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February 14, 2000

Re: Indian Point Unit No. 2
Docket No. 50-247

Document Control Desk
US Nuclear Regulatory Commission
Mail Station P1-137
Washington, DC 20555-0001

Subject: NEI Pilot Program for Use of NUREG-1465

As part of the NEI Pilot Program for the use of NUREG-1465, Consolidated Edison Company of New York, Inc., the owner and operator of Indian Point Unit No. 2, hereby submits the following document for the Nuclear Regulatory Commission review:

Responses to Requests for Additional Information
Received from the Regulatory Staff

This document also includes analyses utilizing the Alternate Source Term for postulated Steam Line Break outside containment, Steam Generator Tube Rupture, and Revised Fuel Handling Accident Control Room Dose. Although analyses of neither the postulated Steam Line Break nor Steam Generator Tube Rupture events would be required with our planned application of the Alternate Source Term, these analyses are being docketed at this time for completeness of our full implementation submittal. The Revised Fuel Handling Accident Control Room Dose analysis takes no credit for the Control Room HVAC filters, thus demonstrating that the Control Room filters need not be available during refueling. This significantly facilitates the modification schedule for the upcoming refueling outage. In addition, we have reviewed the proposed modifications to the Central Control Room HVAC System. As a result, we believe that significant simplification of the design can be achieved, and we are therefore revising our response to the appropriate RAI response submitted on August 27, 1999. Please note that the revised design remains consistent with the assumptions contained in the analyses submitted on October 8, 1999 and with this letter, and with the license amendment request submitted on November 18, 1999.

Should you or your staff have any questions regarding this submittal, please contact Mr. John McCann, Manager, Nuclear Safety and Licensing.

Very truly yours,

A001

Enclosure

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Responses to Requests for Additional Information Received from the Regulatory Staff

February 14, 2000

Question:

Please, provide justification for using the same decontamination factor for elemental iodine in the large break LOCA when sprays are operating and in the small break LOCA when the sole elemental iodine removal mechanism consists of its deposition on the containment walls. In the case of iodine removal by sprays there is a direct transfer of iodine to the sump and therefore decontamination factor is determined by relative volumes of the sump water and the net free volume of the containment. In the case of elemental iodine removal by deposition without assistance of the sprays two significant differences occur: (a) deposited iodine does not go directly into the sump water, (b) there is less sump water because spray solution is not included. These differences may have significant impact of the value of the corresponding decontamination factor. Section 6.5.2 of the SRP is quite clear about iodine removal by deposition by including it in the SRP section dealing with elemental iodine removal during spraying of fresh solution. By doing so it implies that elemental iodine removal by deposition is primarily a complementary mechanism to the elemental iodine removal by sprays. Therefore its use as a sole removal mechanism without considering simultaneous operation of sprays should reflect on the corresponding decontamination factor.

Response:

The SRP 6.5.2 model for deposition removal of elemental iodine is taken from Section 5.1.2 of NUREG/CR-0009, "Technological Bases for Models of Spray Washout of Airborne Contaminants in Containment Vessels," October 1978, by A. K. Postma, R. R. Sherry, and P. S. Tam. It is clear from this document that the deposition process is not dependent on operation of containment sprays. The derivation of the elemental iodine deposition removal model is based on tests in which there were no sprays operating. As indicated in NUREG/CR-0009, the value of sprays in the deposition process is that the sprays increase turbulence and, as a result, increase the mass transfer coefficient. However, the deposition removal does not depend on spray-induced turbulence. The mixing of the containment atmosphere is adequately provided for by the fan cooler units. (Even without the fan cooler units in operation there would be substantial mixing from natural circulation.)

Although the discussion of deposition removal in Section 6.5.2 of the SRP is located in a section titled "Elemental iodine removal during spraying of fresh solution," any implication that deposition is dependent on spray operation is mistaken. The SRP does refer to the area in the containment available for deposition as being the wetted area. This misconception about the need for wetted surfaces probably derives from the idea that the iodine is absorbed by the water on the surfaces, viewing the deposition merely as another mechanism by which the iodine is transported to the sump. Absorption of iodine into the water film does occur; however, even without a wetted surface, the iodine is absorbed into the surfaces and, per page 68 of NUREG/CR-0009, a typical plant has sufficient absorptive capacity to accommodate all of the elemental iodine released. Considering the absorption of iodine into the containment surfaces, the water film acts as an additional barrier, although not a substantial one (NUREG/CR-5009, page 65).

Relative to the questions raised regarding the use of a DF of 200 for the small break LOCA, the point is made that, without spray operation, there will be a smaller sump water volume to support the DF. The water volume will be the same as for the large break LOCA since, although there is no containment spray assumed, it is expected that all of the RWST water would be injected. It will take a longer time to inject the water than for the large break LOCA but the iodine removal process will also take longer than for the large break LOCA. The fact that there will be substantial absorption of elemental iodine into the containment surfaces and that a significant portion of this iodine is irreversibly retained (NUREG/CR-5009, page 72) means that there is a reduced inventory of elemental iodine remaining to be distributed between the containment atmosphere and the sump solution. The use of the DF of 200 does not take this into account and this constitutes additional conservatism. The iodine retained in surfaces is not available to participate in the partitioning between the sump and the containment atmosphere.

While it is true that for the large break LOCA in which the containment sprays are assumed to operate, removal of elemental iodine by deposition is primarily a complementary mechanism to the removal of elemental iodine by sprays, the enhancement applies only to the iodine removal rate. The use of deposition as a sole removal mechanism without considering simultaneous operation of sprays does not impact the decontamination factor achieved.

Question 1:

Provide an electronic copy of the meteorological data used to calculate the X/Q values. If the data are compressed, a mechanism to re-inflate the data should be provided. Data should be provided either in the format specified in Appendix A to Section 2.7, "Meteorology and Air Quality," of draft NUREG-1555, "Environmental Standard Review Plan," or in the ARCON96 format described in NUREG/CR-6331, "Atmospheric Relative Concentrations in Building Wakes." If the ARCON96 format is selected, the atmospheric stability categorization should be based on the delta-T methodology.

Response:

An electronic copy of the requested meteorology data was previously sent to the NRC on February 9, 2000.

Question 2:

Were all meteorological data used in the analysis collected under Regulatory Guide 1.23, "Onsite Measurement Programs," guidelines? If not, how were the data collected that did not meet the recommendations of Regulatory Guide 1.23 and why are the collection methodologies/conditions acceptable?

Response:

The data provided in response to question 1 that was used for the ARCON-96 calculations was collected under Regulatory Guide 1.23 guidelines.

Question 3:

During the period of data collection, was the tower area free from obstructions, (e.g., structures, trees) and micro-scale influences to ensure that the data were representative of the overall site area?

Response:

The primary meteorological monitoring tower has been maintained free of obstructions, throughout its operation. The New York Power Authority (NYPA) maintains the Indian Point site tower. NYPA is procedurally bound to prevent any obstructions near the tower, which would cause micro-scale influences. Special permission must be obtained (after evaluation) to allow any structure within 10 feet of the tower for every foot in height of the structure. Further, there are no trees or equivalent vegetation located in the general area of the tower.

Question 4:

What quality assurance checks were performed on the meteorological measurement systems prior to and during the period of collection to assure that the data are of high quality? What additional checks were performed in the data following collection and prior to input into the atmospheric dispersion calculations?

Response:

Met instrumentation channels are calibrated semi-annually to meet the accuracy criteria as set forth in RG 1.23. This includes calibration of the sensors to NIST traceable standards, as well as calibration checks of the wiring, data collection devices, etc. Further, data checks and weekly operational checks are employed to ensure the continued operation of the system. Malfunctioning equipment and unreliable data are replaced upon discovery. In the case of bad data, data from our backup tower or alternate data collection device is used to replace any data from the primary tower which is questionable. In addition to performing the semi-annual calibrations under the NYPA surveillance program, calibrations and O&M are detailed in NYPA procedures.

As part of the weekly operational system checks, the met database is reviewed for trends, problems and inconsistencies. Questionable data is not then included in the database. In preparing the hourly met data files and performing the joint frequency distributions, further data evaluation is performed by the computer code, which eliminates any data outside pre-established acceptance parameters.

Question 5:

Describe the methodology used to calculate the X/Q values. If a computer code was used in the analysis, provide a reference citation. If the methodology is a plant-specific application of a commonly used methodology, discuss the differences between the plant-specific application and the commonly used methodology.

Response:

The X/Q values used for the site boundary and the low population zone are the NRC previously approved values, and they are listed in UFSAR Table 14.3-46 (Sheet 1 of 2).

The X/Q values used for the control room are new, and were calculated using ARCON-96. The source length and width were divided by 6 as suggested by NRC.

Question 6:

Provide a list of each of the inputs to the analysis. If a computer code was used in making calculations and the output is small in size, a copy of the printouts showing input values may be acceptable. Describe the assumptions and bases for selection of the input values so as to result in the limiting dose for each accident scenario.

Response:

In the transmittal from Con Edison Dated October 8, 1999, there is a listing of each analysis input assumptions along with a discussion of the assumptions.

Question 7:

For the fuel handling accident calculations, are the plant vent fans providing 20,000 cfs of flow safety related?

Response:

Yes.

Revised Response to Question 4 of August 27, 1999

Should the Technical Specification amendment be approved, the Central Control Room (CCR) HVAC system will be modified to perform in the following modes of operation:

Mode 1 (Normal) will remain the normal mode of operation mixing approximately 920 cfm of unfiltered outside air with approximately 8280 cfm of return air to the CCR. In this mode, approximately 920 cfm of air is exhausted to the outside atmosphere via the toilet exhaust fan.

Mode 2 (Pressurized) will be the new incident mode of operation. Mode 2 will be automatically initiated by a Safety Injection signal or a high radiation signal. In this mode, 2000 cfm of outside air will be drawn through HEPA and carbon filters via booster fans and discharge into to the CCR system . The outside air serves to pressurize the CCR to a pressure positive to adjacent areas.

Mode 3 (Recirculation) will remain the incident mode of operation during a toxic gas or smoke event. Mode 3 will be automatically initiated by a toxic gas signal or a signal from the smoke detector. The only difference from the current mode 3 is that the booster fans will not start or run in the recirculation mode.

Radiological Consequences of Accidents for the Indian Point Nuclear Generating Station Unit No. 2 Using Source Term Methodology from NUREG-1465

Supplemental Report Addressing:

1. Main Steam Line Break
2. Steam Generator Tube Rupture
3. Revised Fuel Handling Accident Control Room Dose

Prepared for Consolidated Edison Company of New York, Inc.
by Westinghouse Electric Company LLC
February 8, 2000

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1.0 RADIOLOGICAL CONSEQUENCES DUE TO NUREG-1465 SOURCE TERMS

1.1 Introduction

The Indian Point Unit 2 licensing basis for the radiological consequences analyses is currently based on methodologies and assumptions that are derived from TID-14844 (Reference 1) and other early guidance.

In 1995, the NRC issued NUREG-1465, "Accident Source Terms Light-Water Nuclear Power Plants" (Reference 2) which provides an alternate source term model for the large break LOCA with core melt that is based on current understanding of light-water reactor (LWR) accidents and fission product behavior. NUREG-1465 is applicable to LWR designs and is intended to form the basis for the development of regulatory guidance for the analysis of radiological consequences for the large break LOCA and other design basis accidents.

The radiological consequences of a number of design basis accidents had been analyzed for Indian Point Unit 2 using the NUREG-1465 source term model and these analyses were reported in Reference 3. These analyses include the large break Loss-of-Coolant Accident, small break Loss-of-Coolant Accident, Fuel Handling Accident, Locked Rotor Accident, and Rod Ejection Accident.

This supplement to that report reports the results of analyses performed for the Main Steam Line Break and the Steam Generator Tube Rupture accidents. Additionally, the control room dose for the Fuel Handling Accident is reanalyzed with no credit taken for actuation of the control room HVAC emergency mode.

1.2 Common Analysis Inputs and Assumptions

The assumptions and inputs described below are common to more than one of the accident analyses addressed in this report. Inputs and assumptions specific to the individual accidents are discussed in the sections addressing the accident.

1.2.1 Dose Models and Timing

The doses are determined consistent with the direction provided in 10 CFR 50.67. The radiological consequences are reported as total effective dose equivalent (TEDE) dose which is the sum of the whole-body dose and the committed effective dose equivalent (CEDE) dose. Doses are determined at the site boundary (SB) for the worst two-hour interval, at the low population zone boundary (LPZ) for the duration of the accident, and in the control room for the duration of the accident.

1.2.2 Coolant Activity Levels and Iodine Spiking Models

The noble gas activity concentration in the primary coolant at the time the accident occurs is based on a fuel defect level of 1.0% with no credit for primary coolant cleanup.

For modeling of primary coolant iodine activity, the steam line break and the steam generator tube rupture accidents are analyzed considering both the situation of a pre-existing iodine spike and the situation in which the event initiates an iodine spike (accident-initiated spike). Indian Point Unit 2 does not currently have Technical Specification (Tech Spec) limits for the primary coolant iodine activity. For these accident analyses the Standard Tech Spec value of 60 μCi/g Dose Equivalent I-131 is used for the pre-existing spike. For the accident-initiated spike it is assumed that the initial iodine concentration in the primary coolant is at the Standard Tech Spec value of 1.0 μCi/g Dose Equivalent I-131 and that the equilibrium iodine appearance rates are those which support this coolant concentration.

The iodine activity concentration of the secondary coolant at the time an accident occurs is assumed to be 0.15 μCi/gm Dose Equivalent I-131.

The coolant source term data are provided in Table 1-1.

1.2.3 Miscellaneous Inputs

The nuclide data used in the analyses (decay constants, CEDE dose conversion factors, and average gamma disintegration energies) are listed in Table 1-2. Table 1-3 lists control room parameters, offsite atmospheric dispersion factors, and breathing rates that are applicable to all analyses. The atmospheric dispersion factors at the control room air intake are dependent on the activity release point(s) for the individual events.

Table 1-1: Coolant Source Terms

Primary coolant activity

Pre-existing iodine spike of 60 µCi/g D. E. I-131	
I-131	46.9 µCi/g
I-132	16.0 µCi/g
I-133	70.4 µCi/g
I-134	9.7 µCi/g
I-135	37.4 µCi/g
Equilibrium iodine (no spike) at 1.0 µCi/g D. E. I-131	
I-131	0.781 µCi/g
I-132	0.267 µCi/g
I-133	1.173 µCi/g
I-134	0.161 µCi/g
I-135	0.623 µCi/g
Noble gas (1.0 % fuel defects)	
Kr-85m	2.358 µCi/g
Kr-85	6.104 µCi/g
Kr-87	1.151 µCi/g
Kr-88	3.607 µCi/g
Xe-133m	3.052 µCi/g
Xe-133	291.3 µCi/g
Xe-135m	0.153 µCi/g
Xe-135	12.21 µCi/g
Xe-138	0.458 µCi/g

Secondary coolant activity

Iodine (0.15 µCi/g D. E. I-131)	
I-131	0.117 µCi/g
I-132	0.040 µCi/g
I-133	0.176 µCi/g
I-134	0.024 µCi/g
I-135	0.093 µCi/g
Noble gas	none

Table 1-2: Nuclide Data

Iodines	Decay Constant (hr^{-1})	Gamma Energy (Mev/dis)	CEDE Dose Conversion Factor (rem/Ci)
I-131	3.59E-3	3.8E-1	3.29E4
I-132	3.03E-1	2.2E+0	3.81E2
I-133	3.33E-2	6.0E-1	5.85E3
I-134	7.91E-1	2.6E+0	1.31E2
I-135	1.05E-1	1.4E+0	1.23E3

Noble Gases	Decay Constant (hr^{-1})	Whole Body Dose Conversion Factor ($rem \cdot m^3/Ci \cdot sec$)
Kr-85m	1.55E-1	3.707E-2
Kr-85	7.37E-6	4.84E-4
Kr-87	5.47E-1	1.46E-1
Kr-88	2.48E-1	3.70E-1
Xe-131m	2.41E-3	1.52E-3
Xe-133m	1.30E-2	5.53E-3
Xe-133	5.46E-3	6.24E-3
Xe-135m	2.72E+0	7.75E-2
Xe-135	7.56E-2	4.82E-2
Xe-138	2.93E+0	1.98E-1

Table 1-3: Misellaneous Inputs and Assumptions

Coolant Mass	
Primary coolant	2.37E8 g
Secondary coolant mass (per steam generator)	
Minimum	3.188E7 g
Maximum	5.83E7 g
Control Room Parameters	
Volume	102,400 ft^3
Normal ventilation mode	
Filtered intake flow	0.0 cfm
Filtered recirculation flow	0.0 cfm
Unfiltered intake flow	920 cfm
Emergency ventilation mode	
Filtered intake flow	1800 cfm
Filtered recirculation flow	0.0 cfm
Unfiltered intake flow	0.0 cfm
Filter efficiencies (for filtered intake and recirculation)	
Elemental iodine	95%
Organic iodine	90%
Particulates	99%
Unfiltered inleakage	1185 cfm
Operator occupancy factors	
0-24 hours	1.0
24 hours – 4 days	0.6
>4 days	0.4
Offsite Atmospheric Dispersion Factors	
Site Boundary (0 - 2 hours)	7.5E-4 sec/m^3
Low Population Zone Outer Boundary	
0 - 8 hours	3.5E-4 sec/m^3
8 - 24 hours	1.2E-4 sec/m^3
24 - 96 hours	4.2E-5 sec/m^3
96 - 720 hours	9.3E-6 sec/m^3
Breathing Rates	
Offsite	
0 - 8 hr	3.47E-4 m^3/sec
8 - 24 hr	1.75E-4 m^3/sec
>24 hr	2.32E-4 m^3/sec
Control room	3.47E-4 m^3/sec

2.0 STEAM LINE BREAK RADIOLOGICAL CONSEQUENCES

The complete severance of a main steam line outside containment is assumed to occur. The affected steam generator will rapidly depressurize and release radioiodines initially contained in the secondary coolant and the primary coolant activity, transferred via steam generator tube leaks, directly to the outside atmosphere. A portion of the iodine activity initially contained in the intact steam generators and noble gas activity due to tube leakage is released to atmosphere through either the atmospheric relief valves or the main steam safety valves. This section describes the assumptions and analysis performed to determine the offsite and control room doses resulting from the release of activity

2.1 <u>Input Parameters and Assumptions</u>

The analysis of the steam line break radiological consequences uses the analytical methods and assumptions outlined in the Standard Review Plan (Reference 4) as adjusted and modified by draft regulatory guide DG-1081 (Reference 5).

The amount of primary to secondary steam generator tube leakage in each of the steam generators is assumed to be equal to the Technical Specification limit for a single steam generator of 0.3 gpm.

No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of offsite power.

The steam generator connected to the broken steam line is assumed to boil dry within the initial ten minutes following the steam line break. The entire liquid inventory of this steam generator is assumed to be steamed off and all of the iodine initially in the steam generator is assumed to be released to the environment. Also, the iodine carried over to the faulted steam generator by tube leaks is assumed to be released directly to the environment with no credit taken for iodine retention in the steam generator.

For the intact steam generators, an iodine partition coefficient of 0.01 (curies iodine /gm steam)/(curies iodine/gm water) is used (Reference 4). The concentration of iodine in the intact steam generators thus increases over the duration of the accident.

As discussed in Section 1.2.2, for modeling of primary coolant iodine activity the steam line break accidents is analyzed considering both the situation of a pre-existing iodine spike and the situation in which the event triggers an iodine spike (accident-initiated spike). The pre-existing iodine spike is defined in Section 1.2.2 as 60 μCi/g Dose Equivalent I-131 for the pre-existing spike.

For the accident-initiated iodine spike case, it is assumed that the iodine release rate from the fuel to the RCS is increased to a value 500 times greater than the release rate corresponding to the assumed maximum equilibrium RCS concentration of 1.0 μCi/gm of Dose Equivalent I-131 (Reference 5). The equilibrium iodine appearance rates in the RCS are provided in Table 1-1. The duration of the accident-initiated iodine spike is limited by the amount of activity available in the fuel-cladding gap. Based on having 12 percent of the I-131 in the fuel-cladding gap (Reference 5), the gap inventory would be depleted within five hours. The analysis assumes that the spike is terminated at that time.

All noble gas activity carried over to the secondary side through steam generator tube leakage is assumed to be immediately released to the outside atmosphere.

At 42 hours after the accident, the Residual Heat Removal System is assumed to be capable of all decay heat removal and that there are thus no further steam releases to atmosphere from the secondary system. The activity releases due to leakage of primary coolant to the faulted steam generator are assumed to continue until the primary coolant temperature is reduced to less than 212°F at 70 hours.

The major assumptions and parameters used specifically in this analysis are itemized in Table 2-1.

It is assumed that the control room HVAC system begins is initially in the normal operating mode. The system is assumed to be in the emergency mode at one minute after event initiation, where it remains throughout the event.

2.2 Acceptance Criteria

The offsite dose limit for a steam line break with a pre-accident iodine spike is 25 rem TEDE which is the guideline value of 10 CFR 50.67. For the steam line break with an accident-initiated iodine spike, the dose acceptance criterion is a "small fraction of" the 10 CFR 50.67 guideline value, or 2.5 rem TEDE per draft regulatory guide DG-1081 (Reference 5). The criterion for the control room dose is 5 rem TEDE per 10 CFR 50.67.

2.3 Results and Conclusions

For the case with pre-accident iodine spike:

Site Boundary	0.2 rem TEDE
Low Population Zone	0.4 rem TEDE
Control Room	0.6 rem TEDE

For the case with accident-initiated iodine spike:

Site Boundary	0.6 rem TEDE
Low Population Zone	1.4 rem TEDE
Control Room	2.6 rem TEDE

The Site Boundary doses reported are for the worst two hour period. For the pre-existing iodine spike case the worst two-hour period is 0-2 hours but for the accident-initiated spike case the worst two-hour period is 6-8 hours. The low population zone dose is provided for the duration of the accident and the control room dose is calculated for the duration of accident releases plus the time required to reduce the control room activity to inconsequential levels.

All the doses are within the acceptance criteria.

Table 2-1: Steam Line Break Dose Analysis Assumptions

Reactor Coolant Mass	See Table 1-3
Secondary Coolant Mass	See Table 1-3
Reactor Coolant Noble Gas Activity Prior to accident	See Table 1-1
Reactor Coolant Iodine Activity (Pre-existing spike)	See Table 1-1
Reactor Coolant Iodine Activity (equilibrium condition)	See Table 1-1
Equilibrium Iodine Appearance Rates	See Table 1-1
Increase in Iodine Appearance Rate Increase Due to the Accident-Initiated Spike	500 times equilibrium rates
Duration of Accident-Initiated Spike	5 hours
Secondary Coolant Activity Prior to Accident	See Table 1-1
Iodine Chemical Form (After Release to Atmosphere)	
Elemental	97%
Organic	3%
Primary to Secondary Leak Rate	0.3 gpm per steam generator
Time to Reduce Primary Coolant to <212°F	70 hours
Steam Release from Intact Steam Generators	
0-2 hours	6.0E5 lb_m
2-8 hours	1.1E6 lb_m
8-24 hours	1.5E6 lb_m
24-40 hours	1.3E6 lb_m
40-42 hours	1.6E5 lb_m
>42 hours	none
Pre-Trip Condenser Iodine Removal Factor	N/A
Intact Steam Generator Iodine Partition Factor	0.01
Faulted Steam Generator Iodine Partition Factor	1.0
Offsite Power	Lost
Initiation of the Control Room Emergency HVAC Operation	60 seconds
<u>Control Room Parameters</u>	See Table 1-3
<u>Atmospheric Dispersion Factors</u>	
Offsite	See Table 1-3
Control Room	
0 – 2 hours	1.29E-3 sec/m^3
2 – 8 hours	1.01E-3 sec/m^3
8 – 24 hours	3.22E-4 sec/m^3
> 24 hours	3.27E-4 sec/m^3
<u>Breathing Rates</u>	See Table 1-3

3.0 STEAM GENERATOR TUBE RUPTURE

The steam generator tube rupture (SGTR) event is separated into two analyses, a thermal and hydraulic analysis and a radiological consequences analysis. Both are discussed in this section.

3.1 Steam Generator Tube Rupture Thermal and Hydraulic Analysis

3.1.1 Introduction

The major hazard associated with an SGTR event is the radiological consequences resulting from the transfer of radioactive reactor coolant to the secondary side of the ruptured steam generator and subsequent release of radioactivity to the atmosphere. The primary thermal-hydraulic parameters which affect the calculation of offsite doses for an SGTR include the amount of reactor coolant transferred to the secondary side of the ruptured steam generator, the amount of primary to secondary break flow that flashes to steam and the amount of steam released from the ruptured steam generator to the atmosphere.

The SGTR thermal-hydraulic analysis supports a T_{AVG} window range of 549.0°F to 579.7°F. Plant secondary side conditions (e.g., steam pressure, flow, temperature) are based on 25% tube plugging. Two cases have been analyzed. The first is with low T_{AVG} and 25% steam generator tube plugging and the second with high T_{AVG} and 25% tube plugging.

The accident analyzed is the double-ended rupture of a single steam generator tube. It is assumed that the primary-to-secondary break flow following an SGTR results in depressurization of the reactor coolant system (RCS), and that reactor trip and safety injection are automatically initiated on low pressurizer pressure. Loss of offsite power is assumed to occur at reactor trip resulting in the release of steam to the atmosphere via the atmospheric relief valves and/or the main steam safety valves. Following actuation of safety injection, it is assumed that the RCS pressure stabilizes at the value where the injection flow and the break flow rates are equal. The equilibrium primary-to-secondary break flow is assumed to persist until the operators have completed the actions necessary to terminate the break flow and the steam releases from the ruptured steam generator.

After termination of releases from the ruptured steam generator, it is assumed that steam is released only from the intact steam generators in order to dissipate the core decay heat and to cool the plant down to the residual heat removal (RHR) system operating conditions. It is assumed that the RHR system is not able to accommodate the total decay heat load until 42 hours after initiation of the SGTR and that steam releases are terminated at that time. The analysis conservatively does not credit operation of the RHR system until 42 hours. A primary and secondary side mass and energy balance is used to calculate the steam release and feedwater flow for the intact steam generators.

3.1.2 Input Parameters and Assumptions

The SGTR thermal-hydraulic analysis assumptions are listed in Table 3-1. A summary of key input assumptions follows.

Safety Injection Flowrates

A larger safety injection flowrate results in a greater RCS equilibrium pressure and, consequently, higher break flow. Maximum flowrates were therefore assumed for this analysis.

Termination of Releases from the Ruptured Steam Generator

Operator action time required to terminate break flow is 45 minutes or less, as demonstrated by simulator testing. For the purposes of this calculation, a conservative hand calculational approach was used to determine break flow, flashing fraction, and steam release for a 30 minute period. Using this analytical approach, the results obtained using the 30 minute duration bound the results that would be achieved using the demonstrated 45 minutes together with detailed computer modeling.

Termination of Steaming to Remove Decay Heat

Because of the potential for high service water temperature, the Residual Heat Removal System may not be capable of removal of all decay heat until 42 hours after the initiating event. While the system would be removing some of the decay prior to this time, thus reducing the amount of steam releases required for heat removal, no credit is taken for the system's operation until 42 hours. The effect of this 42 hour delay to terminate steaming is not significant since the activity released from the intact steam generators is small relative to that released by the ruptured steam generator.

3.1.3 Description of Analyses Performed

Break Flow, Steam Releases and Feedwater Flows

Two cases were considered to bound the operating conditions for the RCS temperature range and to determine the limiting steam releases and break flow from the ruptured steam generator. A separate calculation was performed to determine long-term steam releases from the intact steam generators.

Break Flow Flashing Fraction

A portion of the break flow will flash directly to steam upon entering the secondary side of the ruptured steam generator. A conservative calculation of the flashing fraction was performed using the limiting conditions from the break flow calculation cases. Two time intervals were considered, pre- and post- reactor trip (safety injection initiation occurs concurrently with reactor trip). Since the RCS and steam generator conditions are different before and after the trip, different flashing fractions would be expected.

The flashing fraction is based on the difference between the primary side fluid enthalpy and the saturation enthalpy on the secondary side. Therefore, the highest flashing will be predicted for the case with the highest primary side temperatures. For the flashing fraction calculations it is conservatively assumed that all of the break flow is at the hot leg temperature (the break is assumed to be on the hot leg side of the steam generator). Similarly, a lower secondary side pressure maximizes the difference in the primary and secondary enthalpies resulting in more flashing.

For conservatism in the radiological calculations, reactor trip is assumed at time 0 seconds. Although the pre-trip flashing fraction would be higher due to the lower secondary pressure, the pre-trip flashed break flow passes through the condenser, which reduces the iodine concentration in the steam by a factor of 100. This benefit of the condenser more than offsets the penalty produced by the higher pre-trip flashing fraction. The limiting radiological consequences would therefore be the case with the highest post-trip flashed break flow. Both cases consider the same post-trip RCS pressure of 1500 psia and post-trip steam generator pressure of 885.4 psia. The highest post-trip flashing fraction, based on the range of operating temperatures covered by this analysis, is for a case with a hot leg temperatures of 611.7°F. It is conservatively assumed that the hot leg coolant temperature is not reduced over the duration of the break flow. However, with a hot leg temperature of 611.7°F, and the RCS pressure of 1500 psia, the RCS fluid would be superheated. It is reasonable to model the post-trip RCS fluid as saturated liquid at 1500 psia, with a corresponding hot leg temperature, since this is the minimum amount of cooling that could keep the RCS subcooled.

3.1.4 Results of Thermal and Hydraulic Analysis

For the SGTR, the amount of radioactivity released to the atmosphere is calculated from the activity released through the safety valves associated with the ruptured steam generator (from the flashed break flow and steam released). Therefore, the worst radiological consequences result from the SGTR case with the greatest amount of flashed break flow and steam released. Likewise, a greater break flow results in greater radiological contamination of the secondary side which in turn results in a greater amount of activity released along with the steam. The break flow and steam releases calculated therefore represent bounding values which are conservative for a radiological consequences calculation.

The limiting calculated values for the tube rupture break mass release to the secondary side, the flashing fraction for the break flow, the ruptured steam generator steam releases, and the long-term steam releases from the intact steam generators are provided below:

Mass of Primary Coolant Released to Secondary Side by Break	128,000 lb_m
Break Flow Flashing Fraction	0.13
Ruptured Steam Generator Steam Release	73,000 lb_m
Intact Steam Generator Steam Release	
0-2 hours	514,000 lb_m
2-8 hours	1,039,000 lb_m
8-42 hours	2,870,000 lb_m

Additionally, the analysis determined that safety injection would be actuated at 320 seconds into the event. This would also initate the transition from the normal mode to the emergency mode for the control room HVAC.

3.2 Steam Generator Tube Rupture Radiological Analysis

As discussed in Section 3.1, the complete severance of a single steam generator tube is assumed to occur. Due to the pressure differential between the primary and secondary systems, radioactive reactor coolant is discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through the main condenser, the atmospheric relief valves, or the main steam 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.

3.2.1 Input Parameters and Assumptions

The analysis of the SGTR radiological consequences uses the analytical methods and assumptions outlined in the Standard Review Plan Section 15.6.3 (Reference 6) and the draft regulatory guide DG-1081 (Reference 5).

In the pre-accident iodine spike case it is assumed that a reactor transient has occurred prior to the SGTR and has raised the RCS iodine concentration to 60 μCi/gm of Dose Equivalent (DE) I-131 (60 times the assumed maximum coolant equilibrium concentration limit of 1.0 μCi/gm of Dose Equivalent I-131).

For the accident-initiated iodine spike case, the reactor trip associated with the SGTR creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 335 times greater than the release rate corresponding to the assumed maximum equilibrium RCS concentration of 1.0 μCi/gm of Dose Equivalent I-131 (Reference 5). The duration of the accident-initiated iodine spike is limited by the amount of activity available in the fuel-cladding gap. Based on having 12 percent of the iodine in the fuel-cladding gap, the gap inventory would be depleted within 7.5 hours and the analysis assumed that the spike is terminated at that time.

The noble gas activity concentration in the RCS at the time the accident occurs is based on a one percent fuel defect level. The iodine activity concentration of the secondary coolant at the time the SGTR occurs is assumed to be equivalent to the Technical Specification limit of 0.15 μCi/gm of Dose Equivalent I-131.

The amount of primary to secondary steam generator tube leakage in the intact steam generators is assumed to be equal to the Technical Specification limit for the leakage of 0.3 gpm per steam generator.

An iodine partition factor in the steam generators of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) is used (Reference 5). Prior to reactor trip and concurrent loss of offsite power an iodine removal factor of 0.01 could be taken for steam released to the condenser, but conservatively, the pre-trip condenser iodine removal is ignored.

All noble gas activity carried over to the secondary side through steam generator tube leakage is assumed to be immediately released to the outside atmosphere.

The safety injection actuation setpoint will be reached at ~320 seconds from event (Section 3.1.4). This signal also causes the control room HVAC to switch from the normal operation mode to the accident mode of operation. It is conservatively assumed that the switchover of control room HVAC to the accident mode of operation is not completed until 400 seconds after the event initiation.

At 42 hours after the accident, the Residual Heat Removal System is assumed to be capable of all decay heat removal and that there are thus no further steam releases to atmosphere from the secondary system.

The major assumptions and parameters used in this analysis are itemized in Table 3-2.

3.2.2 Dose Acceptance Criteria

The offsite dose limits for a SGTR with a pre-accident iodine spike is the guideline value of 10 CFR 50.67. This guideline value is 25 rem TEDE. For a SGTR with an accident-initiated iodine spike, the acceptance criterion is a "small fraction of" the 10 CFR 50.67 guideline value, or 2.5 rem TEDE per draft regulatory guide DG-1081 (Reference 5). The criteria for the control room dose are 5 rem TEDE per 10 CFR 50.67.

3.2.3 Results and Conclusions

For the pre-accident iodine spike:

Site Boundary	4.4 rem TEDE
Low Population Zone	2.1 rem TEDE
Control Room	3.0 rem TEDE

For the accident-initiated iodine spike:

Site Boundary	1.3 rem TEDE
Low Population Zone	0.7 rem TEDE
Control Room	0.9 rem TEDE

The Site Boundary doses reported are for the worst two hour period which is the first two hours. The low population zone dose is provided for the duration of the accident and the control room dose is calculated for the duration of accident releases plus the time required to reduce the control room activity to inconsequential levels.

The doses are all within the acceptance criteria.

Table 3-1 - Assumptions Used for SGTR Thermal-Hydraulic Analysis

Core Power	3071.4 MWt
RCS T_{AVG}	549 - 579.7°F
Tube Plugging Level	25%
RCS Pressure Post-Accident	1500 psia
Maximum Hot Leg Temperature	611.7°F
Time of Loss of Condenser Operation	0 seconds
Time of Safety Injection Initiation	320 seconds
Offsite Power	Lost
Time to Terminate Break Flow	30 minutes[1]
Low Pressurizer Pressure Setpoint for Safety Injection Actuation	1847.7 psia
Lowest Steam Generator Safety Valve Reseat Pressure (Includes 18% Main Steam Safety Valve Blowdown Which Covers the -3% Setpoint Tolerance)	885.4 psia
Minimum Auxiliary Feedwater Flow (total)	380 gpm

[1] Operator action time required to terminate break flow is 45 minutes or less, as demonstrated by simulator testing. For the purposes of this calculation, a conservative hand calculational approach was used to determine break flow, flashing fraction, and steam release for a 30 minute period. Using this analytical approach, the results obtained using the 30 minute duration bound the results that would be achieved using the demonstrated 45 minutes together with detailed computer modeling.

Table 3-2 - Assumptions Used for SGTR Dose Analysis

Reactor Coolant Mass	See Table 1-3
Secondary Coolant Mass	See Table 1-3
Reactor Coolant Noble Gas Activity Prior to accident	See Table 1-1
Reactor Coolant Iodine Activity (Pre-existing spike)	See Table 1-1
Reactor Coolant Iodine Activity (equilibrium condition)	See Table 1-1
Equilibrium Iodine Appearance Rates	See Table 1-1
Increase in Iodine Appearance Rate Increase Due to the Accident-Initiated Spike	335 times equilibrium rate
Duration of Accident-Initiated Spike	7.5 hours
Secondary Coolant Activity Prior to Accident	See Table 1-1
Iodine Chemical Form (After Release to Atmosphere)	
Elemental	97%
Organic	3%
Primary to Secondary Leak Rate	0.3 gpm per steam generator
Tube Rupture Break Flow (0 - 0.5 Hours)	128,000 lb_m
Percentage of Break Flow Which Flashes	13.0%
Steam Release from Ruptured Steam Genrator (0 – 0.5 Hours)	73,000 lb_m
Steam Release from Intact Steam Generators	
0 - 2 Hours	514,000 lb_m
2 - 8 Hours	1,039,000 lb_m
8 - 42 Hours	2,870,000 lb_m
Pre-Trip Condenser Iodine Removal Factor	N/A
Steam Generator Iodine Partition Factor	0.01
Duration of Activity Release from Secondary System	42 hours
Offsite Power	Lost
Initiation of the Control Room Emergency HVAC Operation	400 seconds
<u>Control Room Parameters</u>	See Table 1-3
<u>Atmospheric Dispersion Factors</u>	
Offsite	See Table 1-3
Control Room (steam generator steaming pathway)	
0 – 2 hours	1.09E-3 sec/m^3
2 – 8 hours	8.14E-4 sec/m^3
8 – 24 hours	2.64E-4 sec/m^3
> 24 hours	2.67E-4 sec/m^3

4.0 FUEL HANDLING ACCIDENT – NO CREDIT FOR CR HVAC EMERGENCY MODE

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. Reference 3 reported the results of the fuel handling accident dose analysis of this event with the assumption that the control room HVAC operation is switched from normal operating mode to the emergency mode. This reanalysis is identical to that reported in Reference 3 except that it is assumed that the control room HVAC remains in the normal operating mode. The discussion provided in Reference 3 is repeated here with modifications appropriate to the change in assumptions.

4.1 Input Parameters and Assumptions

The major assumptions and parameters used in the analysis are itemized in Table 4-1. Analysis of the accident is performed with assumptions selected so that the results are bounding for the accident occurring either inside containment or in the fuel handling building. All of the activity released from the damaged fuel is assumed to be released within two hours. Since the assumptions and parameters for a FHA inside containment are identical to those for a FHA in the fuel-handling building, the offsite doses are the same regardless of the location of the accident.

4.1.1 Source Term

The current licensing basis analysis assumes that one fuel assembly is damaged releasing the gap inventory from all fuel rods. The gap inventory is assumed to be 10% of the iodines and noble gases in the rod. The assembly 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 assembly fission product inventory). A decay time of 174 hours is specified by the Technical Specifications.

The revised analysis of the radiological consequences following a fuel handling accident (FHA) uses the source terms outlined in NUREG-1465 (Reference 2). While NUREG-1465 specifies that the gap fraction is 3.0 percent for immediate release (with an additional 2.0% released if the fuel cooling is maintained), a gap fraction of 5.0% is conservatively assumed. The gap inventory includes the iodines, noble gases, and the alkali metals (cesium & rubidium).

As in the existing licensing basis, it is assumed that all of the fuel rods in the equivalent of one fuel assembly are damaged to the extent that all their gap activity is released. Also consistent with the existing licensing basis, the assembly inventory is based on the assumption that the subject fuel assembly has been operated at 1.7 times core average power.

The decay time used in the analysis is 100 hours. Thus, the analysis supports reduction of the Technical Specifications limit of 174 hours decay time prior to fuel movement to 100 hours.

4.1.2 Fission Product Form

While NUREG-1465 specifies that the iodine released from the fuel is in the form of 95% cesium iodide, 4.85% elemental, and 0.15% organic, the FHA reanalysis assumes that the iodine is 99.75% elemental and 0.25% organic. This is consistent with NRC guidance in draft DG-1081 (Reference 5). In actuality, the nonvolatile cesium iodide would all be retained in the water although gradual conversion of the cesium iodide to form elemental iodine would slowly increase the amount of iodine in the volatile form and which might be released to the environment.

This assumption of 99.75% elemental iodine and 0.25% organic iodine also is consistent with the existing licensing basis analysis and with Safety Guide 1.25 (Reference 7).

4.1.3 Pool Scrubbing Removal of Activity

Per the technical specifications, 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 (DF) of 500 specified by draft DG-1081 (Reference 5) for elemental iodine would apply. However, in recognition that fuel rod pressure might exceed 1200 psig (but would be less than 1500 psig), the DF is reduced to 400. The DF for organic iodine and noble gases is 1.0.

The cesium released from the damaged fuel rods is assumed to remain in a nonvolatile form and would not be released from the pool.

4.1.4 Isolation and Filtration of Release Paths

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 two hour period. Since no filtration or containment isolation is modeled, this analysis supports refueling operation with the equipment hatch or personnel air lock remaining open.

4.1.5 Control Room HVAC Operation with No Credit for Emergency Mode

In the originally reported analysis (Reference 3), credit was taken for the CR HVAC entering the emergency mode of operation at ten minutes after the accident occurs. In this revised analysis no credit is taken for operation of the CR HVAC in the emergency mode. Instead, it is assumed that the HVAC remains in the normal mode of operation. This addresses the possibility of fuel movement at a time when maintenance is being performed on the filters.

4.2 Acceptance Criteria

From draft DG-1081 (Reference 5), the offsite doses should be "well within" the guidelines from 10 CFR 50.67, or 6.25 rem TEDE. This applies for both the site boundary dose (worst two hour interval) and for the low population zone boundary dose (duration of accident). From 10 CFR 50.67, the dose limit for the operators in the control room is 5.0 rem TEDE.

4.3 Results

The fuel handling accident doses are:

Site Boundary	1.6 rem TEDE
Low Population Zone	0.75 rem TEDE
Control Room	1.0 rem TEDE

The acceptance criteria are met.

Table 4-1: Fuel Handling Accident Assumptions

Delay after shutdown before fuel movement	100 hours
Core activity at 100 hours after shutdown	
I-131	6.18E7
I-132	5.21E7
I-133	6.42E6
I-135	4.60E3
Kr-85	1.08E6
Xe-131m	9.30E5
Xe-133m	2.25E6
Xe-133	1.20E8
Xe-135m	7.37E2
Xe-135	2.23E5
Number of fuel assemblies in core	193
Radial peaking factor	1.7
Fuel rod gap fraction	0.05
Fuel damaged	one assembly
Iodine species	
Elemental	99.75%
Organic	0.25%
Water depth	23 feet
Pool scrubbing factor	
Elemental iodine	400
Organic iodine	1
Noble gases	1
Release path filter efficiency	N/A
Isolation of release paths	None
Control Room X/Q (0-2 hr) (based on plant vent release at 20,000 cfm)	5.53E-4 sec/m^3
Offsite X/Q	See Table 1-3
Breathing rates	See Table 1-3
Control Room parameters	See Table 1-3
Time to switch CR HVAC from normal to emergency mode	No credit taken for emergency mode operation

5.0 REFERENCES

1. TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," U.S. AEC, Division of Licensing and Regulation, J. J. DiNunno, et al, March 23, 1962
2. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995, U.S. Nuclear Regulatory Commission.
3. "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, October 8, 1999.
4. NUREG-0800, Standard Review Plan 15.1.5, Appendix A, "Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR," Revision 2, July 1981.
5. Draft Regulatory Guide DG-1081, "Alternative Radiological Source Terms for Evaluating Design Basis Accident at Nuclear Power Reactors," December 1999.
6. NUREG-0800, Standard Review Plan 15.6.3, "Radiological Consequences of a Steam Generator Tube Rupture (PWR)," Revision 2, July 1981.
7. U. S. AEC Safety Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," 3/23/72.