14.24 MAXIMUM HYPOTHETICAL ACCIDENT

14.24.1 GENERAL

The MHA involves a gross release of fission products from the fuel to the Containment. During an accidental release, air containing radionuclides may enter the Control Room through inleakage into the Control Room ventilation system. The Control Room TEDE doses from inleakage and shine must meet 10 CFR Part 50, Appendix A, General Design Criteria 19 limits. Similarly, during an accidental release, air containing radionuclides may travel offsite. The offsite TEDE doses must meet 10 CFR 50.67 limits. The Control Room and offsite TEDE doses were calculated with the dose conversion factors extracted from References 5 and 6.

The design-basis MHA utilizes the AST methodology of 10 CFR 50.67 and Reference 1 to calculate offsite and Control Room doses for an MHA. Per Reference 1, the TEDE analysis should include all sources of radiation that will cause exposure to Control Room personnel; including the following pathways:

- Containment pathway
- Hydrogen Purge Line pathway
- Ventilation Stack pathway
- RWT pathway
- Containment Shine
- Plume Shine
- Control Room Filter Shine

The results of the design-bases MHA AST analysis (Reference 2) were submitted to the NRC in Reference 3. The NRC subsequently approved the license amendment request in Reference 4.

14.24.2 METHOD OF ANALYSIS

14.24.2.1 <u>Control Room</u>

The main Control Room inleakage points include the West Road inlets, the Turbine Building, and Access Control Units 11 and 13 on the Auxiliary Building roof. Installation of automatic isolation dampers and radiation monitors at the Access Control Units 11 and 13 on the Auxiliary Building roof were credited.

Assumptions used are:

- The Control Room volume is 289194 ft^3.
- A Control Room inleakage rate of 3500 cfm was based on measured inleakage measurements.
- Control Room recirculation filtration is credited assuming 10,000 $\pm$ 10% cfm flow at 90% filter efficiency for elemental and organic iodine and 99% for particulates for a 20 minute delay time.
- 0-8, 8-24, and 24-720 hour breathing rates of 3.5E-04, 1.8E-04, and 2.3E-04 m^3/sec are assumed.
- 0-24, 24-96, and 96-720 hour Control Room occupancy factors of 1.0, 0.6, and 0.4 are assumed.

14.24.2.2 <u>Source Terms</u>

The inventory of fission products in the reactor core and available for release to the containment atmosphere is based on the maximum full power operation of the core with current licensed values for fuel enrichment (5.0 w/o), fuel burnup (62,000 MWd/MTU), and core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty (2754 MWt). The period of irradiation was of sufficient duration to allow the activity of dose-significant radionuclides to reach maximum values. The isotopic activities released from the failed fuel were generated utilizing the isotope generation and depletion computer code SAS2H/ORIGEN-S.

The core inventory release onset, duration, and fractions by radionuclide groups for the gap release phase (0.5-30 minutes) and early in-vessel damage phase (30-108 minutes) for a DBA LOCA were extracted from Reference 1. The activity released from each release phase is modeled as increasing linearly over the duration of the phase. The release fractions for the gap release and early in-vessel damage phases are as follows:

	Gap Release	Early Invessel Damage
• Noble Gases (Xe, Kr)	0.05	0.95
• Halogens (I, Br)	0.05	0.35
• Alkali Metals (Cs Rb)	0.05	0.25
• Tellurium Metals (Te, Sb, Se)	0.00	0.05
• Ba, Sr	0.00	0.02
• Noble Metals (Ru, Rh, Pd, Mo, Tc, Co)	0.00	0.0025
• Cerium Group (Ce, Pu, Np)	0.00	0.0005
• Lanthanides, (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am)	0.00	0.0002

Per Reference 1, of the radioiodine released from the RCS to the containment atmosphere in a LOCA, 95% of the iodine released should be assumed to be particulate iodine, 4.85% elemental iodine, and 0.15% organic iodine. This includes releases from the gap and fuel pellets. With the exception of elemental and organic iodine and noble gases, all other fission products should be assumed to be in particulate form.

14.24.2.3 Containment Pathway

Per Reference 8, the total release from Containment may be apportioned between the exposed and enclosed building surfaces. 72% of the released airborne activity post-MHA is assumed to leak out of the Containment through the containment walls.

Assumptions used are:

- The Containment volume is 1.989E+06 ft^3.
- Two 55,000 cfm cooling units are credited after a 60 second activation delay.
- Two containment iodine removal units are credited after a 63 second activation delay for the first unit and a 20 minute activation delay for the second unit. Each unit consists of activated charcoal filters preceded by HEPA filters. Each filter unit has a 20,000 $\pm$ 10% cfm flowrate with a filter efficiency of 90% for elemental and particulate species and 30% for organic species per Reference 9.
- Reduction in aerosol airborne radioactivity in the Containment by natural deposition within the Containment was credited per Reference 1. The 10th percentile Powers aerosol decontamination model (Reference 10) was

utilized. Aerosol particles grow by coagulating with other aerosol particles or because steam condenses on them thus, gravitational settling of aerosols is usually the dominant aerosol removal process.

- Reduction in airborne radioactivity by containment spray systems that have been designed and maintained in accordance with Reference 11 may be credited.
  - The spray removal constant for aerosols is 3.414/hr with no maximum decontamination factor but with a 90 second activation delay.
  - The spray removal constant for elemental iodine is 14.816/hr with a maximum decontamination factor of 14.04 and with a 90 second activation delay.
  - Organic iodides are not removed by spray.
- The maximum allowable containment leakage rate L_a contained in the Containment Leakage Rate Testing Program of Technical Specification 5.5.16 was assumed, 0.16 percent per day at P_a. Per Reference 1, the Containment is assumed to leak at the leak rate incorporated in the Technical Specifications for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident.
- The 0-2 hour Containment-to-EAB atmospheric dispersion coefficient (χ/Q) was calculated to be 1.30E-04 sec/m^3.
- The 0-2, 2-24, and 24-720 hour Containment-to-LPZ atmospheric dispersion coefficients were calculated to be 3.30E-05, 2.20E-06, and 5.40E-07 sec/m^3, respectively.
- The worst-case 0-2, 2-8, 8-24, 24-96, and 96-720 hour Containment-to-Control Room atmospheric dispersion coefficients were calculated to be 1.11E-03, 7.29E-04, 3.19E-04, 2.36E-04, and 1.98E-04 sec/m^3, respectively, using the ARCON96 computational methodology.

14.24.2.4 Ventilation Stack Pathway

Per Reference 8, the total release from Containment may be apportioned between the exposed and enclosed building surfaces. Containment leakage is more likely at penetrations rather than through liner plates or weld joints. Penetration rooms are built adjacent to the outside surface of each Containment and enclose the areas around the majority of the penetrations. Thus, a fraction of the containment leakage will leak into the Auxiliary Building penetration rooms, be processed by the penetration room emergency ventilation system, and be expelled to the atmosphere through the ventilation stacks. 28% of the released airborne activity post-MHA is assumed to leak out of the Containment through the containment penetrations into the Auxiliary Building penetration rooms.

Assumptions used are:

- The containment volume, circulation, and cleanup mechanisms are the same as described previously for the containment pathway.
- The 0-2 hour ventilation stack-to-EAB χ/Q was calculated to be 1.44E-04 sec/m^3.
- The 0-2, 2-24, and 24-720 hour ventilation stack-to-LPZ χ/Qs were calculated to be 3.39E-05, 2.20E-06, and 5.40E-07 sec/m^3, respectively.
- The worst case 0-2, 2-8, 8-24, 24-96, and 96-720 hour ventilation stack-to-Control Room χ/Qs were calculated to be 1.68E-03, 1.34E-03, 5.14E-04, 3.84E-04, and 3.12E-04 sec/m^3, respectively, using the ARCON96 computational methodology.

- The Penetration Room Emergency Ventilation System is designed to collect and process containment penetration leakage, so as to reduce to a minimum the environmental radioactivity levels from post-accident containment leaks. To minimize the release of radioactive material to the environment, penetration room ventilation is continuously routed through a prefilter, a HEPA filter, and an activated charcoal filter, positioned in series. Following a LOCA, a containment isolation signal will start both the two full-size blowers. The entire system is designed to operate under negative pressure up to the fan discharge. Per Technical Specification 5.5.11, each Penetration Room Emergency Ventilation System unit has a design flowrate of 2,000 ± 10% cfm with an efficiency of 90% for elemental particulate species and 30% for organic species.

14.24.2.5 Hydrogen Purge Line Pathway

Per Reference 1, if the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The containment 4" hydrogen purge line is used for an unlimited amount of pressure and containment radioactivity control purposes; however, the vent isolation valves may also be opened for surveillance testing. The 4" vent line isolates on a SIAS and containment radiation signal.

Assumptions used are:

- The purge release evaluation assumes that 100% of the radionuclide inventory in the RCS liquid is released to the containment atmosphere at the initiation of the LOCA. This inventory is based on the Technical Specification RCS equilibrium activity of 0.5 μCi/gm DEQ I-131. Iodine spikes need not be considered. The purge system is isolated before the onset of the gap release phase, thus release fractions associated with gap release and early in-vessel phases need not be considered.
- 100% of the Technical Specification primary iodine and noble gas activities are assumed to be released instantaneously and homogeneously throughout the containment atmosphere at the initiation of the accident.
- The radioiodine that is postulated to be available for releases from the RCS to the environment should be assumed to be 97% elemental and 3% organic.
- The hydrogen purge line flow rate value is conservatively assumed to be 1645 cfm.
- The χ/Q and Penetration Room Emergency Ventilation System values are identical to those for the ventilation stack pathway.

14.24.2.6 Refueling Water Tank Pathway

There is a potential for an unmonitored release pathway resulting from the post-LOCA leakage of isolation valves in the safety injection and containment spray system recirculation lines to the RWT, which is vented directly to the atmosphere (Reference 7). During the recirculation phase, sump water is recirculated through the ECCS pumps and could leak through various valves and reach the RWT. The two pathways include the two valves in series in the minimum flow recirculation line header (MOV659/660) and the valve from the containment spray pumps (SI459).

Assumptions used are:

- The liquid volume of the containment sump is 68,329 ft^3.
- The air volume of the RWT is 52,109.75 ft^3.
- The minimum time to RAS is 30 minutes.
- The leakage through SI459 and MOV659/660 is limited to 1,000 cc/hr. Per Reference 1, this leakage must be doubled to account for valve degradation between testing.
- A 10% flashing fraction is conservatively assumed.
- A 4.2 cfm leakrate from the RWT atmosphere to the environment is assumed.
- Radioiodine that is postulated to be available for releases from the RWT to the environment is assumed to be 97% elemental and 3% organic.
- The 0-2 hour RWT-to-EAB χ/Q was calculated to be 1.44E-04 sec/m^3.
- The 0-2, 2-24, and 24-720 hour RWT-to-LPZ χ/Qs were calculated to be 3.39E-05, 2.20E-06, and 5.40E-07 sec/m^3, respectively.
- The worst-case 0-2, 2-8, 8-24, 24-96, and 96-720 hour RWT-to-Control Room χ/Qs were calculated to be 2.57E-03, 2.13E-03, 8.50E-04, 5.71E-04, and 4.85E-04 sec/m^3, respectively, using the ARCON96 computational methodology.

14.24.2.7 Containment Shine

Per Reference 1, the MHA analysis should consider radiation shine from radioactive material in the Containment.

Assumptions used are:

- All of the failed fuel isotopics emitted during the gas release and early in-vessel damage phases are assumed to be released into the containment atmosphere at the MHA initiation.
- These isotopic values together with the Containment and Control Room geometries are input into the Microshield point kernel computer code to calculate containment shine doses in the Control Room. Note that Microshield will allow parent decay and daughter buildup, so that subsequent isotopic quantities as a function of decay time are automatically calculated by Microshield.
- No removal of airborne radioactivity is assumed except for decay.

14.24.2.8 Plume Shine

Per Reference 1, the MHA analysis should consider shine from the external radioactive plume released from Containment.

Assumptions used are:

- All of the failed fuel isotopics emitted during the gas release and early in-vessel damage phases are assumed to be released into the containment atmosphere at the MHA initiation.
- No removal of airborne radioactivity is assumed except for decay.
- The maximum allowable containment leakage rate L_a contained in the Containment Leakage Rate Testing Program of Technical Specification 5.5.16 was assumed, 0.16% per day at P_a. Per Reference 1, the Containment is assumed to leak at the leak rate incorporated in the Technical Specifications for the first 24 hours, and at 50% of this leak rate for

the remaining duration of the accident. Thus the fraction of the containment activity that leaks from the Containment over 30 days can be calculated to be 0.0016+0.0008*29=0.0248.

- These isotopic values are input into the Microshield point kernel computer code to calculate plume shine doses in the Control Room. The Microshield results assume that all of the released activity is released from Containment at the beginning of the accident and sits over the Control Room for 30 days. To correct for release timing and dispersion via wind, a dilution factor is calculated which conservatively assumes that all containment release passes directly over the Control Room at a wind speed that is one-tenth the wind speed integrated over 8 years.

14.24.2.9 Control Room Filter Shine

Per Reference 1, the analysis should consider radiation shine from the radioactive material buildup of the Control Room recirculation filters.

Assumptions used are:

- All of the failed fuel isotopics emitted during the gas release and early in-vessel damage phases are assumed to be released into the containment atmosphere at the MHA initiation.
- The maximum allowable containment leakage rate is 0.16% per day at P_a. Per Reference 1, the Containment should be assumed to leak at the leak rate incorporated in the Technical Specifications for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident.
- For the Control Room filter shine calculations, the containment atmosphere is cleansed by the iodine removal system, which filters particulates and elemental and organic iodine and by spray, which removes particulates.
- The activity released to the environment is transported to the Control Room via appropriate atmospheric dispersion coefficients.
- The Control Room inleakage is a constant 3,500 cfm in this analysis.
- Control Room filtration is credited based on a recirculation flow at a nominal 10,000 cfm. A charcoal filter efficiency of 100% is credited for elemental, organic, and particulate iodine.
- The resulting isotopic values are input into Microshield to calculate Control Room filter shine. Note that Microshield will allow parent decay and daughter buildup, so that subsequent isotopic quantities as a function of decay time are automatically calculated by Microshield.

14.24.3 RESULTS

The EAB, LPZ, and Control Room doses for the design-basis MHA are detailed in the following table.

MHA Results

Results	**EAB Rem**	**LPZ Rem**	**Control Room Rem**
Containment Pathway	1.70	0.423	3.780
Penetration Room Pathway	0.16	3.5 E-02	0.37
RWT Pathway	3.7 E-05	1.9 E-03	0.33
Hydrogen Purge Pathway	6.5 E-05	1.5 E-05	7.7 E-05
Containment Shine			5.5 E-02
Plume Shine			3.0 E-03

MHA Results

Results	EAB Rem	LPZ Rem	Control Room Rem
Control Room Filter Shine			1.4 E-02
Total	1.86	0.46	4.57
Regulatory Limits	25	25	5

Note that all values are below the regulatory limits.

This event includes the transition to Advanced CE-14 HTP™ fuel (with Gd_2O_3 burnable poison irradiated to a maximum burnup of 62 GWd/MTU).

14.24.4 REFERENCES

1. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000
2. CCNPP Calculation CA06449, Revision 1, "Maximum Hypothetical Accident Using Alternative Source Terms," dated August 25, 2010
3. Letter from B. S. Montgomery (CCNPP) to Document Control Desk (NRC), License Amendment Request: Revision to Accident Source Term and Associated Technical Specifications, dated November 3, 2005
4. Letter from D. V. Pickett (NRC) to J. A. Spina (CCNPP), "Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Amendment Re: Implementation of Alternative Radiological Source Term (TAC Nos. MC8845 and MC8846)," dated August 29, 2007
5. Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988
6. Federal Guidance Report (FGR) 12, "External Exposure to Radionuclides in Air, Water, and Soil," September 1993
7. NRC Information Notice No. 91-56, "Potential Radioactive Leakage to Tank Vented to Atmosphere," dated September 19, 1991
8. Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessment at Nuclear Power Plants," dated June 2003
9. Regulatory Guide 1.52, Revision 2, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," dated March 1978
10. NUREG/CR-6189 SAND94-0407, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments", dated July 1996
11. NUREG-0800, SRP 6.5.2, Revision 2, "Containment Spray as a Fission Product Cleanup System," dated December 1988