March 22, 2005

Mr. Michael Kansler, President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 - ISSUANCE OF AMENDMENT RE: FULL SCOPE ADOPTION OF ALTERNATIVE SOURCE TERM (TAC NO. MC3351)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 224 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3 (IP3). The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated June 2, 2004, as supplemented on December 8, 15, and 22, 2004, and January 5 and 28, February 11 and 22, and March 14, 2005.

The amendment revises the TSs to fully adopt the alternative source term methodology for design-basis accident dose consequence evaluations in accordance with Section 50.67 of Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.67). In particular, the amendment revises the TS Definition regarding dose equivalent iodine and TS Section 5.5.10, “Ventilation Filter Testing Program (VFTP)” to remove filter testing requirements for the Fuel Storage Building Emergency Ventilation System and the Containment Purge System. The amendment also revises TS surveillance requirement (SR) 3.3.7 to adopt the makeup flow rate associated with the modification to the control room emergency mode of operation and SR 3.7.11 to incorporate editorial changes regarding the terminology for the operating modes for the control room ventilation system.

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

Sincerely,

/RA/

Patrick D. Milano, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-286

Enclosures: 1. Amendment No. 224 to DPR-64
2. Safety Evaluation

cc w/encls: See next page

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440 Hamilton Avenue
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cc w/encls: See next page

Accession Number: ML050750431

OFFICE	PDI-1\PM	PDI-1\LA	EMCB\SC	SPSB/SC	IROB\SC	OGC	PDI-1\SC
NAME	PMilano	SLittle	LLund	RDennig	TBoyce	SBrock	RLaufer
DATE	03/21/05	03/21/05	SE dtd 01/31/05	SE dtd 3/15/05	03/21/05	03/21/05	03/22/05

Official Record Copy

DATED: March 22, 2005

AMENDMENT NO. 224 TO FACILITY OPERATING LICENSE NO. DPR-64 INDIAN POINT UNIT 3

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ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 224
License No. DPR-64

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Nuclear Operations, Inc. (the licensee) dated June 2, 2004, as supplemented by letters dated December 8, 15, and 22, 2004, and January 5 and 28, February 11 and 22, and March 14, 2005, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-64 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 224, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard J. Laufer, Chief, Section I
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 22, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 224

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contains marginal lines indicating the areas of change.

Remove Pages	Insert Pages
1.1-3	1.1-3
3.3.7-1	3.3.7-1
3.7.11-2	3.7.11-2
5.0-21	5.0-21
5.0-22	5.0-22
5.0-23	5.0-23
5.0-24	5.0-24
5.0-25	5.0-25

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 224 TO FACILITY OPERATING LICENSE NO. DPR-64

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

1.0 INTRODUCTION

By application dated June 2, 2004 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML041600619), as supplemented on December 8, 15, and 22, 2004, and January 5 and 28, February 11 and 22, and March 14, 2005 (ADAMS Accession Nos. ML043490159, ML043630313, ML050030060, ML050120141, ML050390514, ML050450202, and ML050700096), Entergy Nuclear Operations, Inc. (the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 3 (IP3) Technical Specifications (TSs). The requested changes would revise the TSs to fully adopt the alternative source term (AST) methodology for design-basis accident (DBA) dose consequence evaluations in accordance with Section 50.67 of Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.67). Specifically, the amendment would revise the TS Section 1.1 “Definition” regarding dose equivalent iodine and would revise TS Section 5.5.10, “Ventilation Filter Testing Program (VFTP)” to remove filter testing requirements for the fuel storage building emergency ventilation system and the containment purge system. The amendment would also revise TS surveillance requirement (SR) 3.3.7 to adopt the makeup flow rate associated with the modification to the control room emergency mode of operation and SR 3.7.11 to incorporate editorial changes regarding the terminology for the operating modes for the control room ventilation system.

The AST methodology for the fuel-handling accident (FHA) was previously approved in Amendment No. 215, dated March 17, 2003 (ADAMS No. ML030760135). However, the licensee has revised the FHA analysis to no longer take credit for the lower fission product gap fractions in Regulatory Guide (RG) 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” and to reflect changes in the flow rates and allowable inleakage for the control room ventilation system.

The December 8, 15, and 22, 2004, and January 5 and 28, February 11 and 22, and March 14, 2005, supplements provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff’s original proposed no significant hazards consideration determination as published in the *Federal Register* on August 31, 2004 (69 FR 53104).

2.0 REGULATORY EVALUATION

The NRC staff addressed the impact of the proposed changes on previously analyzed DBA radiological consequences and the acceptability of the revised analysis results. The regulatory requirements on which the staff based its acceptance are the accident dose criteria in 10 CFR 50.67 and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19, "Control room." Except where the licensee proposed a suitable alternative, the NRC staff utilized the regulatory guidance in Regulatory Position 4.4 of RG 1.183, and in Section 6.4, "Habitability Systems," and Section 15.0.1, "Radiological Consequence Analysis Using Alternative Source Terms," of NUREG-800, "Standard Review Plan," (SRP) in performing this review. In addition, the staff utilized GDC 60, "Control of releases of radioactive materials to the environment," GDC 61, "Fuel storage and handling and radioactivity control," and GDC 64, "Monitoring radioactivity releases," to further assess the proposed changes to the VFTP.

3.0 TECHNICAL EVALUATION

3.1 Proposed Changes to TSs

In its June 2, 2004, application, the licensee proposed changes in Sections 1.1 and 5.5 of the current IP3 TSs as follows:

Section 1.1 - Definition of Dose Equivalent Iodine

The IP3 TS Definition of DOSE EQUIVALENT I-131 currently states:

> DOSE EQUIVALENT I -131 shall be that concentration of I -131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I -131, I -132, I -133, I -134 and I -135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table IIII of TID-14844, AEC, 1962, 'Calculation of Distance Factors for Power and Test Reactor Sites,' or those listed in Table E-7 of Regulatory Guide 1.109, Rev 1, NRC, 1977, or ICRP 30, Supplement to Part I, page 192-212, Table titled, 'Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity.'

The licensee proposed to revise the Definition to state:

> DOSE EQUIVALENT I-131 shall be that amount of I -131 (curies) that alone would produce the same committed effective dose equivalent (CEDE) dose as the quantity and isotopic mixture of I -130, I -131, I -132, I -133, I -134 and I -135 actually present. The CEDE dose conversion factors used for this calculation shall be those listed in Table 2.1 of Environmental Protection Agency Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," 1988.

In a letter dated March 14, 2004, the licensee revised the proposed Definition to remove I-130 from the isotropic mixture as stated above.

Section 5.5 - Ventilation Filter Testing Program

The licensee proposed to revise TS Section 5.5.10, "Ventilation Filter Testing Program (VFTP)," to delete the phrases "Fuel Storage Building Emergency Ventilation System" and "and Containment Purge System" from the first paragraph. Secondly, the reference to the Fuel Storage Building Emergency Ventilation System in Item (2) would be removed. Finally, Item (3) ("every 18 months for the Containment Purge System") would be deleted entirely.

TS Section 5.5.10, Table a, includes removal efficiencies for high-efficiency particulate (HEPA) filters in four ventilation systems. The removal efficiency requirements for (1) the fuel storage building emergency ventilation system and (2) the containment purge system HEPA filters would be removed.

TS Section 5.5.10, Table b, includes removal efficiencies for charcoal adsorbers in four ventilation systems. The removal efficiency requirements for (1) the fuel storage building emergency ventilation system and (2) the containment purge system charcoal adsorbers would be removed.

TS Section 5.5.10, Table c, includes methyl iodide removal efficiencies for charcoal adsorbers in four ventilation systems. The methyl iodide removal efficiency requirements for (1) the fuel storage building emergency ventilation system and (2) the containment purge system charcoal adsorbers would be removed.

TS Section 5.5.10, Table d, includes pressure drop testing for filtration trains in three systems. This requirement for the fuel storage building emergency ventilation system is being removed.

TS SRs 3.3.7 and 3.7.11

The licensee proposed to revise the SRs in TS 3.3.7, “Control Room Ventilation System (CRVS) Actuation System,” and TS 3.7.11, “Control Room Ventilation System (CRVS).” Specifically, SR 3.3.7.A.1, 3.3.7.B.1, and 3.7.11.4 would be revised to change the terminology for the CRVS Mode description to be consistent with the modifications being proposed for the CRVS. In addition, the makeup flow specified in SR 3.7.11.4 would be changed as part of the modification to the CRVS and to be consistent with the licensee’s analyses.

3.2 Background

The licensee’s request was based on a reanalysis of the radiological consequences of accidents for IP3 using the new source term methodology from NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” and applying the criteria of 10 CFR 50.67, “Accident Source Term.” The request includes application of the new source term methodology in evaluating radiological consequences of the following accidents: (1) large-break loss-of-coolant accident (LOCA); (2) FHA; (3) locked rotor accident; (4) rod ejection accident; (5) small-break LOCA; (6) main steam line break (MSLB); and (7) steam generator tube rupture (STGR). The radiological consequences were determined in terms of radiation doses (Total Effective Dose Equivalent, TEDE) at the boundary of the exclusion area, which is also the site boundary, at the low population zone outer boundary, and in the control room (CR).

3.3 Post-LOCA Removal of Iodine from Containment Atmosphere

3.3.1 Containment Sump pH

A variety of acids and bases are produced in containment after a LOCA. The pH value of the containment sump will depend on the chemical species dissolved in the containment sump water. The following chemical species are introduced into the containment sump in a post-LOCA environment: hydriodic acid (HI), nitric acid (HNO_3), hydrochloric acid (HCl) and cesium hydroxide (CsOH). CsOH and HI enter the containment directly from the reactor coolant system (RCS). HCl is produced by radiolytic decomposition of cable jacketing and HNO_3 is synthesized in the radiation field existing in the containment. The resultant containment sump pH will depend on their relative concentrations and on the buffering action of NaOH and boric acid.

Maintaining sump water in an alkaline condition is needed for preventing dissolved radioactive iodine from being released to the containment atmosphere during the recirculation containment spray injection. Most of the iodine leaves the damaged core in an ionic form which is readily dissolved in the sump water. However, in an acidic environment, some of it becomes converted into elemental form which is much less soluble, causing re-evolution of iodine to the containment atmosphere. Per NUREG-1465, the iodine entering the containment is at least 95% cesium iodide (CsI) with the remaining 5% as elemental and organic iodide plus hydriodic acid, with not less than 1% of each as iodine and hydriodic acid. In order to prevent release of elemental iodine to the containment atmosphere after a LOCA, the sump pH has to be maintained equal or higher than 7.

After a LOCA, the containment sump is mostly filled with water coming from the systems containing boric acid: Refueling Water Storage Tank (RWST), four accumulators, and the RCS. This in effect will cause the sump water pH to become acidic. In order to keep the pH above 7, NaOH is used at IP3 as a buffer to maintain the pH above 7 for the 30-day period after a LOCA.

The licensee utilized the BORDER code to determine the containment sump pH 30 days after a LOCA. The calculation uses the volumes and boron concentrations from the RWST, the four accumulators and the RCS, as well as the NaOH spray additive tank volume and concentration. The code results are calculated in terms of allowable spray additive tank NaOH concentrations such that the sump pH limits, given as inputs, are met. The licensee did not consider in their calculation the formation of HI, HCl, HNO_3 and CsOH. Although HI, HCl and HNO_3 are strong acids, the contribution that these acids could have in the pH is minimal due to the buffering action of NaOH and the higher concentration of boron found in boric acid. In addition, the licensee did not consider the addition of CsOH into containment which would increase pH, therefore adding conservatism to the calculation.

The NRC staff reviewed the licensee's methodology, assumptions, and performed confirmatory calculations to verify the resulting pH value after 30 days. The NRC staff's independent verification demonstrated that the containment sump pH would remain above 7 for at least 30 days.

Based on the review of the licensee's analyses and the NRC staff's independent verifications, the NRC staff finds the licensee's pH control acceptable.

3.3.2 Iodine Spray Removal Coefficients

Radioactive iodine is released from the containment sump in three different forms: elemental, particulate and organic. For elemental iodine, there are two distinct mechanisms by which radioactive iodine could be removed: containment spray and natural deposition on containment walls. The licensee's analysis does not take credit for the removal of elemental iodine by natural deposition on containment surfaces. Particulate iodine is only removed by the containment spray. There is no effective mechanism for removing organic iodine from the containment atmosphere. Removal of iodine from the containment atmosphere is controlled by two parameters: those controlling the rates of removal, called lambdas (λ), and those determining the maximum amount that can be removed, called decontamination factors (DF).

Removal rates of iodine by spray are a function of volumetric flow of the spray solution, which is reflected in the corresponding spray removal coefficient λ. For IP3 there are two different volumetric flow rates: one during the injection phase and one during the recirculation phase. The licensee performed its calculations in accordance with the methodology described in Section 6.5.2 of the SRP.

The values of the spray removal coefficient for elemental iodine are calculated by the equation given in Section 6.5.2 of the SRP. However, the SRP sets an upper limit of 20 hr^{-1} on the highest acceptable value. Therefore, regardless of the calculated value of λ, it cannot exceed 20 hr^{-1}. Since the licensee's calculated value for λ was 22.7 hr^{-1}, the upper limit of 20 hr^{-1} was used in this calculation. For the spray removal coefficient for particulate iodine, the licensee's calculated values were 4.6 hr^{-1} for the injection phase and 2.2 hr^{-1} for the recirculation phase. The staff performed independent calculations of the λ values for elemental and particulate spray removal coefficients using the equations set forth in SRP Section 6.5.2. The NRC staff found the licensee's results acceptable because the values calculated and used in the analyses were consistent with the NRC staff's calculations.

The DF for iodine in the containment sump is a function of the partition coefficient for iodine, the volume of the containment sump and sump overflow, and the containment building net free volume. The partition coefficient for iodine is a function of the temperature of the sump water, its pH and its iodine content. According to the SRP, the maximum DF should not exceed the value of 200 for elemental iodine. Since the licensee's calculated value for DF was 202.7, the upper limit of 200 was used. Because the removal mechanisms for particulate and organic iodides are significantly different and slower than that for elemental iodine, there is no need to limit the DF for particulate and organic iodides. The NRC staff performed confirmatory calculations of the DF value for the elemental spray removal coefficient using the equation set forth in SRP Section 6.5.2. The NRC staff found the licensee's results acceptable because the values calculated and used in the analyses were consistent with the NRC staff's calculations.

3.3.3 Summary

The NRC staff performed an independent verification of the iodine removal coefficients and decontamination factor calculated by the licensee for removal of the elemental and particulate iodine from the post-accident containment atmosphere. Based on its results, the NRC staff finds that all the values reported by the licensee in its submittal to be conservative, and therefore acceptable.

After an accident, the pH of the containment sump water is determined by the amounts of acidic and basic chemical materials either released from the damaged core or generated in containment and subsequently dissolved in the sump water. It is important to control this pH because if it falls below 7, radioactive iodine could be released to the containment atmosphere. The addition of a buffering agent such as NaOH will keep the water pH above 7, therefore preventing the iodine from being released. The licensee's analysis has indicated that containment sump water will remain greater than 7 for at least 30 days. The NRC staff reviewed the licensee's methodology for determining pH and performed an independent evaluation of the licensee's calculations. Based on its evaluation, the staff concludes that the licensee's proposed actions will maintain the sump water pH greater than 7 for 30 days following a LOCA, thus preventing the release of radioactive iodine into the containment atmosphere.

The NRC staff also evaluated the methods provided by the licensee for the removal of radioactive iodine from the containment atmosphere following a LOCA. The analysis indicated that all iodine removal coefficients and the decontamination factors are based on conservative assumptions incorporated in the licensee's methodology. Based on its evaluation, the NRC staff concludes that the methods for removal of radioactive iodine from the post-accident containment atmosphere are acceptable.

3.4 Proposed Change in Control Room Emergency Mode of Operation

In its June 2, 2004, application, the licensee indicated that the CR operators' dose evaluations had been performed considering two different CR emergency modes of operation. The first assumed 400 cfm of filtered and adsorbed makeup flow and 1000 cfm of filtered and adsorbed recirculation flow. The second assumed 1500 cfm of filtered and adsorbed makeup flow with no filtered or adsorbed recirculation flow.

In the application, the licensee indicated that the manner of operation for the emergency mode would be determined by the results of the inleakage testing of the control room envelope (CRE). In a letter dated December 22, 2004, the licensee indicated that they had selected the second mode as the permanent mode of operation in the event of a radiological event requiring protection of the control room operators. Consequently, the proposed review of the IP3 amendment request also involved the review of the acceptability of this new mode of operation in the event of a radiological accident. With this new mode of operation, the licensee had to demonstrate not only that the dose to the control room operators at Unit 3 from radiological accidents occurring at IP3 met GDC 19, but also that accidents occurring at Indian Point Unit 2 (IP2) did not exceed GDC 19 limits for the IP3 CR operators. Also, because of the proposed power uprate at IP3,[1] it was necessary to demonstrate that radiological accidents occurring at IP3 would not result in doses to the CR operators at IP2 which exceeded GDC 19.

In a February 22, 2005, letter, the licensee provided the dose consequences to the control room operators of IP2 for accidents occurring at IP3 and vice versa. The licensee's calculations showed that GDC 19 would be met at both Indian Point CRs in the event of an accident

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[1]In its application dated June 3, 2004 (ADAMS No. ML041620508), the licensee proposed to increase the steady-state thermal power at IP3 by 4.85% to 3216 megawatts thermal.

occurring at the opposite reactor. In Attachment 7 to its February 11, 2005, letter, the licensee provided some qualitative arguments as to why radiological accidents at IP3 would not result in unacceptable CR operators' doses at IP2 and vice versa.

The NRC staff assessed the consequences of accidents at IP2 affecting the CR operators at IP3 and vice versa. In Attachment 7 to the February 11, 2005, letter, the licensee provided a comparison of the releases of some of the prominent dose-contributing isotopes for a loss-of-coolant accident (LOCA) occurring at IP2 and at IP3. This comparison shows that the releases of non-noble gas isotopes occurring from the IP3 containment range from 92% to 101% of the releases which would occur from the containment during a LOCA at IP2. For the ^{131}I and ^{133}I releases occurring from containment sump leakage, the ratio of IP3 releases to IP2 releases ranged from 32-77%. Referring to the tables in Section 3.5 of this safety evaluation (SE), a comparison of the atmospheric dispersion factors (χ/Q values) for IP3 accident release points affecting the IP3 CR operators to the χ/Q values for IP2 accident release points affecting the IP3 CR operators shows that the IP3 release points have χ/Q values which range from a factor of 3.6 - 11.4 higher than the χ/Q values for releases from IP2. Taking into account the χ/Q values and the relative source terms, the NRC staff concluded that the IP3 CR operators' doses would be higher for an accident occurring at IP3 than the dose to the CR operators of IP3 from an accident occurring at IP2. Thus, the doses to IP3 CR operators from accidents at IP2 are bounded by the analyses of doses to IP3 CR operators from the IP3 accidents discussed in Section 3.6 of this SE. When a similar comparison of χ/Q values was done for the IP2 CR operators, it was found that for releases occurring from IP2 accidents the χ/Q values were a factor of 2 - 3 higher versus the χ/Q values for releases occurring from accidents at IP3. Because the IP2 source term for a LOCA is greater than the IP3 LOCA source term, the NRC staff concluded that the IP2 CR operators would have higher doses for accidents occurring at IP2. Because the staff determined that accidents at IP2 would not result in doses to IP2 CR operators that exceed GDC 19 limits (ADAMS No. ML042960007), the staff concluded that the power uprate and the switch to AST at IP3 would not result in doses which would exceed GDC 19 at IP2.

Additional details on the χ/Q values follow.

3.5 Atmospheric Dispersion Estimates

In order to evaluate the impact of accidents occurring at IP3 at the exclusion area boundary (EAB) and the low population zone (EPZ) boundary, the IP3 EAB and LPZ atmospheric dispersion factors (χ/Q values) for releases from IP3 are required. In order to evaluate the impact of postulated accidents occurring at IP2 and 3 on CR doses at both IP2 and 3, the IP2 and 3 CR atmospheric dispersion factors for releases from both IP2 and 3 are required.

The licensee calculated new IP3 CR χ/Q values for releases from IP2 and 3 as well as IP2 CR χ/Q values for releases from IP3 using onsite hourly meteorological data collected during calendar years 1995–1997. The resulting IP3 CR χ/Q values for releases from IP3 represent a change from those used in the current IP3 updated final safety analysis report (UFSAR) Chapter 14 accident analysis.

The licensee used existing IP2 CR χ/Q values associated with IP2 Amendment No. 211 dated July 27, 2000 (ADAMS No. ML003727500), to perform CR dose assessments related to potential accidental releases from IP2. The licensee also used existing exclusion area boundary (EAB) and low population zone (LPZ) χ/Q values listed in Chapter 14.3 of the IP3 UFSAR for χ/Q values to perform offsite dose assessments related to potential accidental releases from IP3.

3.5.1 Meteorological Data

The licensee previously provided the 1995–1997 onsite hourly meteorological database in support of an earlier license amendment request. These data are discussed in the SE associated with Amendment No. 211 for IP2.

The NRC staff performed a quality review of the 1995–1997 database using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data." Further review was performed using computer spreadsheets. Wind speed and wind direction were measured on the onsite meteorological tower at a height of 10 meters and 60 meters above the ground. Stability class was calculated using the temperature difference between the 60-meter and 10-meter levels. Examination of the data revealed that stable and neutral atmospheric conditions were generally reported to occur at night and unstable and neutral conditions were generally reported to occur during the day, as expected. Wind speed, wind direction, and stability class frequency distributions for each measurement channel were reasonably similar from year to year. The licensee confirmed that the data were collected under the guidelines specified in RG 1.23, "Onsite Meteorological Programs."

On the basis of a review of the available information relative to the onsite meteorological measurements program and the resulting meteorological database, the staff has concluded that the 1995–1997 onsite data provide an acceptable basis for making estimates of atmospheric dispersion for DBA assessments.

3.5.2 Control Room Atmospheric Dispersion Factors

The licensee used the ARCON96 computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes") and the 1995–1997 onsite meteorological database to calculate new CR χ/Q values for use in this license amendment request. RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," states that ARCON96 is an acceptable methodology for assessing CR χ/Q values for use in DBA radiological analyses.

The licensee generated both the IP2 and 3 CR χ/Q values using the CR air intakes as the receptors. In the case of evaluating CR doses for IP3, the staff asked the licensee in a request for additional information (RAI) letter dated January 13, 2005 (ADAMS No. ML050130459), whether the CR air intake χ/Q values are appropriate for use in modeling unfiltered inleakage (see Question 7c). In its RAI response dated February 22, 2005, the licensee stated that a recently completed tracer gas test of the IP3 CR indicated that inleakage in the emergency mode was low (less than 100 cfm) and showed that the major source of the inleakage was co-located with the CR air intake. Consequently, using CR air intake χ/Q values to model unfiltered inleakage is appropriate.

3.5.2.1 IP3 Releases

The licensee calculated new IP2 and 3 CR χ/Q values for releases from IP3. The licensee provided a copy of its IP3 release to IP2 CR ARCON96 computer runs in Attachment 7 to its February 11, 2005, letter. The licensee's documentation of its IP3 release to IP3 CR ARCON96 computer runs was originally provided in Attachment 1 to its letter dated January 5, 2005. These ARCON96 runs were subsequently revised in response to staff RAIs. The licensee's final set of ARCON96 runs for IP3 releases to the IP3 CR are presented in Attachment 3 to its February 11, 2005, letter.

For each analyzed IP3 accident, the licensee assumed all the releases came from one or more of the following release points: containment leakage, containment vent, and three points in the vicinity of the auxiliary boiler feed pump building (ABFB) (i.e., ABFB south side, ABFB safety valve organ pipes, ABFB atmospheric dump valve silencers). The licensee used the following input data and assumptions in applying the ACRON96 model to generate CR χ/Q values due to releases from IP3:

1. The licensee modeled containment leakage as a ground-level vertical area source. Leakage was assumed to be distributed over the containment surface and all penetrations. The source width was the containment outside diameter and the source height was the distance from the roofs of adjoining buildings to the top of the containment dome. The initial diffusion coefficients (plume dimensions) were determined by dividing the source dimensions by a factor of six in accordance with RG 1.194. The height and distance of the containment surface release were determined using numerical integration over the part of the containment surface available for release (i.e., above adjoining buildings where applicable).

2. The licensee modeled containment vent releases as a ground-level horizontal area source. The source dimension was the vent diameter. The initial horizontal diffusion coefficient was determined by dividing the source dimension by a factor of six in accordance with RG 1.194. The release height was set equal to the containment vent height.

3. The licensee modeled the ABFB side releases as a ground-level vertical area source. The initial diffusion coefficients were one sixth of the height and average width of the blowout panels on the south side of the building, projected onto a vertical plane perpendicular to the line of sight from the release point to the CR intake. The distance-to-receptor was based on the intersection of the containment and the shield wall, which was the closest point to the CR intake. The release height was the average height of the blowout panels.

4. The licensee modeled the ABFB organ pipe releases as a ground-level point source. The release height was the height of the top of the organ pipes (which is above the height of the CR air intakes) and the distance-to-receptor was based on the closest organ pipe distance to the CR air intake.

5. The licensee modeled the ABFB silencer releases as a ground-level point source. The release height was the height of the top of the silencers (which is above the height of

the CR air intakes) and the distance-to-receptor was based on the closest silencer distance to the CR air intake.

3.5.2.2 IP2 Releases

The χ/Q values for postulated releases from IP2 to the IP2 CR operators were previously addressed in the SE associated with Amendment No. 211 for IP2. Atmospheric dispersion factors for postulated IP2 releases to the IP3 CR were provided by the licensee in Attachment 1 to its letter dated February 22, 2005. As clarified in a February 23, 2005, teleconference with the NRC staff, the licensee generated the IP3 CR χ/Q values due to releases from IP2 using updated IP2 Amendment No. 211 ARCON96 input data and assumptions that incorporated guidance from RG 1.194.

3.5.2.3 Resulting CR χ/Q values

The NRC staff evaluated the applicability of the ARCON96 model and concluded that there are no unusual siting, building arrangements, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that preclude use of the ARCON96 model for the Indian Point site. The staff qualitatively reviewed the inputs to the ARCON96 calculations and found them generally consistent with site configuration drawings and staff practice. The staff made an independent evaluation of the resulting atmospheric dispersion estimates by running the ARCON96 computer code and obtained similar results.

In summary, the staff reviewed the licensee's assessments of CR post-accident dispersion conditions generated from the licensee's meteorological data and atmospheric dispersion modeling and performed an independent evaluation. The resulting CR χ/Q values are presented in Table 3.5-1. On the basis of this review and evaluation, the staff concluded that the χ/Q values presented in Table 3.5-1 are acceptable for use in DBA CR dose assessments.

Table 3.5-1

Control Room Atmospheric Dispersion Factors

IP3 Releases to IP3 Control Room

Release Location	**χ/Q Value (sec/m³)**				
	0 to 2 hours	**2 to 8 hours**	**8 to 24 hours**	**1 to 4 days**	**4 to 30 days**
Containment Surface	3.57×10^{-4}	3.12×10^{-4}	1.24×10^{-4}	1.06×10^{-4}	7.99×10^{-5}
Containment Vent	6.00×10^{-4}	5.20×10^{-4}	2.12×10^{-4}	1.76×10^{-4}	1.30×10^{-4}
ABFB Side[2]	9.86×10^{-4}	8.74×10^{-4}	4.50×10^{-4}	3.50×10^{-4}	2.80×10^{-4}
ABFB Organ Pipe[3]	1.14×10^{-3}	1.04×10^{-3}	5.05×10^{-4}	4.01×10^{-4}	3.21×10^{-4}
ABFB Silencer[3]	1.00×10^{-3}	8.79×10^{-4}	4.41×10^{-4}	3.47×10^{-4}	2.78×10^{-4}

IP3 Releases to IP2 Control Room

Release Location	**χ/Q Value (sec/m³)**				
	0 to 2 hours	**2 to 8 hours**	**8 to 24 hours**	**1 to 4 days**	**4 to 30 days**
Containment Surface	1.78×10^{-4}	1.38×10^{-4}	5.86×10^{-5}	3.79×10^{-5}	3.14×10^{-5}
Containment Vent	2.95×10^{-4}	2.25×10^{-4}	9.21×10^{-5}	6.06×10^{-5}	4.87×10^{-5}
ABFB Side	3.51×10^{-4}	2.15×10^{-4}	9.14×10^{-5}	6.33×10^{-5}	5.14×10^{-5}
ABFB Organ Pipe	3.49×10^{-4}	2.25×10^{-4}	9.75×10^{-5}	6.73×10^{-5}	5.43×10^{-5}
ABFB Silencer	3.32×10^{-4}	2.12×10^{-4}	9.26×10^{-5}	6.33×10^{-5}	5.14×10^{-5}

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[2]The ABFB Side χ/Q values presented here are the values submitted by Entergy in Attachment 3 to its February 11, 2005, response to an NRC request for additional information. These values are less limiting than the ABFB Side χ/Q values used in the dose assessments performed by the licensee and the staff as described in Section 3.7 of this SE.

[3]The ABFB Organ Pipe and the ABFB Silencer χ/Q values presented here are the values submitted by Entergy in Attachment 3 to its February 11, 2005, response to an NRC request for additional information. These values are less limiting than the ABFB fans χ/Q values used in the dose assessments performed by the licensee and the staff as described in Section 3.6 of this SE.

Control Room Atmospheric Dispersion Factors

IP2 Releases to IP2 Control Room

Release Location	χ/Q Value (sec/m³)				
	0 to 2 hours	**2 to 8 hours**	**8 to 24 hours**	**1 to 4 days**	**4 to 30 days**
Containment Surface	3.82×10^{-4}	2.81×10^{-4}	1.05×10^{-4}	8.31×10^{-5}	7.04×10^{-5}
ABFB Side	1.09×10^{-3}	1.02×10^{-3}	4.99×10^{-4}	3.86×10^{-4}	2.99×10^{-4}
ABFB Stack	9.49×10^{-4}	8.65×10^{-4}	4.17×10^{-4}	3.30×10^{-4}	2.54×10^{-4}
Containment Vent	6.44×10^{-4}	4.69×10^{-4}	1.72×10^{-4}	1.37×10^{-4}	1.17×10^{-4}

IP2 Releases to IP3 Control Room

Release Location	χ/Q Value (sec/m³)				
	0 to 2 hours	**2 to 8 hours**	**8 to 24 hours**	**1 to 4 days**	**4 to 30 days**
Containment Surface	6.77×10^{-5}	6.03×10^{-5}	2.44×10^{-5}	2.06×10^{-5}	1.54×10^{-5}
ABFB Side	9.25×10^{-5}	8.62×10^{-5}	4.30×10^{-5}	3.46×10^{-5}	2.77×10^{-5}
ABFB Stack	9.30×10^{-5}	8.37×10^{-5}	4.15×10^{-5}	3.38×10^{-5}	2.69×10^{-5}
Containment Vent	9.11×10^{-5}	7.84×10^{-5}	3.20×10^{-5}	2.65×10^{-5}	1.98×10^{-5}

3.5.3 Offsite Atmospheric Dispersion Factors

The licensee evaluated offsite doses for IP3 releases using licensing basis offsite (exclusion area boundary (EAB) and low-population zone (LPZ)) χ/Q values presented in IP3 UFSAR Table 14.3-13. These values, presented in Table 3.5-2, were used in the IP3 UFSAR to evaluate the environmental consequences of a LOCA. Details on the calculation of the licensee's EAB and LPZ χ/Q values can be found in IP3 UFSAR, Section 14.3.5. The NRC staff believes that the licensee's UFSAR offsite χ/Q values predate current NRC guidance, RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants." The staff also generated χ/Q values using the 1995 through 1997 onsite meteorological data and RG 1.145 and notes that some of the resultant values are somewhat higher than the licensing basis χ/Q values previously generated by the licensee using another methodology and other meteorological data.

The NRC staff has reviewed the licensee's use of existing IP3 UFSAR EAB and LPZ χ/Q values and has found them to be appropriate for the application in which they are being used since the licensee's use is consistent with the IP3 licensing basis. On the basis of this review, the staff concludes that the χ/Q values presented in Table 3.5-2, as licensing basis values, are acceptable for use in design basis EAB and LPZ dose assessments.

Table 3.5-2

EAB and LPZ Atmospheric Dispersion Factors

Receptor	Time Interval	χ/Q Value (sec/m^3)
Exclusion Area Boundary	All[4]	1.03×10^{-3}
Low Population Zone	0 – 2 Hours	3.8×10^{-4}
	2 – 24 Hours	1.9×10^{-4}
	> 24 Hours	1.7×10^{-5}

3.6 <u>Radiological Dose Analysis</u>

The licensee performed re-analyses of a select number of the IP3 FSAR Chapter 14 accidents. Such re-analyses were required because of the licensee's proposal to increase the power level at IP3 and because of the licensee's desire to implement 10 CFR 50.67 for AST as its licensing basis.

These changes would result in changes in the licensing basis quantities of radioisotopes, the dose acceptance criteria and the significance of various isotopes. In addition, the chemical form for iodine isotopes would have a different distribution. Also, with the implementation of 10 CFR 50.67, the acceptance criteria for allowable doses would now be in terms of TEDE rather than in terms of whole body and thyroid.

The accidents for which the licensee performed such re-analyses and the NRC acceptance criteria guideline dose limits, as stated in Table 6 in RG 1.183, for these accidents at the EAB and at the LPZ are as follows:

1. Large-Break LOCA (LBLOCA) - 25 rem TEDE

2. Locked Rotor - 2.5 rem TEDE

3. Fuel Handling - 6.3 rem TEDE

4. Rod Ejection - 6.3 rem TEDE

5. Small-Break LOCA (SBLOCA) - No stated criterion but limited to 25 rem TEDE

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[4]This exclusion area boundary atmospheric dispersion factor is conservatively applied during all time intervals in determining the limiting 2-hour period.

6. Main Steam Line Break (MSLB)
   - 25 rem TEDE for pre-existing spike case
   - 2.5 rem TEDE for accident initiated spike case

7. Steam Generator Tube Rupture (SGTR)
   - 25 rem TEDE for pre-existing spike case
   - 2.5 rem TEDE for accident initiated spike case

For the CR operators, the acceptance criterion, as stated in 10 CFR 50.67, is 5 rem TEDE for all accidents.

The licensee assessed the consequences of the above accidents to individuals located offsite at the EAB and LPZ and to the CR operators. Consistent with the guidance in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," some re-analyses were performed at 102% (3280 MWt) of the new core power rating. As noted in Section 3.4 of this SE, the licensee proposes to modify the manner of operation of the CR emergency ventilation system (CREVS) at IP3. Their dose assessment accounted for this change in mode of operation. The re-analyses assumed unfiltered inleakage into the CR envelope (CRE) of 700 cfm for all accidents except the LOCA. For that accident, unfiltered inleakage was assumed to be 400 cfm.

The licensee performed a test in January 2005 in accordance with ASTM E741-00, "Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution." This test confirmed that the inleakage characteristics of the IP3 CRE were less than 100 cfm for the radiological mode of operation.

The following sections provide the staff's assessment of the potential consequences of the above postulated accidents. Table 3.6-1 presents the dose consequences as calculated by the staff. The details of the licensee's analyses are contained in the licensee's application dated June 2, 2004, as supplemented by letters dated December 8, 15, and 22, 2004, and January 28, February 11, February 22, and March 14, 2005.

It should be noted that the licensee's analysis also did not account for the fact that the in-place testing criteria for the CR emergency filtration and for the containment fan cooling units' (CFCUs') HEPA filter and charcoal adsorber are 1%. With this high in-place criterion, the full effectiveness of the HEPA filter and the charcoal adsorber is not realized. For purposes of accident dose assessments, the effectiveness of the filter and the adsorber must be reduced to account for the 1% bypass and penetration criteria for in-place tests. Taking this into account, for the CR ventilation system at IP3, the effective filtration and adsorption efficiencies are 98.01% for particulate and 89.01% for elemental and organic iodine. For the CFCUs, the effective filtration rate for HEPA filters would be 89.01%. In a December 22, 2004, letter responding to a staff RAI, it was clear that the licensee's calculations had not accounted for the 1% bypass and penetration criteria in their accident analyses. The licensee needs to account for 1% in-place acceptance criteria in its accident analyses. However, the staff found through its confirmatory calculation that the effect of the inclusion of the 1% bypass and penetration criteria would not have caused the dose limits to be exceeded.

Table 3.6-1

Radiological Consequence of Accidents at IP3 as Calculated by NRC Staff
(In Rem)

Accident	EAB	LPZ	Control Room
LOCA	19.6	11.4	4.7
Locked Rotor	0.039	0.058	0.307
Fuel Handling	5.3	2.0	2.3
Rod Ejection	3.0	1.3	0.74
Small Break LOCA	13.3	5.9	3.4
Main Steam Line Break			
Pre-existing Spike Case	0.36	0.41	0.9
Accident-initiated Spike Case	0.75	0.56	3.0
STGR			
Pre-existing Spike Case	1.7	0.80	1.2
Accident-initiated Spike Case	8.7	3.0	3.0

3.6.1 Accidents Analyzed

3.6.1.1 LBLOCA

The licensee provided an analysis of the consequences of an LBLOCA. The licensee's analysis assumed that the LOCA resulted in releases from two pathways, containment leakage and leakage from the containment sump solution.

3.6.1.1.1 Containment Pathway

The licensee assumed that activity released to the containment was removed by radioactive decay, leakage from the containment, spray removal and sedimentation. All particulate and chemical forms of iodine and noble gases were assumed to be removed via decay and containment leakage. The licensee assumed no credit for the removal of particulate or the elemental and organic forms of iodine as a result of operation of the containment fan cooler unit (CFCU) HEPA filters and charcoal adsorbers. Containment sprays were assumed to remove elemental and particulate iodine in the sprayed region of containment. Sedimentation or natural deposition was assumed to remove particulate iodine in the unsprayed region of containment and in the sprayed region when the sprays were not operating. The organic forms of iodine and noble gases were assumed to be unaffected by sprays or sedimentation.

The licensee assumed initial injection of the spray solution from the refueling water storage tank (RWST). Solution from the RWST would be provided until a predetermined level was reached in the RWST. When that level was reached, the CR operators would stop the flow from the RWST and begin recirculation of containment sump liquid to the sprays. Recirculation of the sump solution was assumed to continue until 4 hours following the LOCA.

The licensee assumed that the sprays became ineffective in removing elemental iodine when the airborne inventory of elemental iodine was reduced to 0.5% of the total iodine released to containment during the gap and early in-vessel release phases. The licensee determined that this ineffectiveness occurred at 2.765 hours following the LOCA.

The licensee assumed that the sprays effectiveness for removing particulate iodine was diminished when the airborne inventory of particulate iodine was reduced to 2.0% of the total iodine released to containment during the gap and early in-vessel release phases. When this occurred, the spray removal coefficient for particulate was assumed to be reduced by a factor of 10. The licensee’s analysis calculated that the reduction in particulate iodine to 2% occurred 3.445 hours following the LOCA.

In the unsprayed region of the containment and during the time that the sprays were not operating in the sprayed region of containment, the licensee assumed that particulate iodine was being removed by natural deposition. The licensee’s analysis assumed that natural deposition ceased when a decontamination factor (DF) of 1000 was achieved.

3.6.1.1.2 Reactor Coolant Pump (RCP) Seal Leak-off Line and Emergency Core Cooling Sysytem (ECCS) Leakage

The pumping of the containment sump solution to areas of the plant external to containment is the other potential pathway for release of radioactivity following an accident. At IP3, the potential pathways are leakage occurring via the RCP leak-off line and leakage during operation of the ECCS in the recirculation mode.

During the first 4 hours following the LOCA, the licensee assumed containment sump liquid leakage occurred via the RCP seal leak-off line. It was assumed that 10% of the activity in this leakage became airborne. Initially, when ECCS recirculation is begun following the LOCA, the recirculation is cold leg recirculation and any leakage would occur within the containment. At 6.5 hours following the LOCA, the switch is made from the recirculation being within the containment to it being external to the containment. In this alignment, any leakage is a source of exposure both offsite and to the CR operators. The licensee assumed that the leakage which occurred external to containment would be released into the auxiliary building. The licensee assumed no filtration or holdup for this activity. The activity in the containment sump was assumed to be reduced via decay and leakage. The licensee assumed that this leakage continued until 30 days following the accident. For this leakage source, the licensee originally assumed 2.8% of the iodine becomes airborne and available for release. The value of 2.8% was based upon calculations performed for the licensee which took into account the recirculation solution's pH, the temperature of the leakage, the room volumes where leakage occurs and ventilation rates. The assessment to support the 2.8% was submitted on December 22, 2004. In a letter dated March 14, 2005, the licensee increased the airborne fraction to 10% as a result of staff comments. To offset the impact of this increase in airborne activity upon the CR operators dose, the licensee also indicated in the March 14 letter that they would decrease the allowable CR envelope inleakage to 400 cfm for the LOCA. In its March 14, 2005, letter, the licensee presented the results of the revised LOCA analysis with the increased ECCS leakage and with the decrease in CRE leakage. The doses were found to meet the acceptance criteria of 10 CFR 50.67 and GDC 19.

3.6.1.1.3 CR Ventilation Systems Operation

It was assumed that the LOCA occurred with the CR's normal ventilation system operating. It was also assumed that a low pressurizer pressure safety injection (SI) signal would cause the CR heating, ventilation, and air conditioning system (HVAC) to switch from normal ventilation system mode of operation to the emergency mode of operation. The licensee assumed that operation of the CREVS would not occur until 60 seconds following the initiation of the event.

3.6.1.1.4 Staff Assessment

The NRC staff has performed an independent calculation of the offsite and onsite consequences of a LOCA. Tables 3.6.1.1-1 through 3.6.1.1-2 provide a listing of the assumptions utilized by the staff in their calculation. The staff assumptions include 10% of the ECCS leakage becoming airborne and available for release and 400 cfm of unfiltered inleakage into the CRE. The results of the staff's calculations are presented in Table 3.6-1. Both the onsite and offsite doses were found to be acceptable for the proposed amendment because they meet the dose criteria of 10 CFR 50.67 and GDC 19.

In its June 2, 2004, application, the licensee indicated that the CFCU's charcoal adsorbers and the HEPA filters were unnecessary to meet the dose criteria of 10 CFR 50.67 and GDC 19. The licensee stated that this laid the groundwork for eventual removal of the CFCU from the VFTP. The staff has concluded that the licensee's analysis has not laid the groundwork. The licensee's analysis remains dependent upon the CFCUs to mix containment air following a LOCA. In addition, the licensee assumes that the CFCU provides mixing and particulate removal for the rod ejection and SBLOCA. Therefore, the CFCU should remain in the VFTP with certain testing requirements retained, e.g. the measurement of CFCU flow rate.

Because of the impact on the review schedule and the NRC staff resources that would have been needed, the NRC staff did not assess the licensee's analysis that proposed an airborne fraction for ECCS leakage of 2.8%. The licensee may resubmit its analysis justifying the value of 2.8% for separate NRC review and approval. In its March 14, 2004, letter, the licensee provided the results of its analyses using the airborne fraction input assumption of 10%, as described in RG 1.183.

Table 3.6.1.1-1

IP3 LOCA Accident Inputs

Parameter	**Value**
Power (MWt)	3280
Fraction of Core Inventory Released from Gap Phase	Noble Gas = 0.05
	Halogens = 0.05
	Alkali Metals = 0.05
Fraction of Core Activity Released During Early In-vessel Phased	Noble Gas = 0.95
	Halogens = 0.35
	Alkali Metals = 0.25
	Tellurium Metals = 0.05
	Ba, Sr = 0.02
	Noble Metals = 0.0025
	Cerium Group = 0.0005
	Lanthanides = 0.0002
Release Phases in Sequence and Duration	Coolant Activity = 10 - 30 seconds
	Gap Activity = 0.5 hr.
	Early In-vessel = 1.3 hr.
CREVS Initiated	1 minute following the beginning of the accident
CREVS Intake Flow Rate (cfm)	1500
Control Room Normal Ventilation System Flowrate (cfm)	1500
CRE Inleakage During CREVS Operation (cfm)	400
CRE Inleakage During Control Room Normal Ventilation System Operation (cfm)	400

CREVS Makeup Filter & Adsorber Efficiencies (%)	Particulate = 98.01
	Organic I = 89.1
	Elemental I = 89.1

Table 3.6.1.1-2

IP3 LOCA Containment and ECCS Leak Path Inputs

Parameter	**Value**	
Containment Net Free Volume (ft^3)	2.61E6	
Containment Sprayed Volume (ft^3)	2.088E6	
Containment Leak Rate(weight %/day)	0 - 24 hrs = 0.1	
	1- 30 days = 0.05	
Iodine Chemical Form Fraction in Containment	Elemental = 0.0485	
	Organic = 0.0015	
	Particulate = 0.95	
Number of Fan Coolers Operating	3	
Fan Cooler Units (FCU) Flow Rate (cfm/unit)	34,000	
FCUs Filter Efficiency for Iodine and Aerosols (%)	0	
Time Delay Before FCUs are Initiated (sec)	60	
Spray Operation		
Time to Initiate Sprays (sec)	67	
Spray Duration (min)	43.9	
Time Delay for Switchover to Recirculating (min)	3	
Time that Recirculating Sprays are Terminated	4 hours post accident	
Injection Spray Flow Rate (gpm)	2200	
Recirculating Spray Flow Rate (gpm)	1050	
Spray Fall Height (ft^3)	118.5	
Containment Spray Removal Coefficients (hr^{-1})	Elemental Iodine	Particulate
Injection	20	4.6
Recirculating	5	2.2
		0.22 after DF = 50

Containment Spray DF	Elemental = 200
	Particulate = 1000
Aerosol Sedimentation Removal in Containment (hr^{-1})	0.1
Containment Sump Volume (gal)	374,400
Sump Solution Leakage Rate Outside Containment (gph)	0 - 4 hrs = 1.0
	4 - 6.5 hrs = 0.
	> 6.5 hrs = 4.0
Airborne Iodine Fraction for Sump Solution Outside Containment	0 - 4 hrs = 0.10
	4 - 6.5 hrs = NA
	> 6.5 hrs = 0.10
Release Period (hrs)	720
Release Location Containment Leakage	Containment Surface
Atmospheric Dispersion Factors for Intake into Control Room from Containment Surface (sec/m^3)	0 - 2 hr = 3.57E-4
	2 - 8 hr = 3.12E-4
	8 - 24 hr = 1.24E-4
	1 - 4 days = 1.06E-4
	4 - 30 days = 7.99E-5
Release Location Containment Sump Leakage	Containment Vent
Atmospheric Dispersion Factors for Intake into Control Room from Containment Vent (sec/m^3)	0 - 2 hr = 6.00E-4
	2 - 8 hr = 5.20E-4
	8 - 24 hr = 2.12E-4
	1 - 4 days = 1.76E-4
	4 - 30 days = 1.30E-4

3.6.1.2 Locked Rotor

3.6.1.2.1 Licensee's Analysis

The licensee's analysis of a locked rotor assumed that it resulted in fuel clad damage. As a result, fission products would be released to the RCS. As a result of primary-to-secondary leakage, some of these fission products enter the secondary side of the steam generators. As a result of the steaming process to remove decay heat from the reactor, fission products on the secondary side would be released to the environment. This accident would also result in the release of iodine activity which is on the secondary side of the steam generators at the beginning of the accident.

The licensee's analysis assumed that the retention of particulate in the steam generators is limited by moisture carryover. The maximum moisture carryover was assumed to be 0.1%. All noble gas activity carried over to the secondary side was assumed to be released immediately to the environment.

At IP3, cooldown of the RCS following a locked rotor accident is assumed to take 29 hours. After 29 hours, the residual heat removal (RHR) system would be placed in service, and there would be no further release of steam to the environment from the secondary side.

For this accident, the licensee assumed that the CR's normal ventilation system was operating at the time of the accident. A high radiation signal would notify the CR operators to place the CR in the emergency mode of operation. The licensee assumed that this radiation signal would occur from the radiation monitor alarm located in the CR ventilation system duct (monitor R33) and that this monitor's set point alarm would not be exceeded until 12 minutes following the initiation of the accident. It was then assumed that it would take an additional 20 minutes to initiate the CREVS.

3.5.1.2.2 Staff Assessment

The NRC staff has performed an independent calculation of the offsite and onsite consequences of the locked rotor accident. Table 3.6.1.2-1 provides a listing of the assumptions utilized by the staff in their calculation. The results of the staff's calculations are presented in Table 3.6-1. Both the onsite and offsite doses were found to be acceptable for the proposed amendment because they meet the acceptance criteria guidelines in RG 1.183.

Table 3.6.1.2-1

IP3 Locked Rotor Accident Inputs

Parameter	Value
Power (MWt)	3280
Initial Reactor Coolant Iodine Activity (μCi/g Dose Equivalent ^{131}I)	1.0
Initial Reactor Coolant Noble Gas Activity	Based upon 1% failed fuel defects
Initial Alkaline Metal Activity	Based upon 1% failed fuel defects
Initial Secondary Coolant Iodine Activity (μCi/g Dose Equivalent ^{131}I)	0.1
Primary to Secondary Leak Rate (gpm)	1
Radial Peaking Factor	1.7
Steam Generator Iodine Water/Steam Partition Coefficient	0.01
Steam Generator Alkaline Metal Water/Steam Partition Coefficient	0.001
Reactor Coolant System Mass (gm)	1.96E8
Steam Release to the Environment (lbs)	0 - 2 hr = 405,000
	2 - 29 hr = 2,303,000
Fraction of Fuel Rods in Core Assumed to Fail	0.05
Fraction of Fission Product Inventory in the Gap	^{131}I = 0.08
	^{85}Kr = 0.10
	Other Noble Gases and Halogens = 0.05
	Alkaline Metals = 0.12
Containment Engineered Safety Features (ESF) Filter System Efficiencies (%)	Particulate = 0
	Organic I = 0
	Elemental I = 0
Fraction of Chemical Form of Iodine in the Water	Particulate = 0
	Organic = 0.0015
	Elemental = 0.9985

Fraction of Chemical Form of Iodine in Release to Environment	Particulate = 0
	Organic = 0.03
	Elemental = 0.97
Release Period (hrs)	29
Release Location	Aux. Boiler Feed Bldg. Fans
CREVS Initiated	32 minutes following the beginning of the accident
CREVS Intake Flow Rate (cfm)	1500
Control Room Normal Ventilation System Flowrate (cfm)	1500
CRE Inleakage During CREVS Operation (cfm)	700
CRE Inleakage During Control Room Normal Ventilation System Operation (cfm)	700
CREVS Makeup Filter and Adsorber Efficiencies (%)	Particulate = 98.01
	Organic I = 89.1
	Elemental I = 89.1
Atmospheric Dispersion Factors for Intake into Control Room (sec/m^3)	0 - 2 hr = 1.19E-3
	2 - 8 hr = 1.12E-3
	8 - 24 hr = 5.59E-4
	1 - 4 days = 4.27E-4
	4 - 30 days = 3.35E-4

3.6.1.3 Fuel-Handling Accident

3.6.1.3.1 Licensee's Analysis

The licensee provided an analysis of the consequences of a fuel-handling accident (FHA) using assumptions such that the results would be bounding whether the accident occurred within containment or in the fuel storage building. The licensee assumed that activity released from such an accident would occur either via the containment purge system or via the fuel storage building ventilation system. The licensee's analysis assumed no credit for operating the fuel storage building ventilation system nor was credit taken for isolating containment following the accident.

The present licensing-basis analysis for an FHA assumes that 75% of the fuel rods in the damaged assembly had the gap activity in Table 3 of RG 1.183. The remaining rods are assumed to have the gap activity presented here in Table 3.6.1.3-1. In the proposed implementation of AST, the licensee no longer took credit for fuel rods in the damaged assembly having gap activities consistent with those presented in Table 3 of RG 1.183. The licensee's new analysis assumes all of the rods in the damaged assembly have the gap activity presented in Table 3.6.1.3-1. The licensee utilized these higher gap activity fractions because the licensee could not ensure that the conditions required by Note 11 to Table 3 would be satisfied. Due to these uncertainties, the licensee increased the activity in the damaged assembly to account for variations in core average enrichments.

RG 1.183 identifies a DF of 500 for elemental iodine based upon a fuel rod pressure of 1200 psig. At IP3, there is a potential for the fuel rod pressure to exceed 1200 psig but to remain less than 1500 psig. The licensee indicated that even at a fuel rod pressure of 1500 psig, the DF for elemental iodine would remain above 400. However, RG 1.183 also specifies an overall iodine DF of 200. With this overall DF and a DF of 1 for organic iodine, the associated DF for elemental iodine would have to be 285. The licensee's analysis assumed an elemental DF of 285 and an overall iodine DF of 200.

The licensee assumed that all of the cesium released from the gap was non-volatile. None of this cesium was assumed to be released from the water above the spent fuel pool or from the water above the reactor vessel flange during fuel handling operations. All noble gases released from the gap were assumed to be released from the water without holdup or removal. The licensee assumed no credit for the removal of iodine either by HEPA filters or charcoal adsorbers at the point of release nor was it assumed that the release was ever isolated.

It was assumed that the FHA occurred with the CR's normal ventilation system operating and that the CREVS would not be initiated.

3.6.1.3.2 Staff Assessment

The NRC staff has performed an independent calculation of the offsite and onsite consequences of an FHA. Table 3.6.1.3-1 contains details of the assumptions utilized by the staff in its calculation. The results of the staff's calculations are presented in Table 3.6-1. Both the onsite and offsite doses were found to be acceptable for the proposed amendment because they meet the acceptance criteria guidelines in RG 1.183.

Table 3.6.1.3-1

IP3 Fuel-Handling Accident

Parameter	Value
Power (MWt)	3280
Fuel Burnup (GWd/MT)	Not addressed
Radial Peaking Factor	1.7
Increase in Gap Activity due to Variations in Core Average Enrichment	1.04
Number of Damaged Fuel Assemblies	1
Total Number of Fuel Assemblies in the Core	193
Fraction of Fission Product Inventory in the Gap	$^{131}I = 0.12$
	$^{85}Kr = 0.30$
	Other Noble Gases and Halogens= 0.10
Decay Time (hrs)	84
Reactor Cavity & Spent Fuel Pool Water Depth (ft)	23
Reactor Cavity & Spent Fuel Pool DF	Elemental I = 285
	Organic I = 1
Containment ESF Filter System Efficiencies (%)	0
Chemical Form Fraction of Iodine in the Water	Particulate = 0
	Organic = 0.0015
	Elemental = 0.9985
Chemical Form Fraction of Iodine in Release to Environment	Particulate = 0
	Organic = 0.30
	Elemental = 0.70
Release Period (hrs)	2
Release Location	Containment Vent
CREVS Initiated	NA

Control Room Normal Ventilation System Flowrate (cfm)	1500
CRE Inleakage During Control Room Normal Ventilation System Operation (cfm)	700
Atmospheric Dispersion Factors for Intake into Control Room (sec/m^3)	6.00E-4

3.6.1.4 Control Rod Ejection

3.6.1.4.1 Licensee's Analysis

The licensee provided an analysis of the consequences of a control rod ejection accident. The licensee assumed that a rod ejection accident resulted in both fuel clad damage and fuel melt. The licensee assumed that the accident resulted in two release paths, one via containment leakage and the other via secondary side steaming. The containment pathway results from the spill to the containment caused by the ejected rod generating a hole in the reactor vessel. Secondary side steaming occurs due to the necessity to remove decay heat from the reactor following the transient.

Radioactivity is released via the secondary side pathway because the melted fuel and the fuel that has been damaged but not melted release their fission products to RCS. Some of these fission products are released to the secondary side of the steam generators due to primary-to-secondary leakage. A portion of these fission products are then released to the environment as a result of the steaming process. Any activity which was already on the secondary side of the steam generators at the beginning of the accident would also be released due to the steaming process.

The source term for the licensee's analysis consisted of the activity in the primary coolant at the time of the accident along with the activity resulting from the fission products which are released from the damaged and melted fuel. In the licensee's analysis for the containment leak path, all of the noble gas and alkali metal activity released from the damaged and melted fuel was considered available for release. For iodine released to containment, all of the activity released from the damaged fuel and 25% of the activity released from the melted fuel was considered available for release. For the containment leak pathway, no credit was taken for plate-out onto containment surfaces or for removal of particulate and elemental iodine as a result of spray operation. However, credit was taken for the removal of particulate using the CFCUs. No credit was assumed for the use of the charcoal adsorber in the CFCUs.

For the secondary side release path, all of the iodine released from the damaged fuel and 50% of the iodine in the melted fuel was considered available for release from the RCS. All of the noble gases and the alkali metal activity released from the damaged and melted fuel was considered available for release from the primary coolant.

The licensee's analysis assumed that primary to secondary leakage and steaming from the steam generators continued until primary system pressure dropped below the secondary side pressure. The licensee indicated that a rod ejection accident pressure transient was similar to a the pressure transient for an SBLOCA. A bounding time of 1 hour was selected for the analysis even though the licensee indicated that analyses of the SBLOCA pressure transient had shown that this would occur well before 1 hour. The licensee assumed steam releases from the steam generators for 2 hours.

The retention of particulate in the steam generators is limited by the moisture carryover. The maximum moisture carryover is 0.1%. All noble gas activity carried over to the secondary side was assumed to be released to the environment immediately.

The rod ejection was assumed to occur with the CR's normal ventilation system operating at the time of the accident. It was further assumed that a low pressurizer pressure SI signal would cause the CR HVAC to switch from normal ventilation system operation to the emergency mode of operation. The licensee assumed that operation of the CREVS would not occur until 140 seconds following the beginning of the accident.

3.6.1.4.2 Staff Assessment

The NRC staff performed independent calculations of the offsite and onsite consequences of a rod ejection accident. Tables 3.6.1.4-1 through 3.6.1.4-3 contains details of the assumptions utilized by the staff in its calculation. The results of the staff's calculations are presented in Table 3.6-1. Both the onsite and offsite doses were found to be acceptable for the proposed amendment because they meet the acceptance criteria guidelines in RG 1.183.

Table 3.6.1.4-1

IP3 Rod Ejection Accident Inputs

Parameter	Value
Power (MWt)	3280
Initial Reactor Coolant Iodine Activity (µCi/g Dose Equivalent ^{131}I)	1.0
Fraction of Fuel Melting	0.0025
Fraction of Gap Activity Released from Failed Fuel	1.0
Fraction of Activity Released from Melted Fuel	
Containment Leakage	Iodine = 0.25
	Noble Gas and Alkaline Metals = 1.0
Primary to Secondary Leakage	Iodine = 0.50
	Noble Gas and Alkaline Metals = 1.0
Initial Reactor Coolant Noble Gas Activity	Based upon 1% failed fuel defects
Initial Alkali Metal Activity	Based upon 1% failed fuel defects
Radial Peaking Factor	1.7
Fraction of Fuel Rods in Core Assumed to Fail	0.10
Fraction of Fission Product Inventory in the Gap	Noble Gases and Halogens= 0.10
	Alkaline Metals = 0.12
CREVS Initiated	140 seconds following the beginning of the accident
CREVS Intake Flow Rate (cfm)	1500
Control Room Normal Ventilation System Flowrate (cfm)	1500
CRE Inleakage During CREVS Operation (cfm)	700
CRE Inleakage During Control Room Normal Ventilation System Operation (cfm)	700
CREVS Makeup Filter & Adsorber Efficiencies (%)	Particulate = 98.01
	Organic I = 89.1
	Elemental I = 89.1

Table 3.6.1.4-2

IP3 Rod Ejection Containment Leak Path Inputs

Parameter	Value
Containment Net Free Volume (ft^3)	2.61E6
Containment Leak Rate(weight %/day)	0 - 24 hrs = 0.1
	1 - 30 days = 0.05
Iodine Chemical Form Fraction in Containment	Elemental = 0.0485
	Organic = 0.0015
	Particulate = 0.95
Spray Removal	Not Credited
Number of CFCUs Operating	3
CFCU Filtered Flow (cfm/unit)	8000
CFCUs Filter Efficiency for Aerosols (%)	89.01
Time Delay Before Filtration is Initiated (sec)	60
Release Period (hrs)	720
Release Location	Containment Surface
Atmospheric Dispersion Factors for Intake into Control Room (sec/m^3)	0 - 2 hr = 3.57E-4
	2 - 8 hr = 3.12E-4
	8 - 24 hr = 1.24E-4
	1 - 4 days = 1.06E-4
	4 - 30 days = 7.99E-5
Aerosol Sedimentation Removal in Containment (hr^{-1})	0.1

Table 3.6.1.4-3

IP3 Rod Ejection Primary to Secondary Leak Path Inputs

Parameter	Value
Initial Secondary Coolant Iodine Activity (µCi/g Dose Equivalent ^{131}I)	0.1
Initial Secondary Coolant Alkali Metal Activity	10% of Primary Concentration
Primary to Secondary Leak Rate (gpm)	1
Chemical Form Fraction of Iodine in Release to Environment	Particulate = 0
	Organic = 0.03
	Elemental = 0.97
Steam Generator Iodine Water/Steam Partition Coefficient	0.01
Steam Generator Alkaline Metal Water/Steam Partition Coefficient	0.001
Reactor Coolant System Mass (gm)	1.96E8
Secondary Coolant Mass (gm/SG)	2.88E7
Release Period (hrs)	2
Release Location	Aux. Boiler Feed Bldg. Fans
Steam Release to the Environment (lbs)	0 - 2 hr = 405,000
Atmospheric Dispersion Factors for Intake into Control Room (sec/m^3)	0 - 2 hr = 1.19E-3
	2 - 8 hr = 1.12E-3
	8 - 24 hr = 5.59E-4
	1 - 4 days = 4.27E-4
	4 - 30 days = 3.35E-4

3.6.1.5 SBLOCA

3.6.1.5.1 Licensee's Analysis

The licensee provided an analysis of the consequences of a SBLOCA. The licensee assumed that an SBLOCA resulted in fuel clad damage. The licensee also assumed a break in the RCS.

However, the break was small enough that the containment spray system was not immediately actuated by high containment pressure. It was assumed that the core experienced substantial cladding damage such that the fission product gap activity of all fuel rods was released.

The SBLOCA was assumed to result in release via two pathways, containment leakage and secondary side steaming. Containment leakage results from the release from the break to the containment with the containment leaking to the environment at its TS limit leak rate. The release via secondary side steaming occurs due to the necessity for removing decay heat from the reactor. Radioactivity is in the steam as a result of the release of fission products to primary coolant from the damaged fuel. These fission products enter the secondary side of the steam generators due to primary-to-secondary leakage. A portion of these fission products are then released to the environment as a result of the steaming. Also released would be any activity which would be on the secondary side of the steam generators at the beginning of the accident.

For the containment leak pathway, the licensee assumed no credit for plate-out onto containment surfaces or for removal of particulate and elemental iodine as a result of spray operation. However, credit was taken for the removal of particulate using the CFCUs. No credit was assumed for the use of the charcoal adsorber in the CFCUs. The licensee indicated its SBLOCA analysis was in support of the potential future elimination of the CFCU filters, combined with the elimination of the spray additive. As noted in Section 3.6 above, the removal of CFCUs from the VFTP is not supported by this amendment request.

For the steaming release path, all of the iodine released from the damaged fuel and 50% of the iodine in the melted fuel was considered available for release from reactor coolant. All of the noble gases and the alkali metal activity released from the damaged fuel were considered available for release from the primary coolant. The licensee's analysis assumed that primary-to- secondary leakage and steaming from the steam generators continued until RCS pressure dropped below the secondary side pressure. A bounding time of 1 hour was selected for the analysis even though analyses of the SBLOCA pressure transient had shown that this would occur well before 1 hour. The licensee assumed steam releases from the steam generators for 2 hours. The locked rotor steam releases were applied to this analysis. The licensee indicated that such an assumption was conservative since this assumption does not account for ECCS injection to adsorb decay heat.

The retention of particulate in the steam generators was limited by the moisture carryover. The maximum moisture carryover is 0.1%. All noble gas activity carried over to the secondary side was assumed to be released to the environment immediately.

The SBLOCA was assumed to occur with the CR's normal ventilation system operating at the time of the accident. It was further assumed that a low pressurizer pressure SI signal would cause the CR HVAC to switch from normal ventilation system operation to the emergency mode of operation. The licensee assumed that operation of the CREVS would not occur until 140 seconds following the initiation of the event.

3.6.1.5.2 Staff's Assessment

The NRC staff has performed an independent calculation of the offsite and onsite consequences of an SBLOCA. Tables 3.6.1.5-1 through 3.6.1.5-3 contain details of the assumptions utilized by the staff in their calculation. The results of the staff's calculations are presented in Table 3.6-1. Both the onsite and offsite doses were found to be acceptable for the proposed amendment because they meet the dose criteria of 10 CFR 50.67 and GDC 19 and acceptance criteria guidelines in RG 1.183.

Table 3.6.1.5-1

IP3 SBLOCA Accident Inputs

<u>**Parameter**</u>	<u>**Value**</u>
Power (MWt)	3280
Initial Reactor Coolant Iodine Activity (μCi/g Dose Equivalent ^{131}I)	60.0
Fraction of Fuel Melting	0.00
Fraction of Core Gap Activity Released	1.0
Initial Reactor Coolant Noble Gas Activity	Based upon 1% failed fuel defects
Initial Alkali Metal Activity	Based upon 1% failed fuel defects
Fraction of Fuel Rods in Core Assumed to Fail	1.0
Fraction of Fission Product Inventory in the Gap	Noble Gases, Halogens, & Alkaline Metals = 0.05
CREVS Initiated	140 seconds following the beginning of the accident
CREVS Intake Flow Rate (cfm)	1500
Control Room Normal Ventilation System Flowrate (cfm)	1500
CRE Inleakage During CREVS Operation (cfm)	700
CRE Inleakage During Control Room Normal Ventilation System Operation (cfm)	700
CREVS Makeup Filter and Adsorber Efficiencies (%)	Particulate = 98.01
	Organic I = 89.1
	Elemental I = 89.1

Table 3.6.1.5-2

IP3 SBLOCA Containment Leak Path Inputs

Parameter	**Value**
Containment Net Free Volume (ft^3)	2.61E6
Containment Leak Rate(weight %/day)	0 - 24 hrs = 0.1
	1 - 30 days = 0.05
Iodine Chemical Form Fraction in Containment	Elemental = 0.0485
	Organic = 0.0015
	Particulate = 0.95
Spray Removal	Not Credited
CFCUs Operating	3
CFCUs Filtered Flow (cfm/unit)	8000
CFCUs Filter Efficiency for Aerosols (%)	89.01
Time Delay Before Filtration is Initiated (sec)	60
Release Period (hrs)	720
Release Location	Containment Surface
Atmospheric Dispersion Factors for Intake into Control Room (sec/m^3)	0 - 2 hr = 3.57E-4
	2 - 8 hr = 3.12E-4
	8 - 24 hr = 1.24E-4
	1 - 4 days = 1.06E-4
	4 - 30 days = 7.99E-5

Table 3.6.1.5-3

IP3 SBLOCA Primary to Secondary Leak Path Inputs

Parameter	**Value**
Initial Secondary Coolant Iodine Activity (µCi/g Dose Equivalent ^{131}I)	0.1
Initial Secondary Coolant Alkali Metal Activity	10% of Primary Concentration
Primary to Secondary Leak Rate (gpm)	1
Chemical Form Fraction of Iodine in Release to Environment	Particulate = 0
	Organic = 0.03
	Elemental = 0.97
Steam Generator Iodine Water/Steam Partition Coefficient	0.01
Steam Generator Alkaline Metal Water/Steam Partition Coefficient	0.001
Reactor Coolant System Mass (gm)	1.96E8
Secondary Coolant Mass (gm/SG)	2.88E7
Release Period (hrs)	2
Release Location	Aux. Boiler Feed Bldg. Fans
Steam Release to the Environment (lbs)	0 - 2 hr = 405,000
	>2 hr = 0
Atmospheric Dispersion Factors for Intake into Control Room (sec/m^3)	0 - 2 hr = 1.19E-3
	2 - 8 hr = 1.12E-3
	8 - 24 hr = 5.59E-4
	1 - 4 days = 4.27E-4
	4 - 30 days = 3.35E-4

3.6.1.6 MSLB

3.6.1.6.1 Licensee's Analysis

The licensee calculated the potential consequences of an MSLB accident. Doses were calculated for two cases. The first case assumed that the MSLB occurred following an iodine spike. This is referred to as the pre-existing spike case. For this case, the MSLB accident is assumed to occur with primary coolant activity at the maximum value allowed by the TSs for dose equivalent ^{131}I. The second case, referred to as the accident-initiated spike case, assumes that the MSLB accident induces an iodine spike. At the time of the accident, primary coolant is assumed to occur at the equilibrium TS value for dose equivalent ^{131}I.

The licensee's analysis assumed that the steam generator with the broken steam line, commonly referred to as the faulted steam generator, would boil dry within 5 minutes following the line break. The entire liquid inventory of this steam generator would be steamed off and all of the iodine initially in the steam generator would be released to the environment. In addition, any iodine carried over to the faulted steam generator by tube leaks would be assumed to be released directly to the environment with no credit taken for iodine retention in the steam generator. All noble gas activity carried over to the secondary side as a result of primary to secondary leakage, whether to the faulted or to the remaining steam generators (commonly referred to as intact) was assumed to be released immediately to the environment.

For the accident-initiated spike case, the licensee terminated the iodine spike at 3 hours. The licensee indicated that based upon an iodine gap fraction of 8%, the iodine gap activity would all be released within 3 hours.

The licensee's assessment assumed that it took 72 hours to cool the RCS below 212 EF. When this occurred, releases from the faulted steam generator would cease. The licensee indicated that reactor cooldown at IP3 can be accomplished within 29 hours following an MSLB. For the MSLB analysis, it was assumed that the residual heat removal (RHR) was placed into service at 29 hours and that there was no further release to the environment from the intact steam generators.

When the MSLB occurred, it was assumed that the CR's normal ventilation system was operating. The licensee indicated that a low pressurizer pressure SI signal would cause the CR HVAC system to switch from the normal ventilation system mode of operation to the emergency mode of operation. The licensee assumed that operation of the CREVS would not occur until 60 seconds following the initiation of the event.

3.6.1.6.2 Staff Assessment

The NRC staff has performed an independent calculation of the offsite and onsite consequences of an MSLB. Table 3.6.1.6-1 contains details of the assumptions utilized by the staff in their calculation. The results of the staff's calculations are presented in Table 3.6-1.

The staff noted that the licensee had assumed that the spiking release occurred after 3 hours. It appeared that the basis for this assumption was derived from: (1) the I-131 activity level in primary coolant with primary coolant at the dose equivalent I-131 value of 1 μCi/g; (2) the equilibrium release rate of I-131 to obtain its specific activity level and (3) the fuel rods available to contribute to the spike. For the latter, it appeared that the licensee assumed that the I-131 in primary coolant equated to a certain defect percentage of fuel rods in the core. For the accident-initiated spike case, it seems that the licensee assumed that the accident would not result in any additional defective fuel rods. Therefore, the licensee concluded that the spike would only occur from those previously identified rods with defects. The NRC staff has assessed this approach. The staff has concluded that it is inappropriate to presume that the accident will not initiate additional defective fuel rods. The staff believes that the transient may initiate previously intact fuel rods to leak. Therefore, the licensee needs to assess their assumption of 3 hours for the spike duration. The staff assumed a spike duration of 4 hours based upon 8% of the core iodine activity being in the gap and the gap release occurring from 1% of the fuel rods. However, based on the results of its confirmatory calculation, the staff finds the results of the MSLB accident acceptable because they meet the acceptance criteria guidelines in RG 1.183.

TABLE 3.6.1.6-1

IP3 MSLB Accident

Parameter	**Value**
Pre-existing Spike Initial Reactor Coolant Iodine Activity (μCi/g Dose Equivalent ^{131}I)	60
Accident-Initiated Spike Initial Reactor Coolant Iodine Activity (μCi/g Dose Equivalent ^{131}I)	1.0
Initial Reactor Coolant Noble Gas Activity	Based upon 1% failed fuel defects
Initial Secondary Coolant Iodine Activity (μCi/g Dose Equivalent ^{131}I)	0.1
Faulted Steam Generator Primary to Secondary Leak Rate (gpm)	0.3
Total Intact Steam Generators Primary to Secondary Leak Rate (gpm)	0.7
Iodine Spike Appearance Rate	500 Times Equilibrium Rate
Faulted Steam Generator Iodine Water/Steam Partition Coefficient	1.0
Intact Steam Generator Iodine Water/Steam Partition Coefficient	0.01
Reactor Coolant System Mass (gm)	1.96E8
Initial Mass in Faulted Steam Generator (gm)	6.459E7

Initial Mass in Intact Steam Generator (gm/SG)	2.88E7
Duration of Accident-Initiated Spike (hrs)	4
Time to Cool Reactor Coolant System below 212 EF and Stop Releases from Faulted Steam Generator (hrs)	72
Steam Release to the Environment (lbs)	0 - 2 hr = 402,000
	2 - 29 hr = 2,273,500
Chemical Form Fraction of Iodine in Release to Environment	Particulate = 0
	Organic = 0.03
	Elemental = 0.97
Release Period for Intact SGs. (hrs)	29
Release Location for Faulted Steam Generator	Aux. Boiler Feed Bldg. Fans
Release Location for Faulted Steam Generator	Aux. Boiler Feed Bldg. Side
CREVS Initiated	1 minute following the beginning of the accident
CREVS Intake Flow Rate (cfm)	1500
Control Room Normal Ventilation System Flowrate (cfm)	1500
CRE Inleakage During CREVS Operation (cfm)	700
CRE Inleakage During Control Room Normal Ventilation System Operation (cfm)	700
CREVS Makeup Filter and Adsorber Efficiencies (%)	Particulate = 98.01
	Organic I = 89.1
	Elemental I = 89.1
Auxiliary Boiler Feed Bldg. Fans Atmospheric Dispersion Factors for Intake into Control Room (sec/m^3)	0 - 2 hr = 1.19E-3
	2 - 8 hr = 1.12E-3
	8 - 24 hr = 5.59E-4
	1 - 4 days = 4.27E-4
	4 - 30 days = 3.35E-4

Auxiliary Boiler Feed Bldg. Side Atmospheric Dispersion Factors for Intake into Control Room (sec/m^3)	0 - 2 hr = 1.18E-3
	2 - 8 hr = 1.06E-3
	8 - 24 hr = 5.42E-4
	1 - 4 days = 4.09E-4
	4 - 30 days = 3.27E-4

3.6.1.7 Steam Generator Tube Rupture (SGTR)

3.6.1.7.1 Licensee's Analysis

The licensee calculated the potential consequences of an SGTR accident. Doses were calculated for two cases. The first case assumed that the SGTR occurred following an iodine spike. This is referred to as the pre-existing spike case. For this case, the SGTR accident is assumed to occur with primary coolant activity at the maximum instantaneous value allowed by TSs for dose equivalent ^{131}I. The second case, referred to as the accident-initiated spike case, assumes that the SGTR accident induces an iodine spike. At the time of the accident, primary coolant is assumed to be at the equilibrium TS value for dose equivalent ^{131}I.

The licensee's analysis assumed that a reactor trip would not occur until 592 seconds following the tube rupture. They also assumed that a loss of offsite power would occur coincident with the reactor trip. The steam generator with the ruptured tube is commonly referred to as the faulted steam generator. The steam generators without the tube rupture are commonly referred to as the intact steam generators.

With the tube rupture, primary coolant would be discharged through the break. A portion of this flow would flash and be released directly to the environment. That portion which did not flash would be collected in the steam generator and available for release as steam via the steaming process. All noble gas activity released from the break was assumed to be released immediately to the environment.

The licensee's analysis assumed that break flow would be terminated within 30 minutes following the tube rupture. The licensee indicated that they did not view this assumption as constituting a requirement that operators demonstrate the ability to terminate break flow within 30 minutes of an SGTR. The licensee also indicated that they would not be able to terminate break flow for all postulated SGTR events. The licensee indicated that their analysis remained conservative because they assumed a constant break flow at the equilibrium flow rate with a constant flashing fraction that does not credit plant cooldown. This analysis took into account charging flow and the potential for steam generator overfill. Westinghouse also utilized a more precise mass release analysis. The licensee stated that the Westinghouse analysis showed that the 30-minute constant break flow model bounded the 60-minute response time analysis. The licensee indicated that this justified changing the emergency operating procedures (EOPs) to allow a 60-minute operator response time.

For the accident-initiated spike case, the licensee terminated the iodine spike at 4 hours. The licensee indicated that, based upon an iodine gap fraction of 8%, all iodine gap activity would be released within 4 hours.

The licensee indicated that an RCS cooldown at IP3 can be accomplished within 29 hours following an SGTR. At that time, the RHR system would be placed into service and there was no further release to the environment from the intact steam generators.

The licensee's assessment assumed that when the SGTR occurred, the CR's normal ventilation system was operating. The licensee indicated that a low pressurizer pressure SI signal would be reached in approximately 6.5 minutes. This signal would cause the CR ventilation system to switch from normal ventilation system operation to the emergency mode of operation. The licensee assumed that operation of the CREVS would not occur until 60 seconds following the low pressurizer pressure SI signal or 7.5 minutes following the SGTR.

3.6.1.7.2 Staff's Assessment

The NRC staff has performed an independent calculation of the offsite and onsite consequences of an MSLB. Table 3.6.1.7-1 contains details of the assumptions utilized by the staff in their calculation. The results of the staff's calculations are presented in Table 3.6-1.

For similar reasons as noted in Section 3.6.1.6 for the MSLB, the staff believes that the licensee needs to assess its assumption for spike duration. The staff assumed that the spike during an SGTR would continue for 8 hours.

Table 3.6.1.7-1

IP3 SGTR Accident

Parameter	Value
Pre-existing Spike Initial Reactor Coolant Iodine Activity (µCi/g Dose Equivalent ^{131}I)	60
Accident Initiated Spike Initial Reactor Coolant Iodine Activity (µCi/g Dose Equivalent ^{131}I)	1.0
Initial Reactor Coolant Noble Gas Activity	Based upon 1% failed fuel defects
Initial Alkaline Metal Activity	Based upon 1% failed fuel defects
Initial Secondary Coolant Iodine Activity (µCi/g Dose Equivalent ^{131}I)	0.1
Tube Rupture Break Flow (lb)	Pre-trip = 38,500
	Post-trip until 30 minutes = 99,500
Reactor Trip (seconds post-accident)	392
Intact Steam Generators Primary to Secondary Leak Rate (gpm/SG)	0.3
Iodine Spike Appearance Rate	335 times equilibrium rate
Tube Rupture Break Flow Flashing Fraction	Pre-trip = 0.21
	Post-trip = 0.15
Time to Isolate Faulted SG (min)	30
Faulted Steam Generator Iodine Water/Steam Partition Coefficient	1.0
Faulted and Intact Steam Generator Iodine Water/Steam Partition Coefficient	0.01
Steam Generator Alkaline Metal Water/Steam Partition Coefficient	0.001
Reactor Coolant System Mass (gm)	1.96E8
Mass in Steam Generator (gm/SG)	2.88E7
Duration of Accident-Initiated Spike (hrs)	8
Time to Cool Reactor Coolant System below 212°F & Stop Releases from Intact Steam Generator (hrs)	29

Steam Release to the Environment from Faulted SG	0 - 392 sec = 1070.2 lb/sec
	392 sec - 30 minutes = 72,000 lbs
Steam Release to the Environment from Intact SGs. (lbs)	0 - 392 sec = 3210.6 lb/sec
	392 sec - 2 hrs = 526,000 lbs
	2 - 8 hr = 1,160,000 lbs
	8 - 29 hr = 1,580,000 lbs
Chemical Form Fraction of Iodine in Release to Environment	Particulate = 0
	Organic = 0.03
	Elemental = 0.97
Release Location for Faulted Steam Generator	Aux. Boiler Feed Bldg. Fans
Release Location for Faulted Steam Generator	Aux. Boiler Feed Bldg. Side
CREVS Initiated	7.5 minutes following the beginning of the accident
CREVS Intake Flow Rate (cfm)	1500
Control Room Normal Ventilation System Flowrate (cfm)	1500
CRE Inleakage During CREVS Operation (cfm)	700
CRE Inleakage During Control Room Normal Ventilation System Operation (cfm)	700
CREVS Makeup Filter and Adsorber Efficiencies (%)	Particulate = 98.01
	Organic I = 89.1
	Elemental I = 89.1
Auxiliary Boiler Feed Bldg. Fans Atmospheric Dispersion Factors for Intake into Control Room (sec/m^3)	0 - 2 hr = 1.19E-3
	2 - 8 hr = 1.12E-3
	8 - 24 hr = 5.59E-4
	1 - 4 days = 4.27E-4
	4 - 30 days = 3.35E-4

3.6.2 Summary

The NRC staff has assessed the licensee's proposed implementation of AST. The staff has confirmed that those accidents for which the licensee chose to implement AST at the proposed stretch power uprate level of 3216 MWt would meet the dose criteria presented in 10 CFR 50.67 and GDC 19, would meet the acceptance criteria guidelines in RG 1.183, and would be an acceptable implementation of 10 CFR 50.67. Therefore, the staff finds the proposed implementation of AST acceptable.

3.7 Summary of TS Changes

As a result of the proposed implementation of the AST, it was necessary that the licensee propose to alter the definition for dose equivalent I-131. As stated in its March 14, 2005, letter, the licensee proposed the following definition:

> DOSE EQUIVALENT I-131 shall be that amount of I-131 (curies) that alone would produce the same committed effective dose equivalent (CEDE) dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The CEDE dose conversion factors used for this calculation shall be those listed in Table 2.1 of Environmental Protection Agency Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," 1988.

The NRC staff has reviewed the proposed definition, as modified by the March 14, 2005, letter, and finds the proposed definition acceptable because it is consistent with the historical definition of dose equivalent I-131 and the common definition for dose equivalent I-131 presently in the Improved Standard TSs.

The licensee also proposed changes to the VFTP. Specifically, the licensee proposed to delete the containment purge and the fuel storage building emergency ventilation systems from the VFTP. The licensee proposed these systems for deletion because the re-analyses of the postulated accidents for IP3 did not assume credit for these systems as part of the mitigation actions for the facility. The NRC staff has reviewed their proposed deletion. The staff finds their deletion acceptable because the licensee demonstrated that acceptable doses could be obtained without taking credit for these systems and because GDC 60, 61, and 64 would continue to be met even though these systems were being removed from the VFTP. Regarding the latter, in response to a staff RAI, the licensee indicated that even though these systems will be removed from the VFTP, the charcoal will continue to be maintained and tested in these systems. In addition, no changes are being made to the systems other than the relocation of the testing requirements and releases of radioactive materials will be suitably controlled and monitored. Therefore, based upon the licensee's response to the RAI, GDCs 60, 61, and 64 will continue to be met.

The licensee proposed to revise the SRs in TS 3.3.7 regarding the CRVS actuation system and TS 3.7.11 regarding the CRVS. Specifically, SR 3.3.7.A.1, 3.3.7.B.1, and 3.7.11.4 would be revised to change the terminology for the CRVS Mode description to be consistent with the modifications being proposed for the CRVS. In addition, the makeup flow specified in SR 3.7.11.4 would be changed as part of the modification to the CRVS and to be consistent with the licensee's analyses. The NRC staff finds the proposed changes to the CRVS operating mode description inTS Sections 3.3.7 and 3.7.11.4 acceptable since the changes are editorial in nature. The change to the makeup flow in SR 3.7.11.4 is consistent with the emergency mode of operation of the CRVS and with the input assumption in the accident analyses and is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (69 FR 53104). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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