REVISED RADIOLOGICAL CONSEQUENCES OF ACCIDENTS FOR INDIAN POINT UNIT 2 TAKING INTO ACCOUNT SOURCE TERM METHODOLOGY FROM NUREG-1465

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1.0 RADIOLOGICAL CONSEQUENCES DUE TO NUREG-1465 SOURCE TERMS

1.1 Introduction

NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Reference 1) provides a postulated fission product source term that is based on current understanding of light-water reactors (LWR) accidents and fission product behavior. Reference 1 is applicable to LWR designs and is intended to form the basis for the development of regulatory guidance.

The new source terms as described in NUREG-1465 (Reference 1) are being used to calculate the offsite radiological consequences for the Indian Point Nuclear Generating Station Unit No. 2. The accidents analyzed assumed releases from the fuel-cladding gap for which assumptions have changed with the implementation of Reference 1. The radiological consequences for the following accidents were determined: Locked Rotor, Rod Ejection, Fuel Handling and Loss of Coolant Accident (LOCA).

In addition to using the new source term methods, the analyses were also performed to support relaxation in operational requirements. These relaxations include emergency containment filtration and spray additive elimination, reduction in Technical Specification time before fuel movement to 100 hours following shutdown, and allowance of the personnel air lock to be open during refueling operation.

The radiological analysis following a LOCA is detailed in WCAP-14542, "Evaluation of the Radiological Consequences from a Loss of Coolant Accident at Indian Point Nuclear Generating Station Unit No. 2 Using NUREG-1465 Source Term Methodology". As such, the LOCA analysis is not discussed in detail in this report.

1.2 Common Analysis Inputs and Assumptions

The assumptions and inputs described below are common to the Locked Rotor, Rod Ejection and Fuel Handling accidents discussed in this report. Accident specific inputs and assumptions are discussed in Section 2 through 4. The LOCA analysis inputs and assumptions are discussed in detail in WCAP-14542.

The thyroid, γ-body and total effective dose equivalent (TEDE) doses are determined at the site boundary (SB) and low population zone (LPZ). The thyroid doses are based on the dose conversion factors (DCFs) provided in Reference 2. The DCFs used in determining the committed effective dose equivalent (CEDE) are from Reference 3. The TEDE dose is equivalent to the CEDE dose plus the acute dose for the duration of exposure to the cloud. The thyroid and CEDE dose DCFs are given in Table 1.

The γ-body doses are based on average γ disintegration energies from Reference 4 for the iodine and noble gas isotopes and from Reference 5 for the alkali metals group isotopes. The average disintegration energies are listed in Table 3.

The breathing rates (from Reference 6) and the atmospheric dispersion factors (from Reference 7) used in the dose calculations are given in Table 3.

2.0 LOCKED ROTOR RADIOLOGICAL CONSEQUENCES

2.1 Introduction

An instantaneous seizure of a reactor coolant pump rotor is assumed to occur which rapidly reduces flow through the affected reactor coolant loop. Fuel clad damage is assumed to occur as a result of this accident. Due to the pressure differential between the primary and secondary systems, and assumed SG tube leaks, fission products are discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the atmospheric relief valves or safety valves. In addition, iodine activity is contained in the secondary coolant prior to the accident and some of this activity is released to atmosphere as a result of steaming of the SGs following the accident. This section describes the assumptions and analyses performed to determine the amount of radioactivity released and the offsite doses resulting from this release.

2.2 Input Parameters and Assumptions

The analysis of the locked rotor event radiological consequences uses the source terms outlined in NUREG-1465 (Reference 1) and assumes a pre-accident iodine spike in the reactor coolant system. For the pre-accident iodine spike, it is assumed that a reactor transient has occurred prior to the locked rotor and has raised the RCS iodine concentration to 60 μCi/gm of dose equivalent (DE) I-131.

The noble gas and alkali metals group (i.e., cesium and rubidium isotopes) activity concentrations in the RCS at the time the accident occurs are based on a fuel defect level of 1.0%. The iodine activity concentration of the secondary coolant at the time the locked rotor occurs is assumed to be equivalent to 0.10 μCi/gm of DE I-131.

As a result of the locked rotor event, less than 2.5% of the fuel rods in the core undergo DNB (Reference 8). In determining the offsite doses following the locked rotor, it is conservatively assumed that 5% of the fuel rods in the core suffer sufficient damage that all of their gap activity is released to the RCS. The percentage of the total core activity assumed to be in the fuel-cladding gap was derived from Reference 1. NUREG-1465 defines a 3% gap fraction that should be applied to those events where long-term fuel cooling is maintained. To address high burnup fuel, an additional 20% increase in the gap fraction should be applied as described in References 9 and 10. A total of 3.6% of the total core activity was assumed to be present in the fuel-cladding gap for the iodines, alkali metals group and noble gases.

The primary to secondary SG tube leak used in the analysis is 1.0 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.

A partition factor in the SGs of 0.01 (curies I/gm steam) / (curies I/gm water) is used for both iodine and the alkali metals group isotopes.

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

At 8 hours after the accident, the RHR System is assumed to be placed into service for heat removal, and there are no further steam releases to atmosphere from the secondary system.

The major assumptions and parameters used in the analysis are itemized in Table 4.

2.3 Acceptance Criteria

The thyroid and γ-body dose limits for a locked rotor are a "small fraction of" the 10 CFR 100 guidelines, or 30 rem thyroid and 2.5 rem γ-body.

The proposed limit for the TEDE dose is 2.5 rem (Reference 11).

2.4 Results

The offsite thyroid, γ-body and TEDE doses due to the locked rotor are given in Table 5. The offsite doses are within the acceptance criteria.

3.0 ROD EJECTION ACCIDENT RADIOLOGICAL CONSEQUENCES

3.1 Introduction

It is assumed that a mechanical failure of a control rod mechanism pressure housing has occurred, resulting 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 are 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 either the main condenser or the atmospheric relief valves for the main steam safety valves. Iodine and alkali metals group activity is contained in the secondary coolant prior to the accident, and some of this activity is released to atmosphere as a result of steaming the SGs following the accident. Finally, radioactive reactor coolant is discharged to the containment via the spill from the reactor vessel head. A portion of this radioactivity is released through containment leakage to the environment. This section describes the assumptions and analyses performed to determine the amount of radioactivity released and the offsite doses resulting from the rod ejection accident.

3.2 Input Parameters and Assumptions

The analysis of the rod ejection accident radiological consequences uses the source terms described in NUREG-1465 (Reference 1). A pre-accident iodine spike in the reactor coolant system is assumed in the analysis prior to the rod ejection. A reactor transient caused the iodine spike, and the RCS iodine concentration was raised to 60 μCi/gm of dose equivalent (DE) I-131.

The alkali metals and noble gas activity concentrations in the RCS at the time the accident occurs are based on a fuel defect level of 1.0%. The iodine activity concentration of the secondary coolant at the time the rod ejection accident occurs is assumed to be equivalent to 0.10 μCi/gm of DE I-131.

As a result of the rod ejection accident, less than 10% of the fuel rods in the core undergo DNB. In determining the offsite doses following rod ejection accident, it is conservatively assumed that 10% of the fuel rods in the core suffer sufficient damage that all of their gap activity is released to the RCS. Five percent of the total core activity for iodines, alkali metals and noble gases is assumed to be in the fuel-cladding gap. NUREG-1465 (Reference 1) does not propose a gap release magnitude applicable to fission product releases resulting from reactivity insertion accidents such as the rod ejection accident. As such, the maximum gap fraction discussed in NUREG-1465 (5%) was used in the rod ejection calculation.

A small fraction (i.e., 0.25%) of the fuel in the core is assumed to melt as a result of the rod ejection accident. One-half of the iodine activity in the melted fuel is released to the RCS, while all of the alkali metals group and the noble gas activity in the melted fuel is released to the RCS.

Conservatively, all the iodine, alkali metals group and noble gas activity (from prior to the accident and resulting from the accident) is assumed to be in the RCS when determining offsite doses due to the primary to secondary SG tube leakage. All of the iodine, alkali metal and noble gas activity is assumed to be in the containment when determining offsite doses due to containment leakage.

The primary to secondary SG tube leak used in the analysis is 1.0 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.

An iodine partition factor in the SGs of 0.01 curies/gm steam per curies/gm water is used (Reference 12). This partition factor was also used for the alkali metal activity in the SGs.

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

The steam release from the SGs following the rod ejection accident is based on the maximum relief rate of 15.11E6 lb/hr through the main steam safety valves and a steam release duration of 82 seconds. This results in a steam release of 344,000 lbm.

The technical specification design basis containment leak rate of 0.1% by weight of containment air is used for the initial 24 hours. Thereafter, the containment leak rate is assumed to be one-half the design value, or 0.05%/day (Reference 13).

No credit is taken for iodine plateout on containment surfaces or by containment spray operation.

The major assumptions and parameters used in the analysis are itemized in Table 6.

3.3 Acceptance Criteria

The dose limits for a rod ejection accident are "well within" the 10 CFR 100 guideline values, or 75 rem thyroid and 6 rem γ-body. The proposed limit for the TEDE dose is 6 rem (Reference 11).

3.4 Results

The offsite thyroid and γ-body doses due to the rod ejection accident are given in Table 7.

4.0 FUEL HANDLING ACCIDENT RADIOLOGICAL CONSEQUENCES

4.1 Introduction

A fuel assembly is assumed to be dropped and damaged during refueling. Analysis of the accident is performed for the accident occurring both inside containment and in the fuel-handling building. Activity released from the damaged assembly is released to the outside atmosphere through either the containment purge system or the fuel-handling building ventilation systems. This section describes the assumptions and analyses performed to determine the amount of activity released and the offsite doses resulting from this release.

4.2 Input Parameters and Assumptions

The calculation of the offsite radiological consequences following a fuel handling accident (FHA) uses the source terms outlined in NUREG-1465 (Reference 1). The primary feature of the NUREG-1465 source term that is used for the FHA is the gap fraction. The chemical species of the elements are treated as in the UFSAR FHA analysis since this provides conservative results. The percentage of the total core activity assumed to be in the fuel-cladding gap was also derived from Reference 1. Per Reference 1, a 3 percent gap fraction should be applied to accidents where long-term fuel cooling is maintained despite fuel failure, such as a postulated spent fuel-handling accident. To address high burnup fuel, an additional 20% increase in the gap fraction was applied (References 9 and 10). Therefore, 3.6 percent of the total core activity was assumed in the gap for the iodines, the alkali metals group and the noble gases in the fuel-handling accident analyses.

It was assumed that all of the fuel rods in the equivalent of one assembly are damaged to the extent that all their gap activity is released.

One hundred hours of radioactive decay was assumed in this analysis. This decay time prior to refueling is significantly less than the current Technical Specification value of 174 hours. The analysis supports movement of fuel at 100 hours.

Per technical specifications, it is assumed that there is a minimum of 23 feet of water above the reactor pressure vessel flange. With this water depth, decontamination factors (DF) of 133 for elemental iodine and 1 for methyl iodine are used for pool scrubbing (Reference 14). NUREG-1465 (Reference 1) considers a significant fraction of the iodines and alkali metals as particulates (95%). There is a low probability that particulates would be released from the fuel-clad gap since it is more likely that the particulates would bound with the fuel or the cladding. In addition, particulates have a high probability of scrubbing in the fuel pool. Because of the above, it is expected that modeling the iodines and alkali metals as particulates would lead to a significant reduction in the offsite dose consequences. The iodine and alkali metal activity in the fuel rod gap are therefore assumed to be 99.75% elemental and 0.25% methyl as outlined in Regulatory Guide 1.25 (Reference 14). The resulting overall pool scrubbing DF for the iodines and alkali metals is 100.

All of the noble gas released from the damaged assembly is assumed to be released from the pool water (i.e., the pool scrubbing DF is 1) (Reference 14).

A conservatively high radial peaking factor of 1.7 is assumed for the damaged assembly.

No credit is taken for filtration of iodine in either the FHA accident analysis inside containment or the FHA in the fuel-handling building. 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 immediately released to the outside atmosphere. Since no filters or containment isolation is modeled, this allows the personnel air lock to be open during refueling operations.

The major assumptions and parameters used in the analysis are itemized in Table 8. The breathing rates and atmospheric dispersion factors used in the dose calculations are given in Table 3. Only the breathing rate and atmospheric dispersion factors from 0-2 hours are assumed in the analysis. 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.

4.3 Description of Analyses Performed

The activity releases and offsite doses are determined for the a FHA inside containment and the FHA in the fuel-handling building.

4.4 Acceptance Criteria

The dose limits for a FHA are "well within" the guideline values of 10CFR100, or 75 rem thyroid and 6 rem γ-body. The proposed limit for the TEDE does is 6 rem (Reference 11).

4.5 Results

The offsite thyroid and body doses due to the FHA are given in Table 9.

5.0 LOSS-OF-COOLANT ACCIDENT RADIOLOGICAL CONSEQUENCES

5.1 Introduction

An abrupt failure of the main reactor coolant pipe is assumed to occur and it is assumed that the emergency core cooling features fail to prevent the core from experiencing significant degradation (i.e., melting). This sequence cannot occur unless there are multiple failures and thus goes beyond the typical design basis accident that considers a single active failure. Activity that is released to the containment is assumed to be released to the environment due to the containment leaking at its design rate. The discussion in this section is brief since the accident is covered in detail in WCAP-14542, "Evaluation of the Radiological Consequences from a Loss of Coolant Accident at Indian Point Nuclear Generating Station Unit No. 2 Using NUREG-1465 Source Term Methodology".

5.2 Input Parameters and Assumptions

The calculation of the radiological consequences of a LOCA uses the source term methodology defined in NUREG-1465 (Reference 1). The use of NUREG-1465 results in several major departures from the assumptions used in the existing LOCA dose analysis as reported in the FSAR. Instead of assuming instantaneous melting of the core and release of activity to the containment, the release of activity from the core occurs over a 1.8 hour interval. Instead of the iodine being primarily in the elemental form, the iodine is mainly in the form of cesium iodide which exists as particulate and the fraction that is in the organic form is much smaller.

A detailed list of calculational assumptions and input are provided in WCAP-14542.

5.3 Acceptance Criteria

The offsite dose limits for the LOCA are that they meet the dose guidelines of 10 CFR 100. These limits are 300 rem thyroid and 25 rem γ-body. The proposed limit for the TEDE dose is 25 rem (Reference 11).

The control room dose limits for the LOCA are 30 rem thyroid, 5 rem γ-body, and 30 rem β-skin. The proposed limit for the TEDE dose is 5 rem.

5.4 Results

The acceptance criteria are met. See Table 10 for the doses.

6.0 CONCLUSIONS

The offsite doses due to the locked rotor, rod ejection, fuel handling and loss-of-coolant accidents do not exceed the acceptance criteria.

This report supports the relaxation of the Technical Specification time to handle fuel during refueling from 174 hours to 100 hours. The report also supports refueling operations with the personnel air lock open. Details on the elimination of the Emergency Containment filtration and spray additive systems can be found in WCAP-14542.

7.0 REFERENCES

1. U. S. Nuclear Regulatory Commission NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants", February 1995.

2. International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers", ICRP Publication 30, 1979.

3. EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion", EPA-520/1-88-020, September 1988.

4. Letter NS-PL-SL-90-099, "System Licensing Work Standards", J. L. Grover, April 16, 1990.

5. International Commission on Radiological Protection, "Radionuclide Transformations, Energy and Intensity of Emissions", ICRP Publication 38, 1983.

6. U. S. Atomic Energy Commission, Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors", Rev. 2, June 1974.

7. Indian Point Unit 2 Updated FSAR, Table 14.3-46.

8. Indian Point Unit 2 Updated FSAR, Section 14.1.6.5.3.

9. NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors", D. A. Baker, et al, February 1988.

10. Federal Register/Vol. 53, No. 39/Monday, February 29, 1988/Pages 6040 through 6043.

11. U. S. Nuclear Regulatory Commission, SECY-94-194, "Proposed Revisions to 10 CFR Part 100 and 10 CFR Part 50, and New Appendix S to 10 CFR Part 50", July 27, 1994.

12. NUREG-0800, Standard Review Plan 15.6.3, "Radiological Consequences of a Steam Generator Tube Rupture (PWR)", Rev. 2, July 1981.

13. U. S. AEC Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors", May 1974.

14. 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.

TABLE 1

DOSE CONVERSION FACTORS

I. Thyroid Dose

Isotope	DCF (rem/curie)
I-131	1.07E6
I-132	6.29E3
I-133	1.81E5
I-134	1.07E3
I-135	3.14E4

II. CEDE Dose

Isotope	DCF (rem/curie)
I-131	3.29E4
I-132	3.81E2
I-133	5.85E3
I-134	1.31E2
I-135	1.23E3
Cs-134	4.62E4
Cs-136	7.33E3
Cs-137	3.19E4
Rb-86	6.63E3

TABLE 2

AVERAGE GAMMA DISINTEGRATION ENERGIES

Isotope	$\bar{E}$ (mev/dis)
I-131	0.38
I-132	2.2
I-133	0.6
I-134	2.6
I-135	1.4
Cs-134	1.55
Cs-136	2.16
Cs-137	0.564
Rb-86	0.0945
Kr-85m	0.16
Kr-85	0.0023
Kr-87	0.79
Kr-88	2.2
Xe-131m	0.0029
Xe-133m	0.02
Xe-133	0.03
Xe-135m	0.43
Xe-135	0.25
Xe-138	1.2

TABLE 3

BREATHING RATES AND ATMOSPHERIC DISPERSION FACTORS

I. Breathing Rates

Time Period (hr)	Breathing Rate (m^3/sec)
0-8	3.47E-4
8-24	1.75E-4
24-720	2.32E-4

II. Atmospheric Dispersion Factors, sec/m^3

Site Boundary (0-2 hr)	7.54E-4
Low Population Zone	
0-8 hr	3.5E-4
8-24 hr	1.2E-4
24-96 hr	4.2E-5
96-720 hr	9.3E-6

TABLE 4

ASSUMPTIONS USED FOR LOCKED ROTOR DOSE ANALYSIS

Power	3216.5 MWt
Reactor Coolant Noble Gas and Alkali Metals Group Activity Prior to Accident	1.0% Fuel Defect Level
Reactor Coolant Iodine Activity Prior to Accident	60 μCi/gm of DE I-131
Activity Released to Reactor Coolant from Failed Fuel (Noble Gas, Alkali Metals Group & Iodine)	5.0% of Core Gap Activity
Fraction of Core Activity in Gap (Noble Gas, Alkali Metals Group & Iodine)	0.036
Secondary Coolant Iodine Activity Prior to Accident	0.10 μCi/gm of DE I-131
Total SG Tube Leak Rate During Accident	1.0 gpm
SG Iodine and Alkali Metals Group Partition Factor	0.01
Duration of Activity Release from Secondary System	8 hr
Offsite Power	Lost
Steam Release from SGs to Environment	562,000 lb (0-2 hr) 936,100 lb (2-8 hr)

TABLE 5

LOCKED ROTOR OFFSITE DOSES

	Dose (Rem)	
	SB (0-2 Hr)	LPZ (0-8 Hr)
Thyroid	6.9E-1	2.7E-0
γ-Body	8.0E-2	7.4E-2
TEDE	1.1E-1	1.9E-1

TABLE 6

ASSUMPTIONS USED FOR ROD EJECTION ACCIDENT DOSE ANALYSIS

Power	3216.5 MWt
Reactor Coolant Alkali Metal and Noble Gas Activity Prior to Accident	1.0% Fuel Defect Level
Reactor Coolant Iodine Activity Prior to Accident	60 μ/Ci/gm of DE I-131
Activity Released to Reactor Coolant AND Containment from Failed Fuel (Alkali Metal, Noble Gas & Iodine)	5.0% of Core Gap Activity
Activity Released to Reactor Coolant AND Containment from Melted Fuel	
Iodine	0.125% of Core Activity
Noble Gas & Alkali Metal	0.25% of Core Activity
Secondary Coolant Activity Prior	
Iodine	0.10 μCi/gm of DE I-131
Total SG Tube Leak Rate During Accident	1.0 gpm
Iodine and Alkali Metal Partition Factor in SGs	0.01
Steam Release from SGs	344,000 lbm (0-82 sec)
Containment Free Volume	2.60×10^6 ft^3
Containment Leak Rate	
0-24 hr	0.1%/day
>24 hr	0.05%/day

TABLE 7

ROD EJECTION ACCIDENT OFFSITE DOSES (REM)

	EB (0-2 Hr)	LPZ (0-30 Day)
Thyroid	1.7E1	5.0E1
γ-Body	1.1E-1	1.5E-1
TEDE	8.0E-1	2.4E0

TABLE 8

ASSUMPTIONS USED FOR FHA DOSE ANALYSIS

Power	3216.5 MWt
Radial Peaking Factor	1.7
Damaged Fuel	1 Fuel Assembly
Fuel Rod Gap Fractions	0.036 for iodines, alkali metals and noble gases
Percent of Gap Activity Released	100%
Pool Decontamination Factors	
Elemental Iodine	133
Methyl Iodine	1
Noble Gas	1
Iodine Species in Fuel Rod Gap	
Elemental Iodine	99.75%
Methyl Iodine	0.25%
Minimum Water Depth Above Reactor Pressure Vessel Flange	23 feet
Filter Efficiency	no filtration assumed
Containment Isolation	no containment isolation assumed

(1) Regulatory Guide 1.4

TABLE 9

FUEL HANDLING ACCIDENT OFFSITE DOSES

	Dose (Rem)	
	EB (0-2 Hr)	**LPZ (0-2 Hr)**
Thyroid	56	26
γ-Body	0.24	0.11
TEDE	3.0	1.4

TABLE 10

LOSS OF COOLANT ACCIDENT DOSES

	Dose (rem)		
	EB (0-2 Hr)	**LPZ (30 day)**	**CR (30 day)**
Thyroid	224	147	24
γ-Body	1.9	2.7	0.4
β-Skin	not specified	not specified	15
TEDE	13.4	10.2	1.5