure of 42.42 psig as specified in the IP3 Technical Specifications (Section 4.4-A.1.a).

This evaluation assessed the effects of changes to safety analyses input assumptions defined in Section 1.0 consistent with the IP3 extended surveillance interval program.

3.3.1 Short Term Loca Mass and Energy Releases / Subcompartment Pressure Analyses

Containment subcompartment analyses are performed to demonstrate the adequacy of containment internal structures and attachments when subjected to dynamic localized pressurization effects. The short pressure pulse which accompanies a high energy line pipe rupture within a subcompartment is over within a few seconds.

The NRC has already approved Leak-Before-Break (LBB) for IP3. LBB eliminates the need to consider dynamic effects of postulated primary loop pipe ruptures (i.e. larger break sizes) from the mechanical design basis for primary loop piping and supports. Therefore, the decrease in mass and energy releases associated with the smaller RCS nozzle breaks, as compared to the larger RCS pipe breaks, more than offsets the potential penalties associated with the increased releases associated with the changes defined to support the extended surveillance interval program. The current licensing basis subcompartment analyses that consider breaks in the RCS remain bounding.

3.3.2 Long Term Mass and Energy Release / Containment Response Analyses

The LOCA long-term analysis is performed to demonstrate the ability of the containment systems to mitigate the consequences of a hypothetical LOCA. The long term mass and energy release and containment pressure response calculations following a LOCA consider the effects of long term depressurization and secondary side heat transfer. The analyses consider the total energy available to the containment from both the primary and secondary side sources at all particular time segments of the transient.

Based upon the results of this assessment, there is a 0.1 psi effect on the peak calculated containment pressure for a postulated LOCA. The current FSAR licensing basis containment integrity analysis addressed the pressurizer pressure uncertainty to +/- 40 psi as uncertainty on the initial pressure condition, therefore, the current licensing basis remains valid with respect to this parameter. With respect to the VCT level and SAL K1 values, neither are credited in mitigating nor worsening the consequence of a LOCA mass and energy release. Therefore, these do not affect the analyses. Accumulator performance reflecting accumulator initial gas pressure and volume were explicitly modeled in the LOCA long term mass and energy release containment integrity analysis. The accumulators are modeled to conservatively bias (maximize) the effect on the mass and energy release. An evaluation of the changes for the initial accumulator pressure and volume was performed. The evaluation showed that the values assumed in the licensing basis analysis do not bound the proposed changes. Consequently, due the change of the lower bound on initial accumulator pressure (reduced from 600 psia to 556 psia), and of the maximum on accumulator water volume (increase in maximum level from 815 ft³ to 827.5 ft³) a potential containment pressure increase of approximately 0.1 psi could be attained. This raises the licensing basis peak pressure for this event from its current value of 42.29 psig to a new value of approximately 42.39 psig. This is bounded by the current Technical Specification ILRT value of 42.42 psig.

3.3.3 Containment Response Following a Main Steamline Break Release Inside Containment

Containment response calculations for a postulated main steamline break (MSLB) release inside containment are performed to ensure that the containment pressure and temperature do not exceed acceptable levels. Based upon the conclusions of the evaluation for MSLB mass and energy release calculations, there would be no change in the mass and energy release to the containment due to the subject changes to safety analyses input parameter values pertaining to the Technical Specification surveillance intervals. Therefore, the containment response calculations for the current licensing basis analysis remain valid.

3.3.4 Peak Sump Temperature

The peak sump temperature calculation is not an explicit Chapter 14 safety analysis. However, the results are input for the Ultimate Heat Sink Analysis⁽⁶⁾. Based upon the magnitude of the estimated effect (+0.1 psi adder on peak pressure) and the fact that the accumulator would be adding cold water (130°F) relative to the temperature of the RCS, Westinghouse does not expect any significant effect due to the changes to the initial accumulator conditions. The value of the calculated peak sump temperature remains at 255.8°F.

3.3.5 Summary and Conclusions

The effects of the subject changes to pertinent safety analyses input assumptions pertaining to the IP3 Technical Specification surveillance interval extension project have been evaluated. The cumulative effect of the items evaluated on the FSAR licensing basis could result in a possible increase in the containment peak calculated pressure to 42.39 psig. This peak is bounded by both the design pressure is 47 psig, and the Technical Specification ILRT pressure is 42.42 psig. The calculated peak containment temperature is a result of the containment response following a MSLB and, since the current FSAR analysis remains valid, the containment temperature is not affected.

4.0 Assessment of No Unreviewed Safety Questions

The safety significance of the subject changes to certain IP3 safety analyses input parameter values has been evaluated as required according to the criteria of 10CFR50.59, and does not represent an unreviewed safety question on the basis of the following responses to specific related questions.

4.1 Will the probability of an accident previously evaluated in the FSAR be increased?

No. Westinghouse has demonstrated that all pertinent licensing-basis acceptance criteria have been met. The subject changes to safety analyses input parameter values do not affect any of the mechanisms postulated in the IP3 FSAR to cause licensing basis events. Therefore the probability of an accident previously analyzed in the IP3 FSAR will not be increased.

4.2 Will the consequences of an accident previously evaluated in the FSAR be increased?

No. Westinghouse has demonstrated that all pertinent licensing-basis acceptance criteria have been met. Since all pertinent acceptance criteria continue to be met and the integrity of the reactor coolant system pressure boundary is not challenged, the assumptions employed in the calculation of the offsite radiological doses remain valid, and the consequences of the accidents considered in the IP3 licensing basis remain unchanged. Therefore, the consequences of an accident previously evaluated in the IP3 FSAR will not be increased.

4.3 May the possibility of an accident which is different than any already evaluated in the FSAR be created?

No. Westinghouse has demonstrated that all pertinent licensing-basis acceptance criteria have been met. The subject changes to safety analyses input parameter values neither reflect changes to the plant configuration, nor introduce new potential hazards to the plant (i.e. no new failure modes have been created). Therefore, the possibility of an accident which is different than any already evaluated in the IP3 FSAR is not created.

4.4 Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

No. Westinghouse has demonstrated that all pertinent licensing-basis acceptance criteria have been met. Therefore, the probability of a malfunction of equipment important to safety previously evaluated in the IP3 FSAR will not be increased.

4.5 Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

No. Westinghouse has demonstrated that all pertinent licensing-basis acceptance criteria have been met, and current failure modes as analyzed are unchanged. With no change to the analyzed failure modes, the FSAR acceptance criteria for the postulated design basis events remain satisfied. The subject changes to safety analyses input parameter values are not related to the ability of existing components and systems or the integrity of the fission product barriers to mitigate the radiological dose consequences of any accident. Therefore, the consequences of a malfunction of equipment important to safety previously evaluated in the IP3 FSAR will not be increased.

4.6 May the possibility of a malfunction of equipment important to safety different than already evaluated in the FSAR be created?

No. Westinghouse has demonstrated that all pertinent licensing-basis acceptance criteria have been met. No new single failure mechanisms have been introduced, nor will the core operate in excess of pertinent design basis operating limits due to the implementation of the subject changes to safety analyses input parameter values. Therefore, the possibility of a malfunction of equipment important to safety that is different than already evaluated in the IP3 FSAR has not been created.

4.7 Will the margin of safety as defined in the bases to any technical specifications be reduced?

No. Westinghouse has demonstrated that all pertinent licensing-basis acceptance criteria have been met, and the margin of safety as defined in the Technical Specification Bases is not reduced in any of the IP3 licensing basis accident analyses. Therefore, the same margins exists to all pertinent design failure points or system limitations.

Meeting the licensing basis acceptance criteria, as identified in Section 2.0, ensures that there will be no degradation in the margins to safety to pertinent design failure points. For example, demonstrating that the 2200°F PCT limit is met ensures that the margin to safety to fuel melt has not been reduced. Therefore, margin of safety as defined in the bases to any technical specification will not be reduced.

5.0 Assessment of No Significant Hazards Consideration

The safety significance of the subject changes to certain IP3 safety analyses input parameter values has been evaluated as required according to the criteria of 10CFR50.92, and does not involve Significant Hazards Consideration on the basis of the following responses to specific related questions.

5.1 Will the probability or consequences of an accident previously evaluated be increased significantly?

No. Westinghouse has demonstrated that all pertinent licensing-basis acceptance criteria have been met, and the margin of safety as defined in the Technical Specification Bases is not reduced in any of the IP3 licensing basis accident analyses based on the subject changes to safety analyses input parameter values. Therefore, the probability of an accident previously evaluated has not increased. Because design limitations continue to be met, and the integrity of the reactor coolant system pressure boundary is not challenged, the assumptions employed in the calculation of the offsite radiological doses remain valid. Therefore, the consequences of an accident previously evaluated will not be increased.

5.2 Will the possibility of a new or different kind of accident from any accident previously evaluated be created?

No. Westinghouse has demonstrated that all pertinent licensing-basis acceptance criteria have been met, and the margin of safety as defined in the Technical Specification Bases is not reduced in any of the IP3 licensing basis accident analyses based on the subject changes to safety analyses input parameter values. Therefore, an accident which is different than any previously evaluated will not be created.

5.3 Will a significant reduction in a margin of safety be involved?

No. Westinghouse has demonstrated that all pertinent licensing-basis acceptance criteria have been met, and the margin of safety as defined in the Technical Specification Bases is not reduced in any of the IP3 licensing basis accident analyses based on the subject changes to safety analyses input parameter values. Since the evaluations in Section 3.0 demonstrate that all applicable acceptance criteria continue to be met, the subject operating conditions will not involve a significant reduction in margin of safety at IP3.

6.0 Conclusions

Westinghouse has determined that the subject changes to the IP3 safety analyses input parameter values for pressurizer pressure, accumulator pressure and volume, and VCT level will not adversely affect the IP3 licensing basis analyses for Cycle 9 and, therefore, will not compromise plant safety. The subject changes do not introduce any Unreviewed Safety Questions according to 10CFR50.59, and do not introduce Significant Hazards Consideration according to 10CFR50.92. However, IP3 Technical Specification changes are involved as described earlier.

7.0 References

1. NRC Generic Letter 91-04, "Changes to Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle (Generic Letter 91-04)", April 2, 1991, James G. Partlow, Associate Director for Projects, NRC Office of Nuclear Reactor Regulation.
2. New York Power Authority letter IP3-PM-96-016, "Surveillance Interval Extension Project - Potential Effects of Instrumentation Uncertainty Changes on Licensing Basis Safety Analyses", 6/13/96, D. Shih to R.R. Laubham.
3. Indian Point Unit 3 Final Safety Analysis Report.
4. SECL-92-131, "High Head Safety Injection Flow Changes Safety Evaluation", June 1992, R.R. Laubham (Issued to NYPA by letter INT-92-592 on 6/26/92).
5. NRC Safety Evaluation Report Related to Amendment No. 98 (WCAP-12313 on Ultimate Heat Sink Temperature Increases to 95°F at Indian Point Unit 3 and WCAP-12269 on Containment Margin Improvement at Indian Point Unit 3), 5/7/90, J.S. Guo (NRC principal contributor).
6. INT-96-201, "10CFR50.46 Annual Notification and Reporting", 2/13/96, M.J. Proviano to Leslie Hill.
7. INT-91-590, "Minimum RWST Boron Concentration and Related Changes", December 1991, J. Gasperini.

ATTACHMENT IV TO IPN-96-067

COMMITMENTS MADE IN PROPOSED TECHNICAL SPECIFICATION CHANGE

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

List of Commitments

Number	Commitment	Due
IPN-96-067-01	The Pressurizer Pressure, Accumulator Pressure and Level, and Volume Control Tank Level instrumentation will be included in the drift monitoring program by the next refueling outage.	Next refueling outage