July 27, 2000

Mr. A. Alan Blind
Vice President, Nuclear Power
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, NY 10511

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 - RE: ISSUANCE OF AMENDMENT AFFECTING CONTAINMENT AIR FILTRATION, CONTROL ROOM AIR FILTRATION, AND CONTAINMENT INTEGRITY DURING FUEL HANDLING OPERATIONS (TAC NO. MA6955)

Dear Mr. Blind:

The Commission has issued the enclosed Amendment No. 211 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated November 18, 1999, which incorporated supporting analyses provided to the NRC by your letter of October 8, 1999 (Westinghouse analysis of radiological consequences), as supplemented by letters dated February 14, March 21, April 6, April 13, and May 11, 2000.

The amendment consists of changes to the TSs which result from implementation of an alternate radiological source term as permitted by 10 CFR 50.67 and implements plant modifications to the containment air handling systems and the control room air handling systems related to the use of the alternate source term. This amendment is the product of a pilot proposed by Consolidated Edison. The NRC solicited pilot projects such as this to obtain insights in support of the rulemaking for the implementation of alternative source terms (64 FR 71990) and the supporting regulatory guidance.

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, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-247

Enclosures: 1. Amendment No. 211 to DPR-26
2. Safety Evaluation

cc w/encls: See next page

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Vice President, Nuclear Power
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ACCESSION NO. ML003727500

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DATED: July 27, 2000

AMENDMENT NO. 211 TO FACILITY OPERATING LICENSE NO. DPR-26-INDIAN POINT UNIT 2

PUBLIC
PDI-1 Reading
OGC
M. Gamberoni
P. Milano
G. Hill (2), T-5 C3
W. Beckner, 013/H15
ACRS
J. Rogge, Region I

cc: Plant Service list

Indian Point Nuclear Generating Station
Unit 2

Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, NY 10511

Mr. F. William Valentino, President
New York State Energy, Research,
and Development Authority
Corporate Plaza West
286 Washington Ave. Extension
Albany, NY 12203-6399

Mr. John McCann
Manager of Nuclear Safety and
Licensing
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, NY 10511

Senior Resident Inspector
U. S. Nuclear Regulatory Commission
P.O. Box 38
Buchanan, NY 10511

Mr. Brent L. Brandenburg, Esquire
Assistant General Counsel
Consolidated Edison Company
of New York, Inc.
4 Irving Place - 1822
New York, NY 10003

David Lochbaum
Nuclear Safety Engineer
Union of Concerned Scientists
1616 P Street, NW., Suite 310
Washington, DC 20036

Edward Smeloff
Pace University School of Law
The Energy Project
78 North Broadway
White Plains, NY 10603

Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, NY 10271

Ms. Charlene D. Faison, Director
Nuclear Licensing
Power Authority of the State
of New York
123 Main Street
White Plains, NY 10601

Mr. Thomas Rose
Secretary - NFSC
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, NY 10511

Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. Paul Eddy
New York State Department of
Public Service
3 Empire State Plaza, 10th Floor
Albany, NY 12223

Jim Riccio
Public Citizen's Critical Mass Energy
Project
215 Pennsylvania Ave., SE
Washington, DC 20003

Michael Mariotte
Nuclear Information & Resources Service
1424 16th Street, NW, Suite 404
Washington, DC 20036

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 211
License No. DPR-26

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

   A. The application for amendment by Consolidated Edison Company of New York, Inc. (the licensee) dated November 18, 1999, which incorporated supporting analyses submitted by a letter dated October 8, 1999, as supplemented by letters dated February 14, March 21, April 6, April 13, and May 11, 2000, 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-26 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 211 , 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 of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA Elinor Adensam for/

Marsha Gamberoni, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 27, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 211

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

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

Remove Pages	Insert Pages
iv	iv
3.3-3	3.3-3
3.8-1	3.8-1 (retyped only)
3.8-2	3.8-2
3.8-3	3.8-3
3.8-4	3.8-4 (retyped only)
4.5-1	4.5-1 (retyped only)
4.5-2	4.5-2
4.5-3	4.5-3
4.5-4	4.5-4
4.5-5	4.5-5 (retyped only)
4.5-6	4.5-6 (retyped only)
4.5-7	4.5-7 (retyped only)
5.2-1	5.2-1 (retyped only)
5.2-2	5.2-2

TS Bases Sections

3.3-13	3.3-13
3.3-15	3.3-15
3.8-5	3.8-5
3.8-6	3.8-6 (retyped only)
4.5-8	4.5-8 (retyped only)
4.5-9	4.5-9
4.5-10	4.5-10 (retyped only)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 211 TO FACILITY OPERATING LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

1.0 INTRODUCTION

By letter dated November 18, 1999, which incorporated supporting analyses submitted by a letter dated October 8, 1999, as supplemented by letters dated February 14, March 21, April 6, April 13, and May 11, 2000, the Consolidated Edison Company of New York, Inc. (the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 2 (IP2) Technical Specifications (TSs). The requested changes would remove the requirement for charcoal filters and high efficiency particulate (HEPA) filters in the containment fan cooler system (the charcoal and HEPA filters are themselves being removed from the system), revise the time requirement for subcriticality prior to core alterations from 174 hours to 100 hours, revise flow rate requirements for containment fan coolers and control room ventilation units to be consistent with the design basis, state that the control room ventilation system, in the post-accident mode, will be operated with filtered intake of outside air, allow containment personnel access doors to be open during refueling operations, and allow an administrative substitution of “monthly” in place of “every 31 days” in various surveillance requirements.

The licensee’s request was based on a reanalysis (References 1, 3, and 5) of the radiological consequences of accidents for IP2 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) fuel handling accident; (3) locked rotor accident; (4) rod ejection accident; (5) small-break LOCA; (6) main steam line break; and (7) steam generator tube rupture. The radiological consequences were determined in terms of radiation doses (Total Effective Dose Equivalent) at the site boundary, at the low population zone and in the control room. The letters dated February 14, March 21, April 6, April 13, and May 11, 2000, provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

2.1 Technical Specification Changes

The specific changes to TS requested are as follows:

(1) change the title of 4.5.D of the table of contents to delete the words “Air Filtration” (This change is to reflect the revised function of the containment fan cooler system to cooling of containment only, as a result of the removal of HEPA and charcoal filters.);

(2) revise TS 3.3.B.1.b. to delete the words “charcoal filter” (This change reflects the removal of the charcoal filters from the fan cooler units.);

(3) change TS 3.8.B.4 “174 hours” to “100 hours” (This change reflects the reanalysis for the minimum time for radioactive decay before moving fuel.);

(4) revise TS 3.8.B.8 to delete “and at least one personnel door in the equipment door or closure plate and in the personnel air lock” (This change reflects a reanalysis of the fuel handling accident where no credit is taken for containment isolation.);

(5) revise TS 4.5.D. to delete the words “AIR FILTRATION” (This change is to reflect the revised function of the system to cooling of containment only, as a result of the removal of HEPA and charcoal filters.);

(6) modify TS 4.5.D.1 and TS 4.5.E.1 to change “per 31 days” to “monthly,” and delete the words “HEPA filters and charcoal adsorbers” (This change would make the terminology consistent as defined in the specifications. Monthly and 31 days are used synonymously. Deletion of testing requirements is consistent with the removal of the filters themselves.);

(7) revise TS 4.5.D.2 to change “65,600 cfm +/-10%” to “greater than or equal to 64,500 cfm,” and delete the remaining parts of 4.5.D.2 and 4.5.D.3 through 4.5.D.6. (This change is to specify the flows consistent with the reanalysis of design-basis accidents utilizing the NUREG-1465 alternate source term. The +/- 10% is no longer required, since a residence time for charcoal filters need not be specified after the filters are removed. The remaining parts of this specification relate to testing of filters, which are themselves being removed.);

(8) revise TS 4.5.E.2.a, b, and c; 4.5.E.4.a; 4.5.E.5; and 4.5.E.6 to change “1840 cfm” to “2000 cfm” (This change would modify the flow rate to be consistent with the current design of the control room filtration system and assumptions in the reanalysis of the design-basis accidents.);

(9) revise TS 4.5.E.4.b to change “recirculation” to “filtered-intake” (This change would modify the description of the mode of operation of the control room ventilation system in response to a safety injection signal or a high radiation signal to be consistent with the current design of the control room filtration system and assumptions in the reanalysis of the design-basis accidents.);

(10) revise TS 4.5.E.4.c to change “outside atmosphere” to “adjacent areas” (This change would modify the criteria for testing control rooms to conform with regulatory guidance.);

(11) revise TS 5.2.D.2 to delete “All the fan cooler units are equipped with activated charcoal filters to remove following an accident” (This change reflects the removal of the charcoal filters from the fan cooler units.).

TS Basis would be revised as follows:

(1) TS Bases page 3.3-13 would be revised to delete “plus charcoal filters”;

(2) TS Bases page 3.3-15 would be revised to delete “and/or for recirculation”;

(3) TS Bases pages 3.8-5 and 3.8-6 would be modified to change “174 hours” to “100 hours” and the last se 3.8-5 would be modified to state “The analysis of the fuel handling accident inside and outside of the contain credit for removal of radioactive iodine by charcoal filters”; and

(4) TS Bases page 4.5-10 would be revised to delete the fourth paragraph, which describes the testing of the the containment fan cooler system, and to delete “and/or recirculation” from the fifth paragraph, which descr room air filtration system.

2.2 Control Room Air Filtration System Mode of Operation

The mode of operation for the control room air filtration system during a toxic gas event, smoke event, or radiological accident was the recirculation mode, in which approximately 1840 cfm of control room air was recirculated through HEPA and charcoal filters.

The proposed mode of operation for the control room air filtration system during a radiological accident is the pressurized mode, in which approximately 2000 cfm of outside air is drawn in through HEPA and charcoal filters via booster fans and discharged into the control room envelope. The outside air would serve to pressurize the control room envelope to a positive pressure with respect to adjacent areas. This mode of operation is automatically initiated by a safety injection signal or a high radiation signal. This modification was discussed with NRC staff as part of the pilot plant program. The control room air filtration system was modified during the outage in the spring of 2000 to accommodate this new mode of operation and to ensure that the design basis single active failure criterion is met.

2.3 Reanalysis of radiological consequences for a large-break LOCA

As described in Section 6.4.1.9 of the IP2 Updated Final Safety Analysis Report (UFSAR), the HEPA and charcoal filters located in the containment air recirculation cooling and filtration system were designed to remove fission products from the containment atmosphere should they be released in the event of an accident. The filtration capacity of the current system is sufficient to reduce the concentration of fission products in the containment atmosphere following a loss of reactor coolant to levels ensuring that the 2 hour and 30-day thyroid doses will be within the limits of 10 CFR Part 100. The supporting radiological dose analyses have been reanalyzed in order to support modification of the plant design to remove the HEPA and charcoal filters located in the containment air recirculation cooling and filtration system. In addition to the removal of these filters, supporting systems, components, TSs, and maintenance operations would be eliminated.

With the deletion of the in-containment filters from the LOCA dose reanalysis, the main mechanisms for removal of elemental iodine and particulates from the containment atmosphere are sedimentation (for particulates), radioactive decay, and through operation of the containment sprays.

2.4 Fuel Handling Accident (FHA) Reanalysis

Consolidated Edison has also proposed to relax TS requirements during shutdown conditions. The premise is to take credit for the alternative source term and the normal decay of irradiated fuel, reanalyze the design-basis accident during shutdown conditions (i.e., the FHA), and thus conclude that neither building integrity nor the FHA mitigating systems are required to be OPERABLE during shutdown conditions.

In Amendment Number 102 for the Perry Nuclear Power Plant, Unit 1, dated March 11, 1999, the staff approved relaxation of TS requirements during fuel handling when an appropriately low decay heat generation rate has been achieved while the Perry licensee committed to continue to ensure an available containment during Cold Shutdown and Refueling Mode operation via administrative procedure. Modeled after Perry's original plant-specific proposal, the Nuclear Energy Institute proposed generic changes to the Improved Standard Technical Specifications via Technical Specification Task Force Traveler (TSTF) 51 for all four owners groups. The staff approved TSTF-51 in October 1999.

3.0 <u>EVALUATION</u>

3.1 <u>Large-Break LOCA Reanalysis</u>

After a large-break LOCA only two forms of radioactive iodine are removed from the containment atmosphere: elemental iodine and particulate iodine. In the licensee's analysis, elemental iodine was removed by the operation of the containment sprays only. It is a conservative approach because some elemental iodine will also be deposited on the containment walls, although this deposition mechanism is less effective.

The licensee used the following equation from Section 6.5.2 of the Standard Review Plan (SRP) for iodine removal by sprays:

$$?_s = 6K_gTF/Vd$$

where:

$?_s$ | spray removal rate constant for elemental iodine
K_g | gas phase mass transfer coefficient
T | time of spray drop fall
F | volumetric flow of sprays
V | containment sprayed volume
d | mass mean diameter of the spray drops

The equation is based on the information available at the time this section of the SRP was prepared (1988) and it incorporates many conservative assumptions. In evaluating $?_s$, the licensee used plant-specific parameters and a very conservative value for the gas phase mass transfer coefficient. The following coefficients were calculated:

$?_s = 22.5\ hr^{-1}$ for the injection phase
$?_s = 11.5\ hr^{-1}$ for the recirculation phase

These coefficients were further reduced. The $?_s$ for the injection phase was reduced to 20 hr^{-1} because it was the limiting value specified in the SRP, and the $?_s$ for the recirculation phase was reduced to 5.6 hr^{-1} because recirculation water would contain dissolved iodine, which could reduce absorption of iodine by the spray drops. The resulting values of $?_s$ had, therefore, a large degree of conservatism.

The decontamination factor assumed by the licensee for elemental iodine was 200, which represents the maximum value allowed by the SRP. The expected calculated decontamination factor, based on plant parameters, including a partition coefficient of 10,000, was considerably higher.

Two mechanisms were specified by the licensee in determining removal of particulate iodine from the containment atmosphere: sprays and sedimentation. The analytical expression for spray removal specified in Section 6.5.2 of the SRP, based on conservative assumptions, is:

$$?_p = 3hFE/2Vd$$

where: $?_p$ | spray removal rate constant for particulate iodine
h | drop fall height
E | single drop collection efficiency

The licensee calculated $?_p$ by substituting suitable plant parameters and using E/d = 10 m^{-1}, as specified in Section 6.5.2 of the SRP. The following coefficients were calculated:

$?_p$ = 4.5 hr^{-1} for the injection phase
$?_p$ = 2.28 hr^{-1} for the recirculation phase

There is no limiting value for decontamination factors for particulate iodine since it usually exists in the form of highly soluble compounds, such as Cesium Iodide (Cs I), and its solubility limit is never exceeded during an accident. However, the licensee arbitrarily limited particulate iodine removal by sprays to 98 percent, which corresponds to a decontamination factor of 50. This introduces an additional conservative factor into the licensee's calculation.

For removal of the particulate iodine above 98 percent, the licensee took credit for sedimentation of particles. This mechanism of particulate iodine removal was demonstrated to be effective for concentrations as low as 10 $\mu g/m^3$. The sedimentation coefficient assumed by the licensee was 0.1 hr^{-1}. The staff verified that this value corresponds to a removal rate for particulate iodine having a particle size of approximately 3 µm. For the larger particles, removal rates will be correspondingly higher. The corresponding limiting decontamination factor assumed by the licensee was 1,000. It is an arbitrary value because there is no actual limit for removal of particles by sedimentation. The only reason for having this limit is to establish a cut off point beyond which the rates of removal become so low that they would not have any practical significance.

The staff reviewed the containment modeling aspects of the licensee's revised LOCA analysis. One train of the containment spray system is assumed to operate following the large-break LOCA. For the determination of spray removal of elemental iodine independent of the use of spray additive, the licensee used the methodology identified in Section 6.5.2 of the staff's SRP. The staff finds the licensee's application of this methodology acceptable. Noble gases and organic iodine were removed primarily by radioactive decay. Particulate removal by spray was determined using the model described in SRP Section 6.5.2. The staff finds the licensee's application of this methodology acceptable.

During spray operation no credit is taken for sedimentation removal of particulates. It was assumed that containment spray operation is terminated when particulates are reduced to 2.0 percent of total particulates released to the containment. The licensee assumed the sedimentation removal coefficient to be 0.1 hr^{-1} based on the Industry Degraded Core Rulemaking Program Technical Report 11.3, "Fission Product Transport in Degraded Core Accidents," Atomic Industrial Forum, December 1983.

Lower bound (10 percentile) natural processes decontamination coefficients for radiological design-basis accidents were identified in Table 34 of NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," July 1996. The natural processes aerosol removal model in the staff's confirmatory analysis code RADTRAD is based on NUREG/CR-6189. Based on Table 34, the staff finds the sedimentation removal coefficient of 0.1 hr^{-1} to be reasonable.

Section 6.5.2 of the SRP specifies a pH value greater than or equal to 7.0 to assure continued retention of iodine in the sump solution. WCAP-14542, "Evaluation of the Radiological Consequences from a Loss of Coolant Accident at IP2 Using NUREG-1465 Source Term Methodology," dated July 1996, states that, for IP2, the mass of trisodium phosphate (TSP) required to provide an equilibrium sump solution pH of 7.0 is less than 4,000 pounds. To address the potential for long term generation of acids in the containment, this amount is doubled to 8,000 pounds. The initial containment spray will be boric acid solution from the refueling water storage tank which has a pH of approximately 4.5. As the initial spray solution and, subsequently, the recirculation solution comes in contact with the TSP, the TSP dissolves raising the pH of the sump solution to an equilibrium value between 7.0 and 9.5.

In request for additional information (RAI) 5, the staff asked the licensee to address the impact of increased particulate loading on the containment fan cooler units (e.g., fans, cooling coils, and drains). In response to the RAI, the licensee stated that the increased density in the containment atmosphere due to additional aerosols (radioactive and non-radioactive) is only 0.0003 lb/cu ft., which is insignificant when contrasted to the design value of 0.170 lb/cu ft for the coolers. The licensee concluded that particulates associated with the alternate source term are not expected to cause fouling or clogging of the containment fan cooler units because the condensing of moisture on the cooling coils will also cause any plate-out of particulates to be carried away with the condensate (since the particulate loading is so small). This conclusion was confirmed in a report by the staff, AEB-99-01, "Impact of Source Term Aerosols on Fan Cooler Performance for the IP2 Pilot Plant Application," dated March 22, 1999.

The AEB-99-01 report compared the amount of aerosols removed by the fan coolers with the amount of steam condensed by the fan coolers. If the amount of aerosols removed were small compared with the amount of water condensed, then it could be concluded that the aerosols may be washed off the fan cooler coils and have a small impact on fan cooler performance. The staff's analysis estimated that the aerosols would be removed by the fan coolers at a rate of about 150 kg/hr. During the first 2 hours, when nearly all of the aerosols would be removed, steam would condense in the fan coolers at a rate of about 80,000 kg/hr. Thus, the staff concludes that, for the IP2 pilot plant application, both radioactive and non-radioactive aerosols would have a small impact on fan cooler performance because ample water would be available to wash down the fan cooler coils.

3.2 Small-Break LOCA

The main difference between iodine removal from the containment atmosphere after large-break LOCAs and after small-break LOCAs (SBLOCAs) is that after an SBLOCA, containment sprays are not activated, and the only mechanisms assumed by the licensee for iodine removal are deposition of iodine by sedimentation and on the containment walls. For removal of elemental iodine by deposition, the licensee specified an analytical expression from Section 6.5.2 of the SRP, which is:

$$?_d = K_w A/v$$

where:

$?_d$ | deposition removal rate constant for elemental iodine
K_w | mass transfer coefficient
A | area available for deposition
V | total containment volume

In this expression, the mass transfer coefficient K_w is equal to 4.9 m/hr. This value is based on the experimental data from the Containment Systems Experimental Program. Using the values of the plant's parameters, the licensee calculated $?_d = 1.5\ hr^{-1}$. The corresponding decontamination factor was 200, which, similar to the case of iodine removed by sprays, represents the maximum value allowed by the SRP.

In determining particulate iodine removal by sedimentation, the licensee has demonstrated that the same values for removal coefficient and decontamination factors, i.e., $0.1 hr^{-1}$ and 1,000, respectively, are applicable to both large-and small-break LOCAs, because the accident sequence does not affect these factors.

3.3 FHA Reanalysis

The staff reviewed the containment modeling aspects of the licensee's revised FHA analysis. A fuel assembly is assumed to be dropped and damaged during refueling. Activity released from the damaged assembly is released to the outside atmosphere through either the containment purge system or the fuel-handling building ventilation system to the plant vent. No credit is taken for removal of iodine by filters, nor is credit taken for isolation of release paths. Although the containment purge will be automatically isolated on a purge line high radiation alarm, isolation is not modeled in the analysis. The activity released from the damaged assembly is assumed to be released to the environment over a 2-hour period. The staff finds these assumptions consistent with the guidance provided in NRC Draft Guide (DG)-1081.

The fuel assembly fission product inventory is based on the assumption that the subject fuel assembly has been operated at 1.7 times core average power (and thus has 1.7 times the average fuel assembly fission product inventory). The decay time used in the analysis is 100 hours. In accordance with the TSs, it is assumed that there is a minimum of 23 feet of water above the reactor pressure vessel flange and the spent fuel racks. With this water depth, the decontamination factor of 500 specified by DG-1081 for elemental iodine would apply. The licensee reduced the decontamination factor to 400 because the fuel rod pressure may exceed 1200 psig (but would be less than 1500 psig). The decontamination factor for organic iodine and noble gases was 1.0. The staff finds this to be a conservative application of the draft guidance and, therefore, acceptable.

Results of agency-sponsored probabilistic risk assessment studies for Zion, a plant of similar design to IP2, indicate that, during shutdown, the potential for core damage is least when the reactor vessel head is off (thus alleviating concerns regarding overpressurization of shutdown cooling system components) and the vessel water level is raised (thereby providing more time for mitigation of accident initiating events). As stated above, during refueling activities when fuel movement is taking place, TSs require a minimum water level of 23 feet of water above the reactor pressure vessel flange.

There are no TSs requiring containment integrity during shutdown other than the one involving fuel handling. Furthermore, no such TSs were implemented during the Shutdown Rulemaking process.

IP2 has outage management administrative controls in place for re-establishing containment closure consistent with plant conditions.

The FHA analysis was done with no credit taken for HEPA or charcoal filters in the control room air filtration system (Reference 3). Therefore, there is no requirement to have a TS requiring the control room air filtration system to be operable during refueling operations.

3.4 <u>Control Room Air Filtration System Mode of Operation</u>

The proposed revisions to TS Sections 4.5.E.2.a, 4.5.E.2.b, 4.5.E.2.c, 4.5.E.4.a, 4.5.E.5, and 4.5.E.6 involve changing the flow rate of the control room air filtration system from 1840 cfm to 2000 cfm to be consistent with the proposed modification. Also, the proposed revision to TS Section 4.5.E.4.b changes “recirculation” to “filtered intake” to be consistent with the proposed new pressurization mode of operation of the control room air filtration system.

The licensee’s reanalysis of the large-break LOCA, Steam Line Break and Steam Generator Tube Rupture accidents modeled the control room air filtration system in the proposed pressurization mode of operation, in which 1800 cfm of outside air would be drawn through HEPA and charcoal filters via booster fans and discharged into the control room envelope (References 1 and 3). The licensee stated that the design of the control room ventilation system in the proposed pressurization mode is to bring in approximately 2000 cfm of outside air and direct it through the HEPA and charcoal filters into the control room (Reference 4). The licensee also stated that the analysis takes into account 700 cfm of unfiltered leakage into the control room based on tracer gas testing (Reference 5). The final system design has an increase of about 15 percent in the intake of outside filtered air into the control room as compared to the original radiological analyses. However, the dose to personnel is affected more by the inleakage of unfiltered air than by the intake of filtered air, and the calculated dose to an operator in the control room is more than 20 percent below the acceptance criteria in 10 CFR 50.67, so this difference is judged to be acceptable. Sensitivity cases for control room dose that were analyzed for the large-break LOCA demonstrate this (Reference 1, p. 14.).

The proposed revision to TS Section 4.5.E.4.c changes “outside atmosphere” to “adjacent areas” to ensure that unfiltered inleakage is minimized during the proposed pressurized mode of operation of the control room air filtration system.

3.5 Environmental Equipment Qualification

The staff assessed the potential impact of the design changes (removal of containment fan cooler unit charcoal and HEPA filters) on the environmental qualification (EQ) of safety-related electrical equipment inside the containment. As discussed in Section 1.3.5, “Equipment Environmental Qualification,” of Draft Regulatory Guide DG-1081, the staff expects that if the current equipment EQ analyses are affected by a proposed plant modification associated with the alternate source term (AST) implementation, the EQ analyses having assumptions or inputs affected by the plant modification should be updated to address these impacts.

The licensee has confirmed that the proposed design change (removal of fan cooler unit charcoal and HEPA filters) will have no affect on the environmental qualification of safety-related electrical equipment. The current EQ radiation dose basis, as described in the IP2 EQ Program Plan, takes no credit for removal mechanisms inside containment such as sprays or filters. Based on its review, the staff concludes that the removal of the containment fan cooler unit charcoal and HEPA filters will have no impact on the EQ of safety-related electrical equipment at IP2. As a result, the current EQ analyses do not need to be updated with the proposed design change associated with the implementation of AST at IP2.

3.6 Substitution of “monthly” for “per 31 days”

The proposed revision to TS Section 4.5.E.1 changes “per 31 days” to “monthly.” This change is administrative in nature and does not change the substance of the requirement, and is, therefore, acceptable.

3.7 Radiological Dose Analysis for FSAR Accidents

The licensee performed calculations of the potential radiological doses associated with those aspects of the proposed TS amendment involving changes in containment air filtration, control room air filtration and refueling operations at IP2. These results were submitted for staff review and approval. The licensee performed re-analyses of a select number of the IP2 FSAR accidents and analyses of accidents not currently analyzed in the Unit 2 FSAR. The re-analyses were to demonstrate the acceptability of (1) the removal of the in-containment charcoal adsorbers and HEPA filters, (2) the conversion of the control room emergency ventilation system from an isolation and recirculation mode of operation to an isolation and pressurization mode of operation, and (3) changes in fuel handling operation to allow the movement of fuel within 100 hours rather than 174 hours following reactor shutdown, and fuel movement with either the equipment hatch or personnel air locks open. In addition to the assessments that supported the proposed changes to TSs and operations, the licensee also submitted assessments of the consequences of postulated accidents that were independent of the proposed changes in TSs and operations.

The licensee’s assessment to demonstrate the acceptability of the proposed changes in TSs and operations implemented the use of the alternate source term (AST) in NUREG-1465. It was the licensee's intent to demonstrate that such changes could be made without the dose limits of 10 CFR 50.67 being exceeded. The accidents that the licensee analyzed and the appropriate offsite NRC guideline doses for each of these accidents (except for doses to control room operators) are as follows:

1. Large-Break Loss-of-Coolant Accident (LOCA)
   - 25 rem TEDE
2. Main Steam Line Break
   - pre-existing spike case - 25 rem TEDE
   - accident-initiated spike case - 2.5 rem TEDE
3. Steam Generator Tube Rupture
   -pre-existing spike case - 25 rem TEDE
   - accident-initiated spike case - 2.5 rem TEDE
4. Locked Rotor
   - 2.5 rem TEDE
5. Fuel Handling
   - 6.25 rem TEDE
6. Rod Ejection
   - 6.25 rem TEDE
7. Small-Break LOCA
   - 25 rem TEDE

The control room operator dose limit for any of these accidents is 5 rem TEDE.

It was the licensee's desire to have a full implementation of the AST. Accidents that were unaffected by the change in TSs and operations but were re-assessed included the locked rotor, steam generator tube rupture (SGTR) and the main steamline break (MSLB) accidents. Other accidents that were analyzed included the rod ejection and small-break LOCA accidents. In addition to the re-calculation of postulated accident releases and associated doses, the licensee also performed a re-assessment of the atmospheric dispersion parameters for the control room dose estimates associated with previously analyzed accidents. In the assessment of the consequences of these accidents, the licensee utilized much of the guidance contained in Draft Regulatory Guide (DG)-1081.

The staff performed confirmatory calculations for the spectrum of accidents analyzed by the licensee, and a confirmatory evaluation of the licensee's atmospheric dispersion assessment for the revised control room values. Doses were calculated for individuals located offsite at the exclusion area boundary (EAB), and at the low population zone (LPZ), and onsite for the control room operators.

The IP2 control room was originally designed to isolate normal ventilation and to operate in the emergency mode with the air within the control room filtered and re-circulated. The control room has now been modified to isolate normal ventilation and to bring into the control room, through the control room emergency ventilation system charcoal and HEPA filters, approximately 2000 cfm of outside air. A safety injection signal or a high radiation signal automatically isolates normal ventilation for the control room and automatically initiates operation of the control room emergency ventilation system. The time that expires before operation of the control room emergency ventilation system begins varies from accident to accident. The acceptability of the control room operator doses was based upon the control room emergency ventilation system operating in pressurization mode and within the time frame specified in the Tables associated with the particular accident.

The following sections provide the staff's assessment of the potential consequences of the above postulated accidents and the licensee's re-assessment of atmospheric dispersion.

3.7.1 Analyzed Accidents

3.7.1.1 Large-Break LOCA

The licensee assessed the consequences of a large-break LOCA utilizing the NUREG-1465 source terms. In an assessment incorporating NUREG-1465 source terms, it is assumed that a large-break LOCA is a reasonable initiation of the release of gap activity if the plant has not been approved for leak before break (LBB) operation. For plants which have received LBB approval, a small-break LOCA would more accurately model the release timing. With the postulated pipe rupture, it is anticipated that the initial radioactivity release to containment will consist of the radioactivity contained within reactor coolant. The duration of this release is assumed to be 25 seconds for a Westinghouse PWR such as IP2. The gap activity release phase begins when fuel cladding failure commences. In NUREG-1465 it was stated that the significant fission product releases from the bulk of the fuel were estimated to commence no earlier than 30 minutes after the onset of the accident. This release of gap activity was assumed to occur over 30 minutes in accordance with NUREG-1465. The in-vessel release phase (see NUREG-1465) occurs following the release of gap activity, and is 1.3 hours in duration. Table 3.7.1.1-1 presents the duration of each release period and the fraction of the total core inventory released during each period as a function of radionuclide grouping.

Table 3.7.1.1-1 Element Release Fraction as a Function of Release Period

RADIONUCLIDE GROUP	GAP RELEASE (0.5 Hours)	EARLY IN-VESSEL (1.3 Hours)
Noble Gases (Xe, Kr)	0.05	0.95
Halogens (I, Br)	0.05	0.35
Alkalide Metals (Cs, Rb)	0.05	0.20
Tellurium Group (Te, Sb, Se)	0	0.05
Ba, Sr	0	0.02
Noble Metals (Ru, Rh, Pd, Mo, Tc, Co)	0	0.0025
Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am)	0	0.0002

Cerium Group (Ce, Pu, Np)	0	0

The licensee calculated the potential consequences of a postulated large-break LOCA to the control room operators and to individuals located offsite at the EAB and LPZ.

In the licensee's assessment of the consequences associated with emergency core cooling system (ECCS) leakage, the licensee provided proprietary and non-proprietary calculations to demonstrate the fraction of ECCS leakage that would become airborne (Reference 5). The licensee estimated that 5.5 percent of the ECCS leakage would initially become airborne. However, the licensee also estimated that the percent of the leakage becoming airborne would decrease over the course of the accident (average of about 1 percent over the course of the accident). The licensee also took no credit for treatment of the ECCS leakage by the primary auxiliary building ventilation system (PABVFS). The staff found the constant enthalpy method acceptable, but used the 5.5 percent value for the entire course of the accident as a conservatism.

In the licensee's analysis, it was assumed that the containment source term for elemental and particulate forms of iodine was reduced by sprays. In addition, it was assumed that the elemental form of iodine was also subject to removal via sedimentation. The licensee assumed that the sprayed and unsprayed regions were mixed by the containment cooling fans.

The licensee assumed varying removal rates by sprays for elemental and particulate forms of iodine. During the injection phase, the spray removal coefficients were 20/hr and 4.5/hr, respectively. During the recirculation phase, the coefficients were 5.6/hr and 2.28/hr, respectively. The sedimentation removal coefficient was 0.1/hr. The licensee's assessment established a DF (decontamination factor) limit on elemental iodine of 200, on particulates of 50 during the spray removal operation, and 1000 total for particulates.

Details on the assumptions utilized by the staff for the large-break LOCA evaluation are presented in Table 3.7.1.1-2. The TEDE dose at the EAB, LPZ and to the control room operator are presented in Table 3.7.2-1. The doses were found to be acceptable.

3.7.1.2 Main Steam Line Break

As noted previously, the licensee proposed full implementation of the AST. Consequently, the MSLB accident was re-analyzed (Reference 3). The reevaluation of the MSLB involved two cases. One case assumed the accident occurred following an iodine spike, referred to as the pre-existing spike case. The second case assumed that the MSLB resulted in the initiation of an iodine spike, referred to as the accident-initiated spike. In both cases, each of the steam generators was assumed to have a 0.3 gpm primary to secondary leak.

For case one, reactor coolant concentration was assumed to be 60 µCi/gm of dose equivalent ^{131}I. For the second case, reactor coolant concentration was assumed to be at 1 µCi/gm dose equivalent ^{131}I. In both cases, the secondary system activity was assumed to be at 0.15 µCi/gm dose equivalent ^{131}I. For case two, an iodine spike was assumed to result in the release of iodine from the fuel gap to the reactor coolant at a rate that is 500 times the normal iodine release rate. As a result of the MSLB, no failed fuel was assumed to occur in either case.

For both analyses, it was assumed that all of the primary to secondary leakage to the faulted steam generator would be released to the environment with no credit for iodine and particulate retention in the steam generator. The entire liquid inventory in the steam generator with the steamline break, referred to as the faulted steam generator, was assumed to be steamed off and all of the iodine initially in the steam generator was assumed to be released to the environment. After the faulted steam generator was isolated, it was assumed that primary to secondary leakage to the intact steam generators would continue at a rate of 0.3 gpm per steam generator. Because offsite power is assumed to be lost, the main condenser was unavailable for steam dump and cooling of the reactor core must occur through the use of the safety valves.

Any noble gas that would be carried over to the secondary side through primary to secondary leakage would be assumed to be immediately released to the environment. At 42 hours after the accident, the RHR system is assumed to be capable of all decay heat removal and there would be no further steam releases to the environment from the secondary system. The licensee assumed that the activity releases from the faulted steam generator continued until the primary coolant temperature was reduced to less than 212 ^{o}F at 70 hours.

The licensee assumed that the duration of the iodine spike was 5 hours based upon gap activity of iodine being 12 percent of the total core activity. The staff considered the limitation of iodine spiking should be longer than 5 hours. The staff recommends that the value be changed in future analyses. The amount of iodine in the gap, core-wide, is significantly less than 12 percent. Utilization of a value of 12 percent is only appropriate for the limiting fuel assembly in an accident such as the fuel handling accident and is conservative.

Details on the assumptions utilized by the staff in the performance of their confirmatory calculations are presented in Table 3.7.1.2-1. The TEDE dose at the EAB, LPZ and to the control room operator are presented in Table 3.7.2-1. The doses were found to be acceptable.

3.7.1.3 Steam Generator Tube Rupture (SGTR) Analysis

As noted previously, the licensee submitted re-analyses of postulated accidents that were submitted as part of the licensee's full implementation of the AST. The previous section presented the results of one such implementation. This section provides the results of a second, the SGTR . The following provides the results of the staff's assessment of the licensee's re-analysis of the SGTR accident (Reference 3).

The licensee evaluated the consequences of a postulated SGTR accident. For the SGTR, primary to secondary leakage was assumed to be occurring at the TS rate of 0.3 gpm/steam generator from each of the four steam generators. In addition, primary to secondary leakage would occur through the ruptured tube into the ruptured steam generator.

The licensee analyzed two cases. The first assumed a pre-existing spike occurred prior to the SGTR. For the pre-existing spike case, the reactor coolant iodine specific activity was assumed to be at 60 µCi/gm of dose equivalent ^{131}I. The secondary coolant iodine specific activity was assumed to be at the secondary coolant specific activity equilibrium value of 0.15 µCi/g of dose equivalent ^{131}I.

The second case, referred to as the accident-initiated spike case, assumed the SGTR event itself initiated an iodine spike concurrent with the accident. Immediately prior to the accident, the reactor coolant was assumed to be at a reactor coolant activity level of 1 µCi/gm of dose equivalent ^{131}I and secondary system activity was again assumed to be at 0.15 µCi/gm dose equivalent ^{131}I. The SGTR was assumed to initiate an iodine spike that would result in a release of iodine from the fuel gap to the reactor coolant at a rate that is 335 times the normal iodine release rate necessary to maintain the reactor coolant activity level at 1 µCi/gm of dose equivalent ^{131}I. The licensee's submittal indicated that a SGTR accident would not result in any melted fuel being released to the reactor coolant.

For both cases, it was assumed that the primary to secondary leak in the intact steam generators remained at 0.3 gpm per steam generator for the duration of the accident. For both cases, it was assumed that offsite power was lost and the main condenser was unavailable for the steam dump. The licensee's assessment assumed that break flow continued for 0.5 hour after the tube ruptures and that the spike lasted for 7.5 hours.

Table 3.7.1.3-1 presents the assumptions utilized by the staff in their assessment of an IP2 SGTR. The potential dose consequences of a SGTR accident at IP2 are presented in Table 3.7.2-1. The doses were found to be acceptable.

3.7.1.4 Locked Rotor

The existing licensing basis for IP2 does not include an assessment of the radiological consequences of a locked rotor accident. The licensee indicated that they incorporated this event for completeness in their full implementation of the AST as this is one of the accidents in which fuel damage is postulated.

The licensee assumed an instantaneous seizure of a reactor coolant pump rotor, which rapidly reduces reactor coolant flow through the affected loop (Reference 1). Fuel clad damage is assumed to occur as a result of this event. Due to the pressure differential between the primary and secondary side and assumed steam generator tube leakage, fission products are discharged from the primary to secondary side. A portion of this radioactivity is discharged through either the atmospheric relief valves or safety valves. In addition, iodine activity is contained in the secondary coolant prior to the accident and some of this activity is released to the atmosphere as a result of steaming from the steam generators following the accident.

The licensee's analysis assumed a pre-existing spike in reactor coolant activity prior to the locked rotor event. Such a condition would raise the reactor coolant activity level to 60 µCi/g of dose equivalent ^{131}I. The noble gas and alkali metals group activity concentrations in reactor coolant were based upon 1 percent failed fuel. The licensee's assessment incorporated a secondary coolant activity level of 0.1 µCi/g of dose equivalent ^{131}I. However, this value is inconsistent with existing TSs which limit secondary coolant to 0.15 µCi/g of dose equivalent ^{131}I. As a result of the locked rotor accident, the licensee postulated that no more than 2.5 percent of the fuel rods would undergo DNB (departure from nucleate boiling). However, the analysis that they performed assumed that 5 percent of the fuel rods experienced DNB.

In the analysis performed by the licensee, all of the iodine released to reactor coolant was assumed to be elemental and, after the release to the environment, 97 percent of the iodine was considered elemental and the remainder organic. This was consistent with the model in DG-1081. Activity would be released to the environment as a result of the leakage of primary coolant to the secondary side at the TS value of 0.3 gpm/steam generator and steaming from the secondary side to the environment. The RHR system was assumed to be placed into service at 42 hours following the accident and there would be no further releases to the environment. The licensee assumed no credit for iodine removal for any steam released to the condenser prior to the reactor trip and concurrent loss of offsite power. All noble gas activity carried over via primary to secondary leakage through the steam generator tubes was assumed to be immediately released to the environment. A partition factor of 0.01 µCi/g steam per µCi/g water was assumed both for iodine and alkali metal activity in the steam generators.

The staff performed independent calculations of the consequences of the locked rotor accident. Table 3.7.1.4-1 presents the assumptions utilized by the staff in their assessment. The staff's assessment of the potential dose consequences of a locked rotor accident are presented in Table 3.7.2-1. The doses were found to be acceptable.

3.7.1.5 Fuel Handling Accident

The licensee provided a re-assessment of a fuel handling accident (References 1 and 3). It was assumed that a fuel assembly was dropped and damaged during a refueling operation. Activity released from the damaged assembly was assumed to be released to the outside atmosphere through either the containment purge system or the fuel handling ventilation system. It was assumed that the control room HVAC remained in its normal operating mode. The assessment, which was performed by the licensee, assumed that the containment personnel air locks and the equipment hatch were open to atmosphere. The analyses were performed in this manner to justify refueling operations with the control room emergency ventilation system off or inoperable and to justify allowing the containment personnel air locks and the equipment hatch to be open.

The licensee assumed that the dropping of a spent fuel assembly would result in damage to one entire fuel assembly and the release of the gap fission products to the environment through the penetration room filtration system. The gap inventory was assumed to consist of iodides, noble gases, and alkali metals (cesium and rubidium). The damaged fuel assembly was assumed to have been operated at 1.7 times core average power and thus, had 1.7 times the average assembly’s fission product inventory. As part of the proposed TS change, the licensee proposed that the time allowed between reactor shutdown and fuel movement be decreased from 174 hours to100 hours. Consequently, the licensee’s re-analysis assumed that the dropped fuel assembly had decayed for 100 hours rather than 174 hours.

The licensee assumed that the chemical form of iodine in the fuel gap was 99.75 percent elemental and 0.25 percent organic. This was consistent with the guidance of draft RG (DG)-1081. Due to TS requirements, the licensee assumed that there was 23 feet of water over the damaged assembly and that this depth of water provided a decontamination factor (DF) of 500 for elemental iodine. However, the licensee’s re-analysis limited the pool DF to 400 to account for the possibility of fuel rod pressure exceeding 1200 psig. The DF for organic iodine and noble gases was assumed to be 1.

The licensee took no credit for the removal of iodine by any ESF filter unit even though a containment purge high radiation signal would isolate the purge release from an accident occurring within containment. The licensee utilized this assumption to demonstrate that the equipment hatch and personnel airlock could remain open and that acceptable doses would still result in the event of such an accident. The licensee's analysis also assumed that the control room HVAC system was operating in its normal mode. This assumption addressed the possibility of maintenance being performed on the control room emergency ventilation system adsorbers at the time of the accident.

The staff has performed an independent calculation of a fuel handling accident. Table 3.7.1.5-1 contains details of the assumptions utilized by the staff in their assessment. The results of the staff's calculations are presented in Table 3.7.2-1. NUREG-1465 gave a value of 0.05 for the gap fraction of iodine-131 released for a fuel handling accident; the licensee used this value. The value of the gap fraction to be included in the final version of RG 1.183 is still under discussion by the staff; therefore, the staff used a bounding value of 0.08 for the gap fraction. The doses were found to be acceptable and justified operation with the containment personnel air lock and the containment equipment hatch open.

3.7.1.6 Rod Ejection

The licensee assumed that a mechanical failure of a control rod mechanism pressure housing resulted in the ejection of a rod cluster control assembly and drive shaft (Reference 1). As a result of such an accident, fuel clad damage and a small amount of fuel melt would occur. Due to the pressure differential between the primary and secondary side, primary coolant would be discharged to the secondary side. A portion of this radioactivity would be discharged to the environment either through the atmospheric relief valves or main safety valves. Iodine and alkali metals group activity is contained in secondary coolant prior to the accident and some of this activity was also assumed to be released to the atmosphere as a result of steaming the steam generators following the accident. Radioactive reactor coolant would also be discharged to the containment via a spill from the opening in the reactor vessel head. A portion of this radioactivity would be released to the environment via containment leakage.

The licensee determined that in the event of a rod ejection, less than 10 percent of the fuel rods would undergo DNB. However, their analysis assumed that 10 percent of the fuel rods in the core would suffer sufficient damage to release all of their gap activity. For this assessment, the licensee assumed that 5 percent of the core activity of iodine, noble gases and alkali metals was contained in the gap.

A small fraction of the fuel in the failed rods was assumed to melt (0.25 percent). The licensee assumed that all of the alkali metal and noble gases associated with the melted fuel and 50 percent of the iodine would be released. The licensee's analysis assumed that a pre-existing spike existed in reactor coolant activity prior to the rod ejection. Such a condition would raise the reactor coolant activity level to 60 µCi/g of dose equivalent ^{131}I. The noble gas and alkali metals group activity concentrations in reactor coolant were based upon 1 percent failed fuel. The licensee's assessment incorporated a secondary coolant activity level of 0.1 µCi/g of dose equivalent ^{131}I. However, this value is inconsistent with existing TSs, which limit secondary coolant activity to 0.15 µCi/g of dose equivalent ^{131}I. The licensee's future evaluations of this accident should be based upon the TS value of 0.15 µCi/g.

In the analysis performed by the licensee, all of the iodine released to reactor coolant was assumed to be elemental and, after release to the environment, 97 percent of the iodine was considered elemental and the remainder organic. This is consistent with the model in DG-1081. Activity would be released to the environment as a result of the leakage of primary coolant to the secondary side at the TS value of 0.3 gpm/steam generator and steaming from the secondary side to the environment. The licensee assumed no credit for iodine removal for any steam released to the condenser prior to the reactor trip and concurrent loss of offsite power. All noble gas activity carried over via primary to secondary leakage through the steam generator tubes was assumed to be immediately released to the environment. A partition factor of 0.01 µCi/g steam per µCi/g water was assumed both for iodine and alkali metal activity in the steam generators.

For the containment leakage pathways, the licensee assumed that the iodine released from the fuel was 95 percent particulate, 4.85 percent elemental and 0.15 percent organic. Containment leakage was assumed to be 0.1 percent/day for the first 24 hours following the accident and 0.05 percent/day for the remainder of the accident. For the containment leakage pathway no credit was assumed for sedimentation or plateout onto containment surfaces nor for containment spray operation, which would remove airborne particulates and elemental iodine.

The staff has performed a calculation of the dose consequences of a rod ejection accident. Table 3.7.1.6-1 presents the assumptions utilized by the staff in their assessment. The doses that were calculated for this accident are presented in Table 3.7.2-1. The doses were found to be acceptable.

3.7.1.7 Small-Break LOCA

The licensee performed an analysis of the potential consequences of a small-break LOCA (SBLOCA) (Reference 1). In their assessment, they assumed that a break occurred that resulted in substantial fuel damage in the reactor core but that the damage was insufficient to result in a containment pressure that would activate containment sprays. The licensee's assessment assumed that the entire core's gap activity would be released. Two potential pathways for transport to the environment were evaluated. In both cases the gap activity was assumed to be released to primary coolant.

For the one case, all of the activity released to primary coolant was assumed to be released into containment. In containment, the particulate and elemental forms of iodine were assumed to be removed by sedimentation and deposition, respectively. No removal mechanisms were assumed for the alkali metals or the noble gases or the organic form of iodine. For the second case, all of the activity was assumed to be released as a result of the removal of the reactor core's decay heat through the steam generators. For this case, the gap activity in primary coolant is assumed to be released to the secondary side as a result of primary to secondary leakage. The release of the secondary side steam in order to remove decay heat from the core would be a means for transporting radioactivity to the environment. For each case, all of the gap activity was assumed to be released by the assumed transport pathway.

For the secondary side release pathway, it was assumed that the chemical form of iodine released from the secondary side was 97 percent elemental and 3 percent organic. Primary to secondary leak rate was assumed to be 1.2 gpm total for all four steam generators. The licensee assumed no credit for iodine removal for any steam released to the condenser prior to the reactor trip and the concurrent loss of offsite power. All noble gas activity carried over to the secondary side was assumed to be released immediately to the environment. For iodine and alkali metals, a partition factor of 0.01 was assumed for the activity in the steam relative to the activity in the water.

The staff has performed a calculation of the dose consequences of a SBLOCA. Table 3.7.1.7-1 presents the assumptions utilized by the staff in their assessment. The doses that were calculated for this accident are presented in Table 3.7.2-1. The doses were found to be acceptable.

3.7.1.8 Atmospheric Relative Concentrations

The relative concentration (X/Q) values used by the licensee for the exclusion area boundary and low population zone dose assessment are the values presented in the IP2 Updated Final Safety Analysis Report. They were not reviewed by the staff as a part of this amendment.

The licensee used 3 years of onsite meteorological data, 1995 through 1997, to estimate X/Q values for the control room dose assessments. The licensee confirmed that the data were collected under the guidelines specified in RG 1.23, "Onsite Meteorological Programs." The tower area was maintained to be free of obstructions. Quality assurance measures such as semi-annual channel calibration checks and weekly operational checks were performed to ensure data quality and identify any problems, which were addressed upon discovery. Data recovery for the 3-year period exceeded 99 percent and, therefore, exceeded the minimum 90 percent recovery rate guideline set forth in RG 1.23. The staff performed a general review of the data and found them acceptable for use in this dose assessment.

The licensee used the ARCON96 methodology described in NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wake," with two modifications, to estimate the X/Q values used in the control room dose assessments. Calculations were made for postulated releases from four locations: the containment outside surfaces, the side of the auxiliary boiler feedwater building nearest the control room intake, the auxiliary feedwater vents, and the Unit 2 plant vent atop the containment building.

The licensee calculated control room X/Q values for the 0-2 hour, 2-8 hour, 8-24 hour, 1-4 day and 4-30 day time periods. The staff utilized the licensee's values for X/Q for each period except that the staff assumed the 0-2 hour X/Q value for the entire 0-8 hour period.

The two modifications mentioned above resulted from discussions with the staff and are as follows. When estimating the initial diffusion coefficients for the two assumed area sources, the containment building surface and side of the auxiliary boiler feedwater building, the licensee divided both the assumed height and width of the area of release by a factor of 6. In addition, calculations for all four postulated locations were made as ground level releases assuming no vertical momentum. These modifications result in an increase in estimated dose. The licensee provided the revised X/Q values by letter dated April 13, 2000 (Reference 5). The values utilized by the staff are listed in Table 3.7.1.8-1. Based on the conservatism described above, the staff finds the X/Q values acceptable for use in this dose assessment.

3.7.2 Conclusions for Radiological Dose Analyses

The staff has assessed those accidents for which the licensee proposed full implementation of the AST and those which were utilized to support the proposed TS amendment involving changes in containment air filtration, control room air filtration, and refueling operations at IP2. The staff concluded that the licensee's atmospheric dispersion assessment was acceptable.

Based on the above, the staff finds this revised FHA analysis supports the proposed TSs for refueling with containment personnel access doors open during refueling operations, and supports allowing core alterations to begin when the reactor has been subcritical for 100 hours.

Based on the foregoing, the licensee's proposed changes in fuel handling operation, containment filter TSs, and control room design changes are acceptable.

Table 3.7.1.1-2 Assumptions for LOCA Analysis

Parameter	Value
Core Thermal Power (MWt)	3216.5
Activity Released to the Containment	Refer to Table 3.7.1.1-1
Elemental Iodine Spray Removal Rate (1/hr)	
Injection Phase	20
Recirculation Phase	5.6
Particulate Iodine Spray Removal Rate (1/hr)	
Injection Phase	4.5
Recirculation Phase	2.28
DF Limitation	
Elemental Iodine	200
Particulate Iodine (During spray removal)	50
Particulates (total)	1000
Iodine Species (fraction)	
Elemental	0.485
Particulate	0.95
Organic	0.015
Activity Released to Sump (fraction)	
Iodine	0.5
Noble Gases	0.0
Containment Free Volume (ft^3)	2.61E6
Leakage Rate (percent/day)	
0-24 hours	0.10
> 24 hours	0.05
Containment Fan Coolers Flow Rate	
Fan (cfm)	64,500
Number of Fans Operating	3

Table 3.7.1.1-2 Assumptions for LOCA Analysis (cont.)

Parameter	Value
Sump Liquid Mass (lb)	1.78E6
Fraction of Containment Unsprayed	0.2
Recirculation Loop Leakage Rate (gpm)	4
Minimum Time to External Recirculation (hr)	24
Time to Initiate Sprays (seconds)	80
Time to Switch to Recirculation Spray Operation (minutes)	20
Control Room Free Volume (ft^3)	102,400
Filtered Emergency Intake Flow (cfm)	1800
Control Room Emergency Intake Filter System Efficiency (percent)	
Elemental and Organic	90
Particulate	99
Control Room Unfiltered Air Infiltration Rate (cfm)	700
Control Room Occupancy Factors	
0-1 day	1.0
1-4 days	0.6
4-30 days	0.4

Table 3.7.1.1-2 Assumptions for LOCA Analysis (cont.)

Parameter	Value
Atmospheric Dispersion Factors (sec/m^3)	
EAB	7.5E-4
LPZ	
0-8 hours	3.5E-4
8-24 hours	1.2E-4
1-4 days	4.2E-5
4-30 days	9.3E-6
Control Room	
0-8 hours	3.8E-4
8-24 hours	1.1E-4
1-4 days	8.3E-5
4-30 days	7.0E-5
Breathing Rates (m^3/sec)	
Offsite	
0-8 hours	3.47E-4
8-24 hours	1.75E-4
1-30 days	2.32E-4
Control Room	3.47E-4

Table 3.7.1.2-1 Assumptions for Main Steamline Break Accident

Iodine Partition Factor	
Intact Steam Generator	0.01
Faulted Steam Generator	1
Steam Release from Intact SGs (lbs)	
0-2 hours	
2-8 hours	6.0E5
8-24 hours	1.1E6
24-40hours	1.5E6
40-42 hours	1.3E6
> 42 hours	1.6E5
	none
Duration of Plant Cooldown (hrs)	42
Chemical Form of Release	
Organic (percent)	3
Elemental (percent)	97
Breathing Rate	
0-8 hours (m^3/sec)	3.47E-4
8-24 hours (m^3/sec)	1.75E-4
> 24 hours (m^3/sec)	2.32E-4
Primary coolant concentration @60 µCi/g of dose equivalent ^{131}I. (µCi/g)	
^{131}I	46.5
^{132}I	15.9
^{133}I	36.1
^{134}I	9.46
^{135}I	36.1
Mass of Primary Coolant (g)	2.37E8
Secondary Coolant Mass/Steam Generator (g)	3.19E7- 5.83E7
Primary Coolant DE ^{131}I Concentration (µCi/g)	
Maximum Instantaneous Value	60

Table 3.7.1.2-1 Assumptions for Main Steamline Break Accident (cont.)

48 Hour Value	1.0
Secondary Coolant DE ^{131}I concentration (µCi/g)	0.15
Equilibrium Release Rate from Fuel for a Spiking Factor of 500 times the Release Rate for 1 µCi/g of Dose Equivalent ^{131}I (Ci/hr)	
^{131}I	7,420
^{132}I	11,938
^{133}I	7,863
^{134}I	16,086
^{135}I	12,857
Control Room	
Free Volume (ft3)	1.02E5
Normal Ventilation Flow (cfm)	920
Time to Initiate Control Room Emergency Ventilation System (s)	90
Makeup Filter Efficiency for elemental and organic forms of Iodine (percent)	90
Makeup Air Filtration Rate (cfm)	1800
Unfiltered Air Infiltration Rate (cfm)	700
Occupancy Factors	
0-1 day	1.0
1-4 days	0.6

Table 3.7.1.2-1 Assumptions for Main Steamline Break Accident (cont.)

Atmospheric Dispersion Factors (sec/m^3)	
Control Room	
0-8 hours	1.09E-3
8-24 hours	4.99E-4
1-4 days	3.86E-4
4-30 days	2.99E-4
Atmospheric Dispersion Factors (sec/m^3)	
EAB	7.5E-4
LPZ	
0-8 hours	3.5E-4
8-24 hours	1.2E-4
1-4 days	4.2E-5
4-30 days	9.3E-6
Spiking Factor for Accident Initiated Spike	500
Breathing Rate (m^3/sec)	3.47E-4

Table 3.7.1.3-1 Assumptions for Steam Generator Tube Rupture

Iodine Partition Factor	0.01
Steam Release from Defective Steam Generator	
0-0.5 hours (lbs)	7.3E4
>0.5 hours (lbs)	0
Steam Release from Intact SGs (lbs)	
0-2 hours	5.14E5
2-8 hours	1.04E6
8-42 hours	2.87E6
Estimated Break Flow to Faulted Steam Generator (lbs)	1.28E5
Primary to Secondary Leak Rate (gpm/Steam Generator)	0.3
Time to Isolate Faulted Steam Generator (sec)	1800
Flashing Fraction	0.13
Scrubbing Fraction	0
Primary Bypass Fraction for Intact SGs	0
Duration of Plant Cooldown (hrs)	42
Chemical Form of Release	
Organic (percent)	3
Elemental (percent)	97
Breathing Rate	
0-8 hours (m^3/sec)	3.47E-4
8-24 hours (m^3/sec)	1.75E-4
> 24 hours (m^3/sec)	2.32E-4
Primary coolant concentration of 60 µCi/g of dose equivalent ^{131}I.	
Pre-existing Spike Value (µCi/g)	46.5
^{131}I	15.9
^{132}I	36.1
^{133}I	9.46
^{134}I	36.1
^{135}I	

Table 3.7.1.3-1 Assumptions for Steam Generator Tube Rupture (cont.)

Mass of Primary Coolant (g)	2.37E8
Secondary Coolant Mass/Steam Generator (g)	3.19E7- 5.83E7
Primary Coolant DE ^{131}I concentration (µCi/g)	
Maximum Instantaneous Value	60
48 Hour Value	1.0
Secondary Coolant DE ^{131}I concentration (µCi/g)	0.15
Technical Specification Limits for the primary to secondary leak rate.	
Primary to secondary leak rate, any Steam Generator (gpm)	0.3
Primary to secondary leak rate, total (gpm)	1.2
Letdown Flow Rate (gpm)	120
Equilibrium Release Rate from Fuel for a Spiking Factor of 335 times the Release Rate for 1 µCi/g of Dose Equivalent ^{131}I (Ci/hr)	
^{131}I	4,972
^{132}I	7,999
^{133}I	5,268
^{134}I	10,777
^{135}I	8,614
Control Room	
Free Volume (ft^3)	1.02E5
Normal Ventilation Flow (cfm)	920
Time to Initiate Control Room Emergency Ventilation System (s)	90

Table 3.7.1.3-1 Assumptions for Steam Generator Tube Rupture (cont.)

Makeup Filter Efficiency for elemental and organic forms of Iodine (percent)	90
Makeup Air Filtration Rate (cfm)	1800
Unfiltered Air Infiltration Rate (cfm)	700
Occupancy Factors	
0-1 day	1.0
1-4 days	0.6
Atmospheric Dispersion Factors (sec/m^3)	
Control Room	
0-8 hours	1.09E-3
8-24 hours	4.99E-4
1-4 days	3.86E-4
4-30 days	2.99E-4
EAB	7.5E-4
LPZ	
0-8 hours	3.5E-4
8-24 hours	1.2E-4
1-4 days	4.2E-5
4-30 days	9.3E-6
Spiking Factor for Accident Initiated Spike	335

Table 3.7.1.4-1 Assumptions for Locked Rotor Accident

Parameter	Value
Core Thermal Power Level (MWt)	3216.5
Duration of Plant Cooldown by Secondary System (hr)	42
Gap Fraction	0.05
Failed Fuel Rods (percent)	5
Primary to Secondary Leak Rate (gpm/SG)	0.3
Iodine Partition Factor in Steam Generators	0.01
Steam Released from SGs (g/min)	
0-2 hours	2.27E6
2-8 hours	1.39E6
8-24 hours	7.09E5
24-42 hours	6.14E5
Primary Coolant Mass (g)	2.37E8
Secondary Coolant Mass (g)	1.275E8
Primary to Secondary Leak Rate (g/min)	4,550
Iodine Form (steam generator steaming path)	
Prior to release to atmosphere	100 percent Elemental
Following release to atmosphere	97 percent Elemental 3 percent Organic
Primary Coolant Activity Level - Dose Equivalent ^{131}I (µCi/g)	
Pre-existing Spike	60
Primary Coolant Activity Level Other Nuclides	Based upon operation with 1 percent fuel defects
Secondary Coolant Activity Level - Dose Equivalent ^{131}I (µCi/g)	0.15
Secondary Coolant Activity Level Other Nuclides	10 percent of Primary Coolant Activity

Table 3.7.1.4-1 Assumptions for Locked Rotor Accident (cont.)

Control Room Operating Parameters	Refer to Table 3.7.1.1-2
Offsite ?/Q Values	Refer to Table 3.7.1.1-2
Breathing Rates	Refer to Table 3.7.1.1-2
Control Room Parameters	Refer to Table 3.7.1.1-2
Time to Switch from Control Room Normal Operating Mode to Emergency Mode (minutes)	10
Control Room ?/Q Values (sec/m^3)	
0-8 hours	1.1E-3
8-24 hours	5.0E-4
1-4 days	3.9E-4
4-30 days	3.0E-4

Table 3.7.1.5-1 Assumptions for Fuel Handling Accidents

Parameter	Value
Core Power (MWt)	3216.5
Total Number of Assemblies in Core	193
Highest Power Discharged Assembly	
Axial Peak to Average Ratio	1.7
Radial Peak to Average Ratio	1.7
Occurrence of Accident (hours after shutdown)	100
Damaged fuel rods	one assembly
Gap Fraction	
^{131}I	0.08
85 Kr	0.10
Noble Gasses and	0.05
Other Halogens	0.12
Alkali Metals	
Iodine Gap Inventory	
Organic(percent)	0.25
Elemental(percent)	99.75
Pool DF	
organic(percent)	1
Elemental(percent)	400
Purge Isolation Time (seconds)	NA
Adsorber Efficiency Filter System	NA
0-2 Hour Control Room ?/Q Value (sec/m^3) (Based upon plant vent release at 0 cfm)	6.44E-4
Offsite ?/Q Values	Refer to Table 3.7.1.1-2
Breathing Rates	Refer to Table 3.7.1.1-2

Table 3.7.1.5-1 Assumptions for Fuel Handling Accidents (cont.)

Control Room Parameters	Refer to Table 3.7.1.1-2
Time to Switch from Control Room Normal Operating Mode to Emergency Mode	No credit taken for emergency mode of operation.
Iodine Form Following release to atmosphere	97 percent Elemental 3 percent Organic

Table 3.7.1.6-1 Assumptions for Rod Ejection Accident

Parameter	Value
Core Thermal Power (MWt)	3216.5
Fuel Defects	
Clad Failure (percent)	10
Fuel Melting (percent)	0.25
Primary to Secondary Leak Rate (gpm/STEAM GENERATOR)	0.3
Per cent of Fuel which melts and releases activity to reactor coolant	
Alkali Metals &Noble Gases (percent)	100
Iodides (percent)	50
Per cent of Fuel which melts and releases activity to containment	
Noble Gases (percent)	100
Iodides (percent)	50
Iodine Partition Factor in the SGs before and after the accident	0.01
Containment Volume (ft^3)	2.61E6
Containment Leak Rate (percent/day)	
t = 0-1 day	0.10
t > 1 day	0.05
Iodine Form in Containment (fraction)	
Particulate	0.95
Organic	0.0015
Elemental	0.0485
Iodine Form (steam generator steaming path)	
Prior to release to atmosphere	100 percent Elemental
Following release to atmosphere	97 percent Elemental 3 percent Organic
Steam Dump from Relief Valves (g/min)	2.268E6

Table 3.7.1.6-1 Assumptions for Rod Ejection Accident (cont.)

Duration of Steam Dump from Relief Valves (sec)	4000
Primary Coolant Mass (g)	2.37E8
Secondary Coolant Mass (g)	1.275E8
Primary to Secondary Leak Rate (g/min)	4,550
Steaming Partition Factor	0.01
Control Room Operating Parameters	Refer to Table 3.7.1.1-2
Offsite ?/Q Values	Refer to Table 3.7.1.1-2
Breathing Rates	Refer to Table 3.7.1.1-2
Control Room Parameters	Refer to Table 3.7.1.1-2
Time to Switch from Control Room Normal Operating Mode to Emergency Mode (minutes)	3
Control Room ?/Q Values (Containment Pathway) (sec/m^3)	
0-8 hours	3.8E-4
8-24 hours	1.1E-4
1-4 days	8.3E-5
4-30 days	7.0E-5
Control Room ?/Q Values (Steaming Pathway) (sec/m^3)	
0-8 hours	1.1E-3
8-24 hours	5.0E-4
1-4 days	3.9E-4
4-30 days	3.0E-4
Gap Fraction	
All Isotopes except ^{85}Kr	0.10
^{85}Kr	0.30

Table 3.7.1.7-1 Assumptions for Small Break LOCA Analysis

Parameter	Value
Core Thermal Power (MWt)	3216.5
Activity Available for Release to the Containment	Refer to Table 3.7.1.1-1
Elemental Iodine Spray Removal Rate (1/hr)	NA
Fission Product Gap Fraction for Noble Gases, Iodine and Alkali Metals (percent)	3
Fraction of Fuel Rods Failing	1
Containment Sedimentation Removal Coefficient (1/hr)	0.1
Elemental Iodine Deposition Removal Coefficient (1/hr)	1.5
DF Limit for Elemental Iodine	200
DF Limit for Particulates	1000
Iodine Species (fraction)	
Elemental	0.485
Particulate	0.95
Organic	0.0015
Duration of Release (Days)	30
Containment Free Volume (ft^3)	2.61E6
Leakage Rate (percent/day)	
0-24 hours	0.10
> 24 hours	0.05
Control Room Free Volume (ft^3)	102,400
Filtered Emergency Intake Flow (cfm)	1800
Control Room Emergency Intake Filter System Efficiency (percent)	
Elemental and Organic	90
Particulate	99

Table 3.7.1.7-1 Assumptions for Small Break LOCA Analysis (cont.)

Parameter	Value
Control Room Unfiltered Air Infiltration Rate (cfm)	700
Time to Switch Control Room HVAC from Normal to Emergency Mode	Immediately
Control Room Occupancy Factors	
0-1 day	1.0
1-4 days	0.6
4-30 days	0.4
Atmospheric Dispersion Factors (sec/m^3)	
EAB	7.5E-4
LPZ	
0-8 hours	3.5E-4
8-24 hours	1.2E-4
1-4 days	4.2E-5
4-30 days	9.3E-6
Control Room (Containment Pathway)	
0-8 hours	3.8E-4
8-24 hours	1.1E-4
1-4 days	8.3E-5
4-30 days	7.0E-5
Control Room (Steam Generator Steaming Pathway	
0-2 hours	1.1E-3
Steam Generator Steaming Release Path	
Primary Coolant Mass (g)	2.37E8
Secondary Coolant Mass (g)	1.28E8
Primary to Secondary Leak Rate (g/min)	4,550
Steaming Rate from Secondary Side (g/min)	2.268E6
Steaming Partition Coefficient	0.01
Duration of Releases (seconds)	4,000

Table 3.7.1.7-1 Assumptions for Small Break LOCA Analysis (cont.)

Breathing Rates (m^3/sec)	
Offsite	
0-8 hours	3.47E-4
8-24 hours	1.75E-4
1-30 days	2.32E-4
Control Room	3.47E-4

Table 3.7.1.8-1 95 Percentile ?/Q Values from ARCON96 (sec/m^3)

Release Location	0 to 8 hours	8 to 24 hours	1 to 4 days	4 to 30 days
Unit 2 Containment Surface	3.83E-4	1.05E-4	8.31E-5	7.04E-5
Unit 2 Aux. Boiler Feed - Side	1.09E-3	4.99E-4	3.86E-4	2.99E-4
Unit 2 Aux. Boiler Feed - Stack	9.49E-4	4.17E-4	3.30E-4	2.54E-4
Unit 2 Vent - 0 cfm	6.44E-4	1.72E-4	1.37E-4	1.17E-4

Table 3.7.2-1 Radiological Consequences from Postulated Accidents (rem as TEDE)

Accident	EAB	LPZ	Control Room
1. Large Break LOCA			
ECCS Leakage	0	0.35	0.02
Containment Leakage	10	5.5	1.29
2. MSLB			
Pre-existing Spike	0.12	0.24	0.28
Accident Initiated Spike	0.088	0.81	1.11
3. SGTR			
Pre-existing Spike	3.53	1.68	2.12
Accident Initiated Spike	0.56	0.37	0.55
4. Locked Rotor			
Noble Gas only	0.12	0.14	0.017
Iodine & Particulates only	0.71	0.93	2.2
5. Fuel Handling Accident	2.2	1.0	1.2
6. Rod Ejection			
Containment (Gap)	0.56	1.69	0.66
Containment (Fuel Melt)	0.049	0.34	0.14
Primary to Secondary (Cs & I)	0.054	0.025	0.022
Primary to Secondary (Noble Gas)	0.48	0.22	0.23
7. Small Break LOCA			
Containment	3.0	4.37	1.34
Secondary Side (Cs & I)	0.30	0.14	0.14
Secondary Side (Noble Gas)	0.96	0.45	0.049

3.8 TS Bases

The licensee modified the TS Bases to reflect the changes. Since the Bases are not a part of the TS, the staff reviewed the Bases only to assure consistency with the proposed change. The staff is not approving the Bases change, but is including the revised Bases pages for completeness.

4.0 EVALUATION CONCLUSION

The proposed TS changes, as stated in Consolidated Edison's letter (Reference 2), would eliminate the TSs related to containment air filtration, reduce the time the reactor must be subcritical before beginning core alterations, relax containment integrity during fuel handling, and change the mode of post-accident control room ventilation system operation.

As previously discussed in this evaluation, the staff finds the proposed TS changes acceptable because:

The control room and offsite dose calculations meet the acceptance criteria.

The licensee adequately addressed the impact of increased radioactive and non-radioactive particulate loading on the containment fan cooler units.

Adequate defense in depth is maintained by natural sedimentation and the requirements for containment sprays, water level, and the natural decay of irradiated fuel.

Administrative controls over shutdown safety are in effect that ensure containment closure, should it be needed, and to control monitoring and filtration of any releases that might occur from a FHA.

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 (65 FR 3256). 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.

7.0 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.

8.0 REFERENCES

1. Letter from James S. Baumstark (Consolidated Edison Company) to NRC dated October 8, 1999, “NEI Pilot Program for Use of NUREG-1465", submitting the document “Radiological Consequences of Accidents for the Indian Point Nuclear Generating Station Unit No. 2 Using Source Term Methodology from NUREG-1465", prepared by Westinghouse Electric Company, LLC, dated October 8, 1999.

2. Letter from James S. Baumstark (Consolidated Edison Company) to NRC dated November 18, 1999, “Proposed Amendment Consisting of Changes to Technical Specifications for Containment Air Filtration, Control Room Air Filtration, and Refueling Conditions”.

3. Letter from James S. Baumstark (Consolidated Edison Company) to NRC dated February 14, 2000, “NEI Pilot Program for Use of NUREG-1465", submitting responses to RAIs and the document “Radiological Consequences of Accidents for the Indian Point Nuclear Generating Station Unit No. 2 Using Source Term Methodology from NUREG-1465; Supplemental Report Addressing Main Steam Line Break, Steam Generator Tube Rupture, and Revised Fuel Handling Accident Control Room Dose”, prepared by Westinghouse Electric Company, LLC, dated February 8, 2000.

4. Letter from James S. Baumstark (Consolidated Edison Company) to NRC dated March 21, 2000, “NEI Pilot Program for Use of NUREG-1465".

5. Letter from James S. Baumstark (Consolidated Edison Company) to NRC dated April 13, 2000, “NEI Pilot Program for Use of NUREG-1465", submitting responses to RAIs and the calculations PSAT 126.02CT.QA.04, “Calculation of IP2 ESF Component Leakage Iodine Release” and PSAT 126.02CT.QA.05, “Calculation of IP2 ESF Component Leakage Iodine Release with Boundary Layer Effect”, by Polestar Applied Technology.

6. Letter from James S. Baumstark (Consolidated Edison Company) to NRC dated May 11, 2000, “Post Accident ECCS Leakage”.

Principal Contributors: M. Snodderly, J. Segala, K. Parczewski, J. Hayes, L. Brown, D. Nguyen

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