June 20, 2000

MEMORANDUM TO: Marsha K. Gamberoni, Acting Chief
Project Directorate 1
Division of Licensing Project Management

FROM: Mark Reinhart, Chief ***/RA by Jay Lee Acting For/***
Licensing Section
Probabilistic Safety Assessment Branch

SUBJECT: ASSESSMENT OF AMENDMENT CONSISTING OF CHANGES TO TECHNICAL SPECIFICATIONS FOR CONTAINMENT AIR FILTRATION, CONTROL ROOM AIR FILTRATION AND REFUELING CONDITIONS AT INDIAN POINT UNIT 2 (TAC NO. MA6955)

We have performed an assessment of the potential radiological doses associated with the proposed technical specification amendment involving the containment air filtration, control room air filtration and refueling conditions at Indian Point Unit 2. In support of these technical specification and operational changes, the licensee presented revised dose assessments of postulated accidents. The assessments were structured such that they involved a complete implementation of the alternate source term (AST) for Indian Point Unit 2 in accordance with 10 CFR 50.67. Indian Point Unit 2 had volunteered to participate in the industry's pilot plant program for the implementation of the AST.

In addition to the re-calculation of postulated accident releases and associated doses, the licensee performed a re-assessment of the atmospheric dispersion parameters for the control room dose estimates associated with previously analyzed accidents. 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 locked rotor, rod ejection and fuel handling accidents, the licensee assumed values for gap activities which are inconsistent with the values which are being published in Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors. The gap values utilized for Indian Point Unit 2 were less for the rod ejection and locked rotor accidents than the values in Regulatory Guide 1.183.

CONTACTS: John J. Hayes, SPSB/DSSA
415-3167

Leta Brown, SPSB/DSSA
415-1232

The staff performed confirmatory calculations utilizing the gap factions expected to be published in Regulatory Guide 1.183. The staff's calculations confirmed that the plant could still meet the requirements of 10 CFR 50.67. Therefore, the licensee's proposed changes in fuel handling operation, containment filter technical specifications and control room design changes can be implemented. The staff concluded that the licensee's atmospheric dispersion assessment was acceptable for use in this dose assessment.

The reviewers for this effort were Leta Brown for atmospheric dispersion and Jack Hayes for the radiological dose assessment.

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Radiological Analysis Indian Point Unit 2 Pilot Plant Assessment

3.7.0 Radiological Analysis

The licensee performed calculations of the potential radiological doses associated with those aspects of the proposed technical specification amendment involving changes in containment air filtration, control room air filtration and refueling operations at Indian Point Unit 2. These results were submitted for staff review and approval. The licensee performed re-analyses of a select number of the Indian Point Unit 2 FSAR accidents and analyses of accidents not presently 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 which supported the changes to technical specifications and operations, the licensee also submitted assessments of the consequences of postulated accident which were independent of the proposed changes in technical specification and operations.

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

1. Large Break Loss of Coolant Accident (LOCA)
   - 25 rem TEDE
2. Main Steam Line Break
   - pre-existing spike case - 25 TEDE
   - accident-initiated spike case - 2.5 rem TEDE
3. Steam Generator Tube Rupture
   - pre-existing spike case - 25 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 which were unaffected by the change in technical specifications and operations but which were re-assessed included the locked rotor, steam generator tube rupture (SGTR) and the main steamline break (MSLB). Other accidents which were analyzed included the rod ejection and small break LOCA. 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.

For the locked rotor, rod ejection and fuel handling accidents, the licensee assumed values for gap activities which are inconsistent with the values which are being published in Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors. The gap values utilized for Indian Point Unit 2 were less for the rod ejection and locked rotor accidents than the values in Regulatory Guide 1.183. The staff cannot approve for Indian Point 2 a permanent change in its design basis which is inconsistent with the implementation policy for the AST.

The staff performed confirmatory calculations utilizing the gap factions expected to be published in Regulatory Guide 1.183. The staff's calculations confirmed that the plant could still meet the requirements of 10 CFR 50.67. Therefore, the licensee's proposed changes in fuel handling operation, containment filter technical specifications and control room design changes can be implemented, but the approval in the change in the design basis associated with the accident analyses can only be temporary. The licensee should be required to update their design basis for those accidents where the licensee's gap activity is less than that assumed in Regulatory Guide 1.183. Such an update should be submitted 6 months after the issuance of the guidance of Regulatory Guide 1.183.

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 Indian Point Unit 2 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. The control room automatically isolates normal ventilation on either a safety injection signal or a high radiation signal and initiates automatically operation of the control room emergency ventilation system. The time which 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 as noted above 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 consists of the radioactivity contained within reactor coolant. The duration of this release is assumed to be 25 seconds for a Westinghouse PWR such as Indian Point Unit 2. 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 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. The licensee originally postulated the pathways for releases in the event of a LOCA to be limited to containment leakage.

Initially, the licensee had excluded the release of activity associated with emergency core cooling system (ECCS) recirculation loop leakage which leaks into the primary auxiliary building beginning 24 hours following the accident. This leakage would be processed through the primary auxiliary building ventilation filtration system (PABVFS) prior to discharge to the environment. The PABVFS is a safety-related system. However, it is not treated as an engineered safety feature and is not included in the Indian Point Unit 2 Technical Specifications. Because of the licensee's treatment of this system and because it is not in technical specifications, in the performance of the typical LOCA analysis, it would be common to assume that a passive failure of a pump seal occurs at 24 hours following the LOCA and at that time a 50 gpm leak occurs for 30 minutes followed by ECCS leakage returning to the pre-passive failure value. However, inclusion of the passive failure results in an unacceptable control room operator dose.

Since the Indian Point Unit 2 PABVFS is safety grade with charcoal adsorbers and HEPA filters, the licensee proposed treatment of the PABVFS as an Engineered Safety Feature (ESF) based upon the incorporation of the following conditions into the Indian Point Unit 2 license:

a. Prior to restart of the plant from the next refueling outage [the refueling outage following the 2000 refueling outage], the Indian Point Unit 2 facility Technical Specifications shall be revised to include testing requirements for the PABVFS. Such testing will include laboratory testing of charcoal consistent with Generic Letter 99-02, in-place testing of the HEPA filters and charcoal adsorbers, differential pressure measurements relative to adjacent areas (Primary Auxiliary Building 1/8 inch negative to outside), and filter differential pressure measurements. The frequency of the tests shall be consistent with Regulatory Guide 1.52 Rev. 2 as modified by current Technical Specifications for 24 month fuel cycles. The laboratory testing of charcoal may be performed following 1440 hours of system operation rather than 720 hours.

b. During the period of time between now and the next refueling outage, the licensee will perform in-place filter testing in accordance with the requirements of Regulatory Guide 1.52 Rev. 2 Sections C.5.a., C.5.c., and C.5.d. except that the frequency of the testing shall be consistent with current Technical Specifications for 24 month cycles. Laboratory testing will meet the requirements of Generic Letter 99-02. The frequency of testing of charcoal will be after 1440 hours of system operation rather than 720 hours. This system will require 50 percent removal credit therefore, the charcoal will have a tested acceptance value of at least 70 percent. If any of the above testing does not meet the appropriate acceptance criteria, the condition shall be corrected within 30 days.

The staff reviewed this license condition and found it acceptable. Since the system is actually safety-related, it is only missing the technical specifications. Usually, the staff would have required that the licensee submit such technical specifications prior to approval of this amendment but the staff is currently re-assessing whether the passive failure needs to be considered in LOCA analyses. It is anticipated that a decision concerning this assumption will be made shortly. If the decision is made to exclude the passive failure, then no technical specifications will be required since the staff and the licensee found the doses to be acceptable when the passive failure was excluded. If it is decided that the passive failure portion of the analysis should remain, the licensee will have an ample period in which to prepare proposed technical specifications for the PABVFS. In the meantime, the same tests are being performed as would be performed if the licensee had technical specifications.

Since the licensee committed to perform the same tests as required by technical specifications, the staff is confident that the PABVFS will perform as intended. Because the PABVFS operates continuously with flow through the charcoal adsorbers and HEPA filters, this necessitates frequent laboratory tests of the charcoal be performed. The licensee proposed testing after 1440 hours of operation, which is twice as long as is generally required by Regulatory Guide 1.52. The staff found such an interval acceptable because the time frame associated with Regulatory Guide 1.52 is focusing on systems which are typically in a standby mode. Since the PABVFS is operated continuously, a test performed every 2 months is performance based and provides a more realistic assessment of the charcoal at any one time. The licensee's proposed acceptance criteria for the laboratory test of the charcoal incorporated a safety factor of 1.67 rather than a safety factor of 2 as detailed in Generic Letter 99-02. The staff concluded that the safety factor was acceptable for this application because the licensee was going to be testing the charcoal every 1440 hours rather than possibly once per 24 months if this system were in standby.

The conditions of a. and b. above shall automatically expire upon adoption by the NRC staff of the position (such as through the issuance of a Regulatory Guide) that a passive failure need not be considered for those plants without ESF filtration systems processing ECCS leakage.

In order for the PABVFS to be affective, it is necessary that any release pass through the charcoal adsorber and the HEPA filters. If the primary auxiliary building is maintained negative with respect to all adjacent areas, such filtration and adsorption will occur. However, when the fuel handling building ventilation system is operating, it is actually more negative than the primary auxiliary building. This would result in flow from the PAB to the fuel handling building thereby bypassing the PABVFS charcoal and HEPA. Consequently, the licensee included a commitment to change the applicable sections of the emergency operating procedures to require the fuel storage building ventilation fan be shut down prior to initiating hot leg recirculation 24 hours post accident. The licensee committed to making these changes prior to returning to service from the current outage.

In the licensee's assessment of the consequences associated with ECCS leakage, the licensee provided proprietary and non-proprietary calculations to demonstrate the fraction of ECCS leakage which would become airborne. These calculations presumed that a passive failure occurred 24 hours following the LOCA. Since the staff accepted the PABVFS as safety-related, the staff did not assess the calculations as to their appropriateness for a passive failure, since such a case was irrelevant for Indian Point 2. Therefore, it should not be presumed that such methodology has been approved by the staff. However, the staff did assess the use of the constant enthalpy method and found that it was acceptable. The staff utilized the value calculated by the constant enthalpy method of 5.5 percent of the ECCS leakage becoming airborne.

With the testing of the PABVFS, the passive failure need not be considered and the licensee's control room operator doses were less than 5 rem TEDE.

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. 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, in 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 which 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 was 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 which 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 are 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 hour based upon gap activity of iodine being 12 percent of the total core activity. The staff considered the limitation of iodine spiking to 5 hours inappropriate and should be changed in future analyses. The spike duration should be longer than that. 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.

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 Analysis

As noted previously, the licensee submitted re-analyses of postulated accidents which 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.

The licensee evaluated the consequences of a postulated SGTR accident. For the SGTR, primary to secondary leakage was assumed to be occurring at the technical specification rate of 0.3 gpm/steam generator from each of the four steam generators. In addition, primary to secondary leakage was occurring through the ruptured tube into the faulted 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 activities were 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 which results in a release of iodine from the fuel gap to the reactor coolant at a rate which 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 did 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 hours after the tube ruptures and that the spike lasted occurred for 7.5 hours.

Table 3.7.1.3-1 presents the assumptions utilized by the staff in their assessment of a Indian Point Unit 2 SGTR. The potential dose consequences of a SGTR accident at Indian Point Unit 2 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 Indian Point Unit 2 did not assess 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. 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 the 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 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 technical specifications 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 which 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 that after the release to the environment, 97 percent of the iodine was considered elemental and the remainder was organic. This was consistent with the model in DG-1081. Activity was released to the environment as a result of the leakage of primary coolant to the secondary side at the technical specification 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 were 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. 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 allowance of the containment personnel air locks and the equipment hatch 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 technical specification change, the licensee was proposing 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 Regulatory Guide (DG)-1081. Due to technical specification requirements, the licensee assumed that there was 23 feet of water over the damaged assembly and that this depth of water provided a 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 hi-rad 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 the 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 Regulatory Guide 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. However, the licensee must assure that when the personnel air locks and the equipment hatch are open, releases via these pathways are monitored consistent with GDC 64.

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. As a result of the accident, fuel clad damage and a small amount of fuel melt occurred. Due to the pressure differential between the primary and secondary side, primary coolant was discharged to the

secondary side. A portion of this radioactivity was 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 is also discharge to the containment via the spill from the opening in the reactor vessel head. A portion of this radioactivity is released to the environment via contaminant leakage.

The licensee determined that in the event of a rod ejection, less than 10 percent of the fuel rods undergo DNB. However, their analysis assumed that 10 percent of the fuel rods in the core 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 was released. The licensee's analysis assumed that a pre-existing spike existed in reactor coolant 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 technical specifications which limit secondary coolant to 0.15 µCi/g of dose equivalent ^{131}I. The licensee's future evaluations of this accident should be based upon the technical specification 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 that after the release to the environment, 97 percent of the iodine was considered elemental and the remainder was organic. This was consistent with the model in DG-1081. Activity was released to the environment as a result of the leakage of primary coolant to the secondary side at the technical specification 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 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 which 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). In their assessment, they assumed that a break occurred which resulted in substantial fuel damage in the reactor core but that the damage was insufficient to result in a containment pressure which would activate containment sprays. The licensee's assessment assumed that the entire core's gap activity was 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 by 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 is 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 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 assured 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 which 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 Indian Point Unit 2 Updated Final Safety Analysis Report. They were not reviewed by the staff as a part of this amendment.

The licensee used three 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 Regulatory Guide 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 three year period exceeded 99 percent and, therefore, exceeded the minimum 90 percent recovery rate guideline set forth in Regulatory Guide 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 surface, 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. The values utilized by the staff are listed in Table 3.7.1.8-1. The staff finds the X/Q values acceptable for use in this dose assessment.

3.7.2 Conclusions

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 technical specification amendment involving changes in containment air filtration, control room air filtration and refueling operations at Indian Point Unit 2. The staff concluded that the licensee’s atmospheric dispersion assessment was acceptable.

The licensee’s proposed changes in fuel handling operation, containment filter technical specifications and control room design changes can be implemented with the incorporation of the licensee condition associated with the PABVFS. The approval in the change in the design basis associated with the accident analyses can only be temporary. The licensee will be required to update their design basis for those accidents where the licensee’s gap activity is less than that assumed in Regulatory Guide 1.183. Such an update should be submitted 6 months after the issuance of the guidance of Regulatory Guide 1.183.

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
PAB Ventilation Filter System Efficiencies	
All forms of Iodine and Particulates (percent)	50

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
Passive Component Failure Leak Rate (gpm) for 30 minutes @24 hours post-LOCA	NA
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
Atmospheric Dispersion Factors (sec/m^3)	
EAB	7.5E-4

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

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)	
^{131}I	46.5
^{132}I	15.9
^{133}I	36.1
^{134}I	9.46
^{135}I	36.1

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 χ/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 χ/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 χ/Q Value (sec/m^3) (Based upon plant vent release at 0 cfm)	6.44E-4
Offsite χ/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 χ/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 χ/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 χ/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 χ/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	0.17	0.0088
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