Attachment 4
LR-N10-0163

H-1-CG-MDC-1795, Revision 5, "Control Rod Drop Accident Radiological Consequences"

ATTACHMENT 2
Design Analysis Minor Revision Cover Sheet

Design Analysis (Minor Revision) — Last Page No. [6] 24

Analysis No.: [1] H-1-CG-MDC-1795 — Revision: [2] 5

Title: [3] Control Rod Drop Accident Radiological Consequences

EC/ECR No.: [4] 80092568 AD M01R0 — Revision: [5] 0

Station(s): [7] Hope Creek

Unit No.: [8] N/A

Safety/QA Class: [9] SR

System Code(s): [10] N/A

Is this Design Analysis Safeguards Information? [11] Yes ☐ No ☒ If yes, see SY-AA-101-106

Does this Design Analysis contain Unverified Assumptions? [12] Yes ☐ No ☒ If yes, ATI/AR#:

This Design Analysis SUPERCEDES: [13] H-1-CG-MDC-1795, Rev 4 In its entirety.

Description of Changes (list affected pages): [14]

There are typos in calculation H-1-CG-MDC-1795, Rev 4, page 6, in the South Plant Vent χ/Qs for time intervals 24-96 hrs and 96-720 hrs. The correct values are 9.08E-05 s/m^3 and 7.01E-05 s/m^3 for time intervals 24-96 hrs and 96-720 hrs, respectively. The pages affected are 1, 1A, 2, 3, 4, 6, & 24.

Disposition of Changes: [15]

The typographic errors in calculation H-1-CG-MDC-1795, Rev 4, on page 6 are corrected to make the χ/Q values consistent with the design basis values.

Preparer: [16] Gopal J. Patel (Print Name) — (Sign Name) — 05/25/2007 (Date)

Method of Review: [17] Detailed Review ☒ Alternate Calculations ☐ Testing ☐

Reviewer: [18] Mark Drucker (Print Name) — (Sign Name) — 05/25/2007 (Date)

Review Notes: [19] Independent review ☐ Peer review ☐

(For External Analyses Only)
External Approver: [20] N/A (Print Name) — (Sign Name) — (Date)

PSE&G Reviewer [21] Michael E. Crawford (Print Name) — (Sign Name) — 06/05/07 (Date)

PSE&G Approver: [22] MOHAMMED ALVI (Print Name) — (Sign Name) — 6/7/07 (Date)

ATTACHMENT 2
Owners Acceptance Review Checklist for External Design Analysis
Page 1 of 1

DESIGN ANALYSIS NO. H-1-CG-MDC-1795 **REV:** 5

		Yes	No	N/A
1.	Do assumptions have sufficient rationale?	☒	☐	☐
2.	Are assumptions compatible with the way the plant is operated and with the licensing basis?	☒	☐	☐
3.	Do the design inputs have sufficient rationale?	☒	☐	☐
4.	Are design inputs correct and reasonable?	☒	☐	☐
5.	Are design inputs compatible with the way the plant is operated and with the licensing basis?	☒	☐	☐
6.	Are Engineering Judgments clearly documented and justified?	☒	☐	☐
7.	Are Engineering Judgments compatible with the way the plant is operated and with the licensing basis?	☒	☐	☐
8.	Do the results and conclusions satisfy the purpose and objective of the Design Analysis?	☒	☐	☐
9.	Are the results and conclusions compatible with the way the plant is operated and with the licensing basis?	☒	☐	☐
10.	Does the Design Analysis include the applicable design basis documentation?	☒	☐	☐
11.	Have any limitations on the use of the results been identified and transmitted to the appropriate organizations?	☒	☐	☐
12.	Are there any unverified assumptions?	☐	☒	☐
13.	Do all unverified assumptions have a tracking and closure mechanism in place?	☐	☐	☒
14.	Have all affected design analyses been documented on the Affected Documents List (ADL) for the associated Configuration Change?	☒	☐	☐
15.	Do the sources of inputs **and** analysis methodology used meet current technical requirements and regulatory commitments? (If the input sources **or** analysis methodology are based on an out-of-date methodology or code, additional reconciliation may be required if the site has since committed to a more recent code)	☒	☐	☐
16.	Have vendor supporting technical documents and references (including GE DRFs) been reviewed when necessary?	☒	☐	☐

PSE&G REVIEWER: MICHAEL E. CRAWFORD **DATE:** 06/05/07
Print / Sign

REVISION HISTORY

Revision	Description
0	Original Issue
1	Revised (see Order 70009023, Activity 0020) to provide information relative to: • Specific assumptions made (that is, the mechanical vacuum pumps are assumed to be tripped) • Evaluation against regulatory limits (that is, 10CFR100 and SRP Section 6.4 guidelines) • Explanation of any qualitative relationships to any other accidents described in the HCGS-UFSAR (that is, LOCA) Moreover, the analysis is revised to correct the TACT5 input error identified in Notification 20035343. Revision bars are not used due to the extent of the revision.
2	Revised (see Order 70020574, Activity 0010) to incorporate a revised 10CFR50.59 Screening relating to Revision 1 of this calculation.
3	Revised (see Order 70022227, Activity 0010) to incorporate a revised 10CFR50.59 Screening relating to Revision 1 of this calculation.
4	Complete revision to perform AST analysis for the EPU As of 12/07/2005, the EPU project decided to adopt the AST analysis performed for the increased core thermal power level for the current design and licensing bases because it conservatively bounds the EPU project design. Section 7.2 indicates that the proposed increase in the EAB and CR doses and total doses are less than the corresponding minimal dose increases and applicable regulatory allowable limits as defined in the 10 CFR 50.59 rule. The implementation or cancellation of the proposed core thermal power related DCP would not have any adverse impact on this analysis. Some of design inputs are taken from the documents that support higher core thermal power operation. If the HCGS license is not amended for the proposed increased power level, these design inputs would become conservative assumptions without having any adverse impact on the validity of this analysis
5	Corrected the typographic errors in the South Plant Vent χ/Q values for 24-96 hrs and 96-720 hrs on page 6.

PAGE REVISION INDEX

PAGE	REV	PAGE	REV
1	5	15	4
1A	5		
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1.0 PURPOSE:

The purpose of this calculation is to determine the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room (CR) doses due to a Control Rod Drop Accident (CRDA) using the Alternative Source Term (AST) and core thermal power level of 4,031 MW_t, including the instrument uncertainty.

2.0 BACKGROUND:

Hope Creek Technical Specification (TS) LIMITING CONDITION FOR OPERATION (LCO) 3/4.3.10 requires that two channels of the main steam line radiation – high, high function for the mechanical vacuum pump (MVP) trip shall be operable. This LCO 3/4.3.10 assures that the post-CRDA fission product release path to the environment would be through the main condenser.

The MVP trip is required to be OPERABLE in OPERATIONAL CONDITIONS 1 and 2 when any mechanical vacuum pump is in service (i.e., taking a suction on the main condenser) and any main steam line is not isolated, to mitigate the consequences of a postulated CRDA. In this condition fission products released during a CRDA could be discharged directly to the environment. Therefore, the MVP trip is necessary to assure conformance with this calculation's assumption that the post-CRDA radiological release path is via the condenser. In OPERATIONAL CONDITION 3, 4 or 5, the consequences of a CRDA are insignificant, and are not expected to result in any fuel damage or fission product releases. When the MVP is not in service or the main steam lines are isolated, fission product releases via the MVP pathway would not occur.

The function of MVP is to evacuate the condenser during startup. Operating Procedure HC.OP-SO.CG-0001(Q) (Ref. 9.11) includes Precaution 3.1.2, which identifies that operation of the mechanical vacuum pumps while radioactive steam is being admitted to the main condenser will result in high radiation levels at the south plant vent. The procedure also includes Limitation 3.2.4, which calls for securing the mechanical vacuum pumps from service and placing the steam jet air ejectors (SJAE) in service prior to reactor power exceeding 5%. The expected MVP response following a CRDA is to be automatically tripped due to either loss of offsite power or a main steam radiation monitor signal (Ref. 9.13). The post-CRDA activity release through the MVP during startup will be insignificant due to the MVP operation limited to 5% core power. For the post-CRDA release through the Gaseous Waste Management System (GWMS) including the SJAE, all of the iodine that enters the off-gas treatment system is retained indefinitely and does not contribute to the CR and off-site dose (Ref. 9.12, page 3). Therefore, the post-CRDA dose impact for the releases through the MVP during the startup at a low power level and GWMS during normal operation at a rated power level will be bounded by the post-CRDA release through the isolated condenser, which is analyzed in the following section.

3.0 ANALYTICAL APPROACH:

This analysis uses Version 3.02 of the RADTRAD computer code to calculate the potential radiological consequences of the CRDA. The RADTRAD code was developed by Sandia National Laboratories, the NRC's technical contractor, for the staff to use in establishing fission product transport and removal models and in estimating radiological doses at selected receptors at nuclear power plants. The RADTRAD code is documented

in NUREG/CR-6604 (Ref. 9.2). The RADTRAD code is maintained as Software ID Number A-0-ZZ-MCS-0225 (Ref. 9.15).

The consequences of a CRDA are analyzed using the as-built plant specific as-built design and licensing bases inputs, which are compatible to the AST and TEDE dose criteria. There is no specific ESF function credited in the analysis.

For the CRD accident, the release from the breached fuel is based on an NRC approved fuel vendor methodology for the number of fuel rods breached and the assumption that 10% of the core inventory of noble gases and iodine, and 12% of the core inventory of alkali metals are in the fuel gaps. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and on the assumption that 100% of the noble gases and 50% of the iodines contained in that fraction are released to the reactor coolant. The activities released from the fuel gaps and melted fuel are assumed to be instantaneously mixed in the reactor coolant within the pressure vessel. Of the activity released to the reactor coolant, 100% of the noble gases, 10% of the iodine, and 1% of the remaining radionuclides are assumed to reach the turbine and condenser. Of the activity that reaches the turbine and condenser, 100% of the noble gases, 10% of the iodine, and 1% of the remaining radionuclides are available for release to the environment. The turbine and condenser leak to the atmosphere as a ground-level release at a rate of 1% per day for a period of 24 hours, at which time the condenser leakage is assumed to terminate. No credit is taken for dilution or holdup within the turbine building. The post-CRDA activity from the turbine and condenser can be released to the turbine building (TB) and to the environment at ground level through the south plant vent when offsite power is available; and through the TB louvers/TB vent during a loss of offsite power (Refs. 9.16 & 9.17). The χ/Qs for these release paths are obtained from Reference 9.5, Section 8.0, and listed in the following table:

Time Interval (hr)	HCGS Control Room 95% Atmospheric Dispersion Factors (X/Qs) (s/m^3)		
	South Plant Vent	TB Louvers (s/m^3)	TB Vent (s/m^3)
0-2	5.75E-04	6.17E-04	3.48E-04
2-8	3.84E-04	4.00E-04	2.55E-04
8-24	1.40E-04	1.44E-04	9.11E-05
24-96	9.08E-05	1.00E-04	5.37E-05
96-720	7.01E-05	7.49E-05	3.82E-05

Comparison of χ/Qs in the above table indicates that the TB louvers release path is the most limiting release path for the 0 to 24 hour post-CRDA release prior to the condenser leakage being terminated. Therefore, the CR dose is calculated using the post-CRDA release through the TB louvers. The Control Room Emergency Filtration (CREF) system is not credited in the analysis. The CR is assumed to operate in a normal mode of operation with a normal HVAC inflow rate of 3,300 cfm (3,000 cfm + 10 % uncertainty) for the entire duration of the accident. The resulting doses at the EAB, LPZ, and CR locations are compared with the dose acceptance criteria in Section 7.0.

The core activity inventory is obtained from Reference 9.3, which is calculated based on a thermal power level of 4,031 MW_t. A radial peaking factor of 1.75 is conservatively used instead of the 1.5 value recommended in Reference 9.6. The isotopic activity available for release from the condenser are calculated in Tables 1 & 2 based on the core activity inventory obtained from Reference 9.3 and the CRDA failed and melted fuel fractions from Reference 9.12 (Section 6.2.2).

The RADTRAD V3.02 (Ref 9.2 & 9.15) default nuclide inventory file (NIF) Bwr_def. NIF is modified based on the isotopic activities calculated in Table 2. The newly developed plant-specific nuclide inventory file (HEPUCRDA_def.txt) is further modified to include Kr-83m, Xe-131m, Xe-133m, Xe-135m, Xe-138, Rb-88, and Cs-138 isotopes. The RADTRAD3.02 dose conversion factor (DCF) File (Fgr11&12) is modified to include the DCFs obtained from References 9.7 & 9.8 for the added noble gas isotopes. The modified DCF file HCRDA_FG11&12.txt is used in the CRDA analysis. The newly developed release fraction and timing file (HCRDA_RFT.txt) is used to postulate an instantaneous post-CRDA release. The NIF is developed based on the actual activity in curies released to the environment from the condenser; therefore, the thermal power level is set to unity in the RADTRAD input.

Determine Compliance of Increased Dose Consequences With 10CFR50.59 Guidance

Consistent with the RG 1.183, Section 1.1.1, once the initial AST implementation has been approved by the staff and has become part of the facility design basis, the licensee may use 10 CFR 50.59 and its supporting guidance in assessing safety margins related to subsequent facility modifications and changes to procedures. The NRC Safety Evaluation Report for Amendment 134 (Ref. 9.26) approved the AST for the HCGS licensing basis analyses.

An increase in control room, EAB or LPZ dose consequence is considered acceptable under the 10 CFR 50.59 rule if the magnitude of the increase is minimal (as defined by the guidance in Refs. 9.23 and 9.24), and if the total calculated dose is less than the allowable regulatory guide 1.183 dose limit. The current licensing basis analysis is documented in the calculation H-1-CG-MDC-1975, Rev 3. Of note is that the current licensing basis analysis does not calculate the LPZ dose consequence; as such, the concept of minimal dose increase cannot be applied to the LPZ dose evaluation. The increases in the proposed EAB and CR doses are compared with the 10 CFR 50.59 allowable minimal dose increases in Section 7.2. Similarly, the proposed calculated total EAB, LPZ, and CR doses are compared with the allowable regulatory guide dose limits. The comparisons in Section 7.2 confirm that the proposed increases in the EAB & CR doses and the total calculated EAB, LPZ and CR doses are less than the corresponding minimal dose increases and allowable regulatory guide limits, respectively. Therefore, pursuant to 10 CFR 50.59 guidance as defined in References 9.23 and 9.24, the proposed increase in the core thermal power level and resulting post-CRDA doses can be adopted as current design and licensing bases for the HCGS.

4.0 ASSUMPTIONS:

Assumptions for Evaluating the Radiological Consequences of a Control Rod Drop Accident (CRDA)

The assumptions in these sections are acceptable for evaluating the radiological consequences of a CRDA. These assumptions supplement the guidance provided in Regulatory Guide 1.183, Appendix C (Ref. 9.1). These assumptions are incorporated as design inputs in Sections 5.3 through 5.5 for the CRDA analysis.

Source Term Assumptions

4.1 Per Reference 9.12 (Section 6.2.1), in the event of a CRDA 850 fuel rods are breached, and 0.77 percent of these breached rods experience fuel melt. Per Reference 9.14 there are 764 fuel assemblies contained in the reactor core, and per Reference 9.20 there are 62 fuel rods in each reactor assembly.

4.2 Per Reference 9.1, Appendix C, Section 1, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodine is in the fuel gap, as incorporated in design input 5.3.1.7. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and on the assumption that 100% of the noble gases and 50% of the iodine contained in that fraction are released to the reactor coolant, as incorporated in design input 5.3.1.11. In addition, per Reference 9.1, Section 3.2, for non-LOCA events the release fraction of Alkali Metals from Table 3 is incorporated in Design Input 5.3.1.7 in conjunction with the core fission product inventory in Design Input 5.3.1.2 for the core thermal power level of 4,031 MW_t. The bromines are neglected from thyroid dose consideration due to their low thyroid dose conversion factors, relatively short half lives, and decaying into insignificant daughters.

4.3 Per Reference 9.1, Appendix C, Section 3.1, the activity released from either the gap or from fuel pellets is assumed to be instantaneously mixed in the reactor coolant within the pressure vessel.

4.4 Per Reference 9.1, Appendix C, Section 3.2, credit is not assumed for partitioning in the pressure vessel or for removal by the steam separators.

4.5 Per Reference 9.1, Appendix C, Section 3.3, of the activity released from the reactor coolant within the pressure vessel, 100% of the noble gases, 10% of the iodine, and 1% of the remaining radionuclides are assumed to reach the turbine and condensers, which is incorporated in the design input 5.3.1.8.

4.6 Per Reference 9.1, Appendix C, Section 3.4, of the activity that reaches the turbine and condenser, 100% of the noble gases, 10% of the iodine, and 1% of the particulate radionuclides are available for release to the environment, which is incorporated in design input 5.3.1.9. The turbine and condenser leak to the atmosphere as a ground-level release at a rate of 1% per day for a period of 24 hours, at which time the leakage is assumed to terminate (see design inputs 5.3.2.1 through 5.3.2.3). No credit is taken for dilution or holdup within the turbine building, which is incorporated in the design input 5.3.2.6. Radioactive decay during holdup in the turbine and condenser is assumed.

4.7 Per Reference 9.1, Appendix C, Section 3.6, the iodine species released from the reactor coolant within the pressure vessel is assumed to be 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic, which is incorporated in the design input 5.3.2.4. The release from the turbine and condenser is assumed to be 97% elemental and 3% organic, which is incorporated in the design input 5.3.2.5.

Offsite Dose Consequences:

The following guidance is used in determining the TEDE for a maximum exposed individual at EAB and LPZ locations:

4.9 The maximum EAB TEDE for any two-hour period following the start of the radioactivity release is determined (Ref. 9.1, Section 4.1.5), and used in determining compliance with the dose acceptance criteria in Reference 9.1, Section 4.4, Table 6:

EAB Dose Acceptance Criterion: 6.3 Rem TEDE

4.10 The breathing rates for persons at offsite locations are given in Reference 9.1, Section 4.1.3, and are incorporated in Design Input 5.3.4.

4.11 The maximum Low Population Zone (LPZ) TEDE is determined for the most limiting receptor at the outer boundary of the LPZ (Ref. 9.1, Section 4.1.6), and used in determining compliance with the dose criteria in Reference 9.1, Section 4.4 Table 6:

LPZ Dose Acceptance Criterion: 6.3 Rem TEDE

4.12 No correction is made for depletion of the effluent plume by deposition on the ground (Ref 9.1, Section 4.1.7).

Control Room Dose Consequences

The following guidance is used in determining the TEDE for maximum exposed individuals located in the control room:

4.13 The CR TEDE analysis considers the following sources of radiation that will cause exposure to control room personnel (Ref 9.1, Section 4.2.1):

- Contamination of the control room atmosphere by the intake or infiltration (i.e., filtered CR ventilation inflow via the CR air intake, and unfiltered inleakage) of the radioactive material contained in the post-accident radioactive plume released from the facility,
- Contamination of the control room atmosphere by the intake or infiltration (i.e., filtered CR ventilation inflow via the CR air intake, and unfiltered inleakage) of airborne radioactive material from areas and structures adjacent to the control room envelope,
- Radiation shine from the external radioactive plume released from the facility (i.e., external airborne cloud),
- Radiation shine from radioactive material in the reactor containment (i.e., containment shine dose; not applicable to a CRDA release occurring outside containment),

- Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters (i.e., CR filter shine dose).

Note: The external airborne cloud shine dose and CR filter shine dose due to a CRDA are insignificant compared to those due to a LOCA (see the core release fractions for LOCA and non-LOCA design basis accidents in Tables 1 and 3 of Reference 9.1). Therefore, these direct dose contributions are considered to be insignificant and are not evaluated for a CRDA.

4.14 The radioactive material releases and radiation levels used in the control room dose analysis are determined using the same source term, transport, and release assumptions used for determining the exclusion area boundary (EAB) and the low population zone (LPZ) TEDE values (Ref 9.1, Section 4.2.2).

4.15 The occupancy and breathing rate of the maximum exposed individual presents in the control room are incorporated in design input 5.3.3 (Ref. 9.1, Section 4.2.6).

4.16 10 CFR 50.67 (Ref 9.4) establishes the following radiological criterion for the control room.

CR Dose Acceptance Criterion: 5 Rem TEDE (50.67(b)(2)(iii))

4.17 Although allowed by Reference 9.1, Section 4.2.4, credit is not taken for the engineered safety features of the CR emergency filtration (CREF) system that mitigate airborne activity within the control room.

4.18 No credits for KI pills or respirators are taken (Ref. 9.1, Section 4.2.5).

5.0 DESIGN INPUTS:

5.1 General Considerations

5.1.1 Applicability of Prior Licensing Basis

The implementation of an AST is a significant change to the design basis of the facility and assumptions and design inputs used in the analyses. The characteristics of the ASTs and the revised TEDE dose calculation methodology may be incompatible with many of the analysis assumptions and methods currently used in the facility's design basis analyses. The HCGS plant specific design inputs and assumptions used in the current TID-14844 analyses were assessed for their validity to represent the as-built condition of the plant and evaluated for their compatibility to meet the AST and TEDE methodology. The analysis in this calculation ensures that analysis assumptions, design inputs, and methods are compatible with the ASTs and the TEDE criteria.

5.1.2 Credit for Engineered Safety Features

Credit is taken only for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The safety-related CR emergency filtration system is not credited for dose mitigation.

5.1.3 Assignment of Numeric Input Values

The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 (Ref. 9.4) are compatible to AST and TEDE dose criteria and selected with the objective of producing conservative radiological consequences. As a conservative alternative, the limiting value applicable to each portion of the analysis is used in the evaluation of that portion.

5.1.5 Meteorology Considerations

The control room atmospheric dispersion factors (χ/Qs) for the turbine building louver release point are developed (Ref. 9.5) using the NRC sponsored computer code ARCON96. The EAB and LPZ χ/Qs were

reconstituted using the HCGS plant specific meteorology and appropriate regulatory guidance. The off-site χ/Qs reconstituted in Reference 9.9 were accepted by the staff in previous licensing proceedings.

5.2 Accident-Specific Design Inputs/Assumptions

The design inputs/assumptions utilized in the EAB, LPZ, and CR habitability analyses are listed in the following sections. The design inputs are compatible with the AST and TEDE dose criteria and assumptions are consistent with those identified in Appendix C of RG 1.183 (Ref. 9.1). The design inputs and assumptions in the following sections represent the as-built design of the plant.

Design Input Parameter	Value Assigned	Reference
5.3 CRDA Parameters		
5.3.1 Source Term		
5.3.1.1 Proposed extended power uprate level	4,031 MW_t	Section 6.1
5.3.1.2 Isotopic Core Inventory In Ci/MW_t		9.3

Isotope	Activity	Isotope	Activity	Isotope	Activity
KR-83M	2.981E+03	I-134	5.937E+04	RB-86	1.300E+02
KR-85	4.711E+02	I-135	5.117E+04	RB-88	1.574E+04
KR-85M	5.908E+03	XE-131M	3.129E+02	CS-134	1.319E+04
KR-87	1.097E+04	XE-133	5.306E+04	CS-136	3.704E+03
KR-88	1.539E+04	XE-133M	1.743E+03	CS-137*	1.096E+04
I-131	2.779E+04	XE-135	1.482E+04		
I-132	3.991E+04	XE-135M	1.118E+04	* CS-137 inventory includes BA-137M inventory	
I-133	5.454E+04	XE-138	4.322E+04		

5.3.1.3 Radionuclide Composition		
Group	**Elements**	9.1, Section 3.4, Table 5
Noble gases	Xe, Kr	
Halogens	I, Br	
Alkali metals	Cs, Rb	
5.3.1.4 Number of fuel rods in fuel assembly	62	9.20
5.3.1.5 Damaged fuel rods: Breached Fuel Rods Melted Fuel Rods	 850 0.77% of the breached fuel rods	9.12, Section 6.2.2
5.3.1.6 Number of fuel assemblies in core	764	9.14
5.3.1.7 Fission products release from breached fuel rods to reactor coolant	10% noble gas in breached rods 10% iodine in breached rods 12% Alkali metal in breached rods	9.1, Appendix C, Section 1 9.1, Appendix C, Section 1 9.1, Section 3.2, Table 3
5.3.1.8 Fission products transfer from reactor coolant to turbine/ condenser	100% noble gas 10% iodine 1% Alkali metal	9.1, Appendix C, Section 3.3
5.3.1.9 Fission products available for release to the environment from turbine/ condenser	100% noble gas 10% iodine 1% Alkali metal	9.1, Appendix C, Section 3.4
5.3.1.10 Radial peaking factor	1.5 (1.75 conservatively assumed)	9.6, Appendix A, Section III.7

Design Input Parameter	Value Assigned	Reference
5.3.1.11 Fission products release from melted fuel rods to reactor coolant	100% noble gas in melted fuel 10% iodine in melted fuel 25% Alkali metal in melted fuel	9.1, Appendix C, Section 1 9.1, Appendix C, Section 1 Assumed based on 9.1, Table 1
5.3.2 Activity Transport in Turbine Building (see Figure 1)		
5.3.2.1 Condenser leak rate	1% per day	9.1, Appendix C, Section 3.4
5.3.2.2 Duration of turbine/condenser leak rate	24 hours	9.1, Table 6 and Appendix C, Section 3.4
5.3.2.3 Turbine/Condenser leak to the atmosphere	Ground level release	9.1, Appendix C, Section 3.4
5.3.2.4 Chemical form of Iodine in reactor coolant released within the pressure vessel		
Aerosol	95%	9.1, Appendix C, Section 3.6
Elemental	4.85%	
Organic	0.15%	
5.3.2.5 Chemical form of iodine available for release from turbine and main condenser		
Elemental	97%	9.1, Appendix C, Section 3.6
Organic	3%	
5.3.2.6 Dilution or holdup within the turbine building	Not credited	9.1, Appendix C, Section 3.4
5.3.2.7 Condenser free volume	235,000 ft^3	9.13, Page 3
5.3.3 Control Room Parameters (see Figure 2)		
5.3.3.1 CR volume	85,000 ft^3	9.10, page 10
5.3.3.2 CR normal air inflow rate during CRDA	3,000 ± 10% cfm for 0-720 hrs (conservatively modeled as 3,300 cfm)	9.18 and Assumption 4.17
5.3.3.3 CR occupancy factors		
Time (Hr)	**%**	9.1, Section 4.2.6
0-24	100	
24-96	60	
96-720	40	
5.3.3.4 CR breathing rate	3.5E-04 m^3/sec	9.1, Section 4.2.6
5.3.3.5 CR atmospheric dispersion factors for Turbine Building louvers release (X/Qs)		
Time (Hr)	**X/Q (sec/m^3)**	
0-2	6.17E-04	9.5, Section 8.3
2-8	4.00E-04	
8-24	1.44E-04	
24-96	1.00E-04	
96-720	7.49E-05	

Design Input Parameter	Value Assigned	Reference
5.3.4 Site Boundary Release Model Parameters		
5.3.4.1 EAB atmospheric dispersion factor (χ/Q)	1.9E-04 (sec/m^3)	9.9, Pages 5 & 9
5.3.4.2 LPZ Atmospheric dispersion factors (X/Qs)		
Time (Hr)	**X/Q (sec/m^3)**	
0-2	1.9E-05	9.9, Pages 5 & 9
2-4	1.2E-05	
4-8	8.0E-06	
8-24	4.0E-06	
24-96	1.7E-06	
96-720	4.7E-07	
5.3.4.3 EAB breathing rate	3.5E-04 m^3/sec	9.1, Section 4.1.3
5.3.4.4 LPZ breathing rates (m^3/sec)		
Time (Hr)	**(m^3/sec)**	
0-8	3.5E-04	9.1, Section 4.1.3
8-24	1.8E-04	
24-720	2.3E-04	

6.0 CALCULATIONS:

6.1 Extended Uprated Power Level

Original Licensed Power Level = 3,293 MW_t (Ref. 9.21)

Proposed Power Level Increase = 20%

Instrument Uncertainty = 2% (Ref. 9.22)

Extended Uprated Power Level = 3,293 $MW_t \times 1.20 \times 1.02 \approx 4{,}031\ MW_t$

6.2 Composite Percentage Release Fractions

This calculation uses the gap activity inventory fractions in Table 3 of RG 1.183 and assumes the release of 50% of the iodine and 100% of the noble gases for fuel reaching melted conditions (per RG 1.183, Appendix C, Section 1). Since the fuel gap can also contain the alkali metals (per RG 1.183 Table 1), this calculation applies a gap activity inventory fraction of 12% consistent with RG 1.183 Table 3. Since Appendix C of RG 1.183 does not address the melt release fraction for alkali metals for a CRDA, this calculation will assume 25% of the alkali metals are released from the melted fuel consistent with RG 1.183 Table 1. Although RG 1.183 Table 1 reports that a small fraction of other nuclide groups are also released from the melted fuel, these source terms are neglected in this calculation due to 1) a very small fraction of fuel exposed to melt condition (<1%), 2) the small in-vessel release fractions for these nuclide groups, and 3) the low volatility of these aerosols from both reactor coolant and condenser.

Group	Gap Release Fraction	Melt Release Fraction
Noble Gases	10%	100%
Iodine	10%	50%
Alkali Metals	12%	25%

Iodine Release Fraction = (1-0.0077)*10% + 0.0077*50% = 10.308% = 0.10308

NG Release Fraction = (1-0.0077)*10% + 0.0077*100% = 10.693% = 0.10693

Alkali Metals Release Fraction = (1-0.0077)*12% + 0.0077*25% = 12.100% = 0.12100

(These composite rod Iodine and NG release fractions are consistent with Reference 9.12,Section 6.2.2)

Total Number of Rods Per Core = 62 rods/assembly (Ref. 9.20) × 764 assemblies (Ref. 9.14) = 47368 rods/core

7.0 RESULTS SUMMARY:

7.1 The results of the CRDA analysis are summarized in the following table:

	Control Rod Drop Accident TEDE Dose (rem)		
	Receptor Location		
	Control Room	EAB	LPZ
Calculated Dose	1.37E-01	2.92E-02 occurs @ 0.0 hr	6.23E-03
Allowable TEDE Limit	**5.0 E+00**	**6.3E+00**	**6.3E+00**
	RADTRAD Computer Run No.		
	HEPU3300CRDA00	HEPU3300CRDA00	HEPU3300CRDA00

Significant assumptions used in this analysis:

- Radial peaking factor = 1.75
- All activity released to the environment at ground level through TB louvers
- CREF system is not credited.
- 850 fuel rods breached and 0.77% of the breached fuel rods have fuel melt
- Core thermal power = 4,031 MW_t

7.2 Compliance of proposed dose increases with the 10 CFR 50.59 rule is shown as follows:

Design Basis Accident	Current Licensing Basis Dose (rem)			Proposed Total Dose (rem) TEDE	Regulatory Dose Limit (rem) TEDE	Proposed Increase (rem) TEDE	Minimal Increase (rem) TEDE	RG Dose Limit (rem) TEDE
	Thyroid	Whole Body	Equivalent TEDE					
	A	B	C=A*0.03+B	D	E	F=D-C	G=0.1(E-C)	H
Control Rod Drop Accident (CRDA)	H-1-CG-MDC-1795, Rev 3			H-1-CG-MDC-1795, Rev 4				
Control Room	0.657	0.0123	0.03201	0.137	5.00	0.105	0.50	5.00
Exclusion Area Boundary	0.35	0.35	0.3605	0.0292	25.00	-0.331	2.46	6.30
Low Population Zone	Not Calculated			0.00623	25.00			6.30

E From 10 CFR 50.67 (Ref. 9.25)

H From RG 1.183, Table 6 (Ref. 9.1)

The current licensing basis (CLB) EAB equivalent TEDE dose is considerably higher than the revised AST analysis TEDE dose because the CLB EAB doses reported in the UFSAR are for a scenario without MSIV closure and consequently with greater iodine and noble gas releases via the Gaseous Waste Management System (GWMS). The CLB release scenario only considers noble gas hold-up times as a dose reduction mechanism. The CLB release scenario is not considered in the AST CRDA event, which models the condenser release path described in Regulatory Guide 1.183 Appendix C (Ref. 9.1).

8.0 CONCLUSIONS:

The analysis results presented in Section 7.1 indicate that the EAB, LPZ, and CR doses due to a control rod drop accident are within their allowable TEDE limits. The results indicate that CREF system initiation is not required during a CRDA.

The comparisons in Section 7.2 document a decrease in the proposed EAB dose; the EAB dose decrease is due to the lower proposed iodine activity release. The comparisons in Section 7.2 confirm that the proposed increase in the CR dose is less than the minimal dose increase regulatory limit, and that the total calculated EAB and CR doses are less than the allowable regulatory guide limits. Therefore, pursuant to 10 CFR 50.59 guidance as defined in References 9.23 and 9.24, the proposed increase in the core thermal power level and resulting post-CRDA doses can be adopted as current design and licensing bases for the HCGS.

9.0 REFERENCES:

1. U.S. NRC Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000
2. S.L. Humphreys et al., "RADTRAD 3.02: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," NUREG/CR-6604, USNRC, April 1998
3. Vendor Technical Document (VTD) No. 430058, Volume 002, Rev 1, EPU TR T0802, Radioactive Source Term – Core Inventory
4. 10 CFR 50.67, "Accident Source Term."
5. Calculation No. H-1-ZZ-MDC-1879, Rev 1, Control Room & Technical Support Center χ/Qs Using ARCON96 Code
6. NUREG-0800, Standard Review Plan 15.4.9 Appendix A, Revision 2, "Radiological Consequences of Control Rod Drop Accident (BWR)," July 1981
7. Federal Guidance Report 11, EPA-520/1-88-020, Environmental Protection Agency
8. Federal Guidance Report 12, EPA-402-R-93-081, Environmental Protection Agency
9. Calculation No. H-1-ZZ-MDC-1820, Rev 0, Offsite Atmospheric Dispersion Factors
10. Calculation No. H-1-ZZ-MDC-1882, Rev 0, Control Room Envelope Volume
11. HCGS Procedure No. HC.OP-SO.CG-0001(R), Rev 32, Condenser Air Removal System Operation
12. GE Report NEDO 31400A, October 1992, "Safety Evaluation for Eliminating The Boiling Water Reactor Main Steam Isolation Valve Closure Function and Scram Function of The Main Steam Line Radiation Monitor."

13. HCGS Procedure No. HC.OP-AB.RPV-0008(Q), Rev 0, Reactor Coolant Activity
14. Hope Creek Technical Specification 5.3, Reactor Core – Fuel Assemblies
15. Critical Software Package Identification No. A-0-ZZ-MCS-0225, Rev 2, RADTRAD Computer Code.
16. HCGS General Arrangement Drawings:
    a. P-0007-0, Rev 7, Plan EL 171'-0" & EL 201'-0"
    b. P-0011-0, Rev 5, Sections C-C & D-D
17. HCGS Architectural Drawing No. A-0221-0, Sheet 1, Rev. 10, General Plant Roof Plan
18. HCGS Mechanical P&ID No. M-78-1, Rev 21, Aux Bldg Control Area Air Flow Diagram.
19. HCGS Technical Specification 3/4.3.10, Mechanical Vacuum Pump Trip Instrumentation.
20. Nuclear Fuel Section Design Input File, T03.5-043, Revised Refueling Accident (Bundle Drop) Analysis
21. NRC Safety Evaluation Report NUREG-1048, October 1984, Operation of Hope Creek Generating Station
22. U.S. NRC Regulatory Guide 1.49, Rev 1, Power Levels of Nuclear Power Plants
23. PSEG Procedure No. NC.NA-AS.ZZ-0059(Q), Rev 11, 10CFR50.59 Program Guidance.
24. Nuclear Energy Institute Report No. NEI 96-07, Rev 1, Guidelines for 10 CFR 50.59 Implementation.
25. 10 CFR 50.67, "Accident Source Term."
26. NRC Safety Evaluation Report, Hope Creek Generating Station – Issuance of Amendment No. 134 for Increase in Allowable MSIV Leakage Rate and Elimination of MSIV Sealing System.

10.0 TABLES:

Table 1
CRDA Activity In Peak Failed Fuel

Isotope	Core Inventory (Ci/MWt) A	Uprated Core Thermal Power Level (MWt) B	Radial Peaking Factor C	Total Number of Fuel Rod In Core D	Number of Fuel Rod Damaged E	Post-CRDA Activity In Damaged Fuel (Ci) F=(A*B*C*E)/D
I-131	2.779E+04	4031	1.75	47368	850	3.518E+06
I-132	3.991E+04	4031	1.75	47368	850	5.052E+06
I-133	5.454E+04	4031	1.75	47368	850	6.904E+06
I-134	5.937E+04	4031	1.75	47368	850	7.515E+06
I-135	5.117E+04	4031	1.75	47368	850	6.477E+06
KR-83M	2.981E+03	4031	1.75	47368	850	3.774E+05
KR- 85	4.711E+02	4031	1.75	47368	850	5.963E+04
KR- 85M	5.908E+03	4031	1.75	47368	850	7.479E+05
KR- 87	1.097E+04	4031	1.75	47368	850	1.389E+06
KR-88	1.539E+04	4031	1.75	47368	850	1.948E+06
XE-131M	3.129E+02	4031	1.75	47368	850	3.961E+04
XE-133	5.306E+04	4031	1.75	47368	850	6.717E+06
XE-133M	1.743E+03	4031	1.75	47368	850	2.206E+05
XE-135	1.482E+04	4031	1.75	47368	850	1.876E+06
XE-135M	1.118E+04	4031	1.75	47368	850	1.415E+06
XE-138	4.322E+04	4031	1.75	47368	850	5.471E+06
RB-86	1.300E+02	4031	1.75	47368	850	1.646E+04
RB-88	1.574E+04	4031	1.75	47368	850	1.992E+06
CS-134	1.319E+04	4031	1.75	47368	850	1.670E+06
CS-136	3.704E+03	4031	1.75	47368	850	4.689E+05
CS-137*	1.096E+04	4031	1.75	47368	850	1.387E+06
CS-138	4.840E+04	4031	1.75	47368	850	6.127E+06

A From Reference 9.3

* CS-137 inventory includes BA-137M inventory

Table 2
Post-CRDA Activity Released From Condenser

Isotope	Post-CRDA Activity In Damaged Fuel (Ci) A*	Activity Release Fraction From Fuel B	Activity Release Fraction To Condenser C	Activity Release Fraction From Condenser D	Activity Available For Release From Condenser (Ci) E=A*B*C*D
I-131	3.518E+06	0.10308	0.10	0.10	.3626E+04
I-132	5.052E+06	0.10308	0.10	0.10	.5208E+04
I-133	6.904E+06	0.10308	0.10	0.10	.7117E+04
I-134	7.515E+06	0.10308	0.10	0.10	.7747E+04
I-135	6.477E+06	0.10308	0.10	0.10	.6677E+04
KR-83M	3.774E+05	0.10693	1.00	1.00	.4035E+05
KR- 85	5.963E+04	0.10693	1.00	1.00	.6377E+04
KR- 85M	7.479E+05	0.10693	1.00	1.00	.7997E+05
KR- 87	1.389E+06	0.10693	1.00	1.00	.1485E+06
KR-88	1.948E+06	0.10693	1.00	1.00	.2083E+06
XE-131M	3.961E+04	0.10693	1.00	1.00	.4235E+04
XE-133	6.717E+06	0.10693	1.00	1.00	.7182E+06
XE-133M	2.206E+05	0.10693	1.00	1.00	.2359E+05
XE-135	1.876E+06	0.10693	1.00	1.00	.2006E+06
XE-135M	1.415E+06	0.10693	1.00	1.00	.1513E+06
XE-138	5.471E+06	0.10693	1.00	1.00	.5850E+06
RB-86	1.646E+04	0.12100	0.01	0.01	.1991E+00
RB-88	1.992E+06	0.12100	0.01	0.01	.2411E+02
CS-134	1.670E+06	0.12100	0.01	0.01	.2020E+02
CS-136	4.689E+05	0.12100	0.01	0.01	.5673E+01
CS-137	1.387E+06	0.12100	0.01	0.01	.1679E+02
CS-138	6.127E+06	0.12100	0.01	0.01	.7413E+02

A From Table 1
B From Section 6.2

11.0 FIGURES:

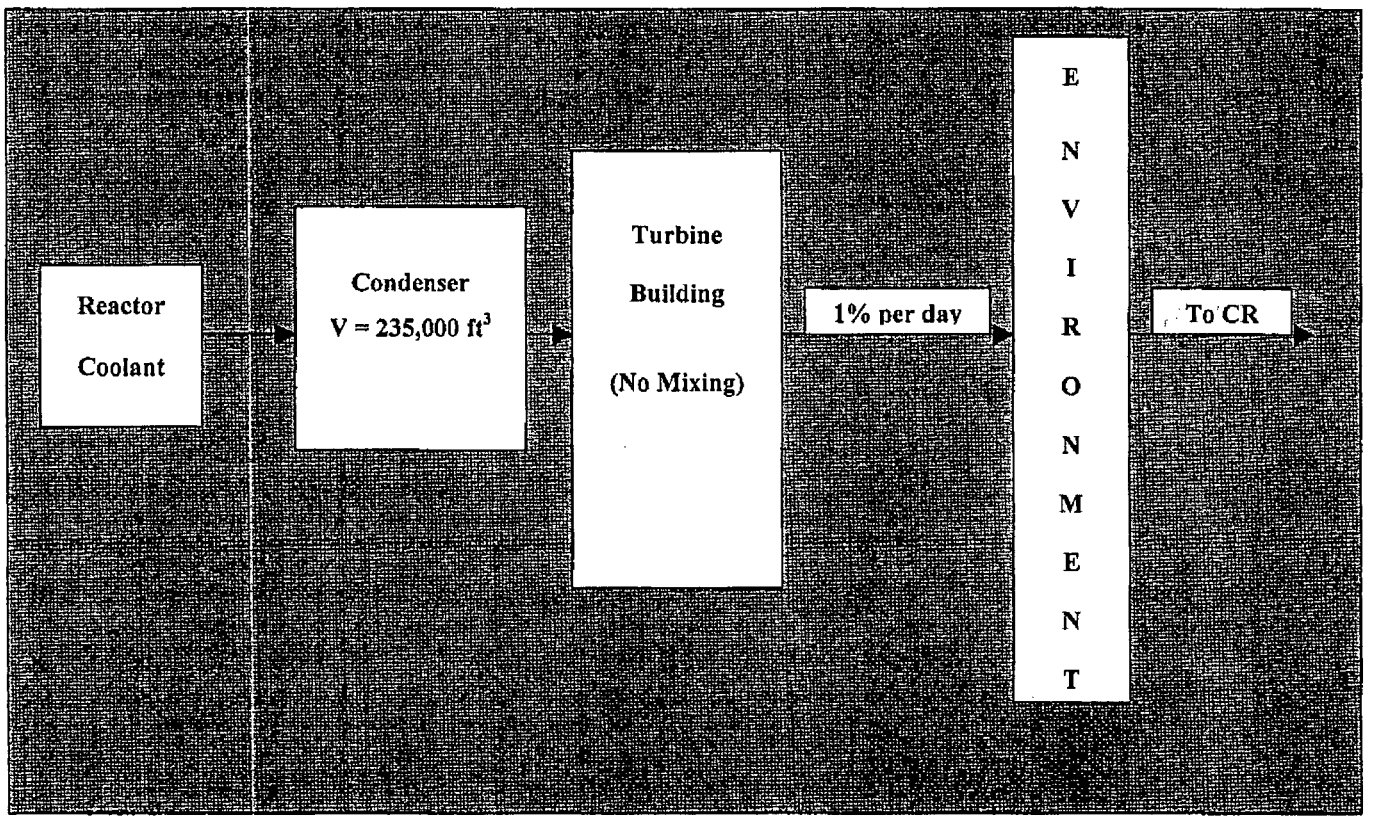

Figure 1: RADTRAD Nodalization For CRDA Release

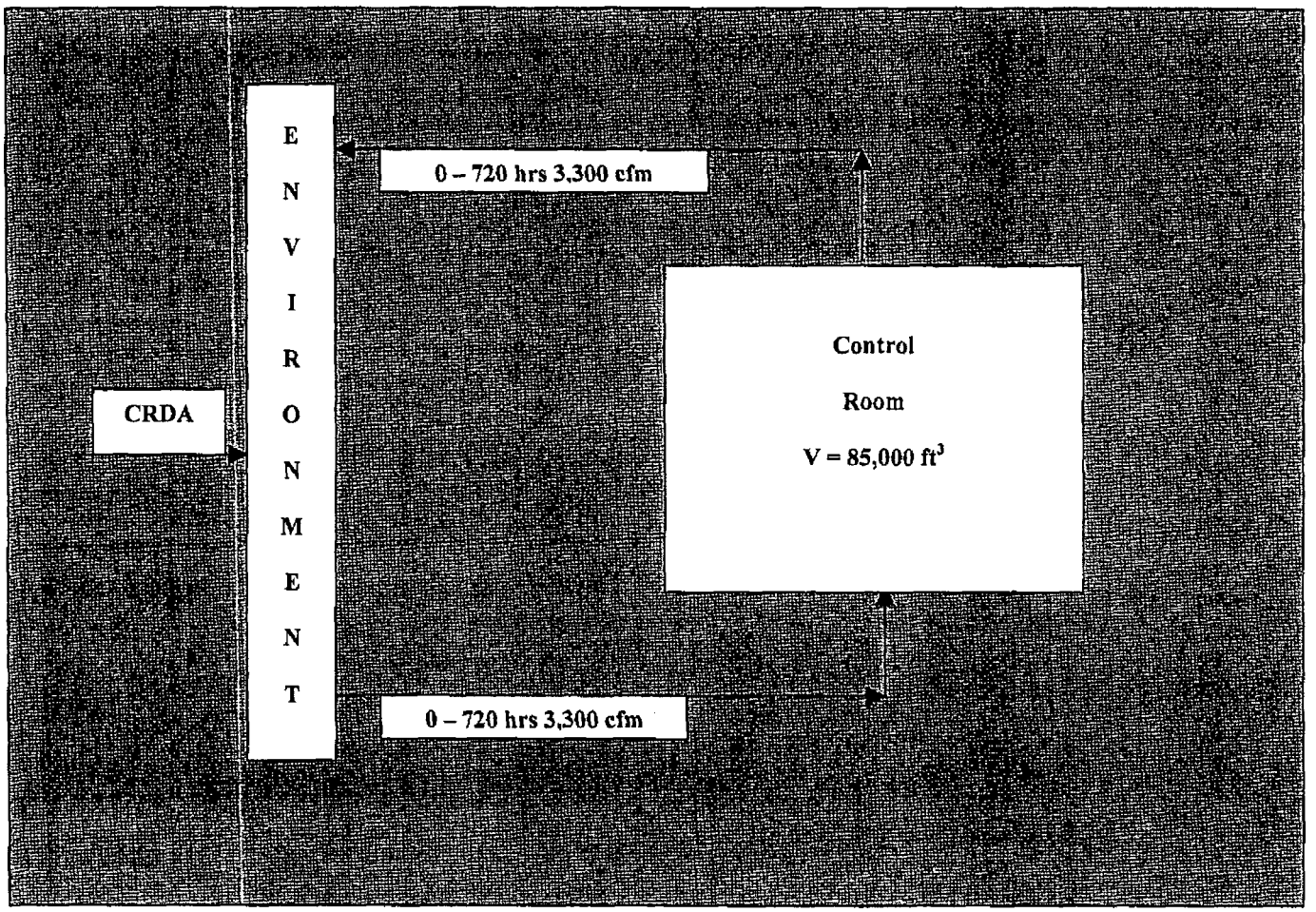

Figure 2 – HCGS Control Room RADTRAD Nodalization

12.0 AFFECTED DOCUMENTS:

The following documents will be either superseded or revised:

Document to be superseded

Calculation H-1-CG-MDC-1795, Rev 4

Documents to be revised:

UFSAR Section 15.4.9

UFSAR Table 15.4-6

13.0 ATTACHMENTS:

13.1 – 1 CD with the following electronic files:

Calculation No: H-1-CG-MDC-1795, Rev 5.
Comment Resolution Form 2 – Mark Drucker
Owner's Acceptance Comment Resolution Form 2 – Michael E. Crawford
Certification for Design Verification Form-1
RCPD Form-1

1 CD With Various Electronic Files

FORM-1

CERTIFICATION FOR DESIGN VERIFICATION

Reference No. H-1-CG-MDC-1795, Rev. 4

SUMMARY STATEMENT

Design verification consisted of a detailed check of the completed engineering evaluation. The method of verification included design review and "line-by-line" examination.

Use of a generic design verification checklist is waived. Design input considerations and assumptions are adequately identified in the body of the design calculation.

The design calculation completely revised existing design calculation H-1-CG-MDC-1795, Rev 3 and assessed the offsite and control room doses due to a postulated control rod drop accident (CRDA) using Alternative Source Term (AST), the guidance in Regulatory Guide 1.183, Appendix C, and TEDE dose criteria. All doses are within guideline values.

Each individual named below in the right column hereby certifies that the design verification for the subject document or document portion has been completed, the questions from the generic checklist have been reviewed and addressed as appropriate, and all comments have been adequately incorporated. The top right column individual is the Lead Design Verifier. SAP Order/Operation final confirmations are the legal equivalent of signatures.

	Mark Drucker 5/15/2006
Design Verifier Assigned By (print name of Supv/Manager/Director)*	Name of **Lead** Design Verifier* / Date
Design Verifier Assigned By (print name of Supv/Manager/Director)*	Name of Design Verifier* / Date
Design Verifier Assigned By (print name of Supv/Manager/Director)*	Name of Design Verifier* / Date
Design Verifier Assigned By (print name of Supv/Manager/Director)*	Name of Design Verifier* / Date

*If the Manager/Supervisor acts as the Design Verifier, the name of the next higher level of technical management is required in the left column.

FORM-2

COMMENT / RESOLUTION FORM FOR DESIGN DOCUMENT REVIEW/CHECKING OR DESIGN VERIFICATION

REFERENCE DOCUMENT NO. /REV. H-1-CG-MDC-1795, Revision 4

COMMENTS	RESOLUTION	
1. General – Editorial Comments are being provided separately in the form of a redline/strikeout mark-up. It is the discretion of the Originator to determine which editorial comments should be incorporated.	Incorporated	MD 5-15-06
2. I found the Design Input 5.3.3.2 normal control room HVAC intake flow of 3000 cfm on the air flow diagram. But I did not find a notation that it is valid only to +/- 10 percent. Can you find a reference for this uncertainty? If not, the calculation should say that is based on engineering judgment.	The flow variation of ±10% is conservatively assumed.	MD 5-15-06
3. Do you have explicit direction from HCGS that this calculation is not to consider unfiltered inleakage during normal control room HVAC operation? If so, you should refer to this direction. If not, you might consider adding the 900 cfm (??) that was used in the LOCA AST dose analysis to the 3300 cfm currently being considered.	The air intake flow rate of 3,300 cfm will pressurize the CR envelop at a relative higher pressure than it would be in the pressurized condition during a LOCA with makeup air intake flow rate of 1,000 cfm. Therefore, the additional inleakage makes the analysis unnecessary conservative. The use of unfiltered normal flow rate is sufficiently conservative.	MD 5-15-06
4. Reference 9.19 is licensing change request LCR H01-03. This LCR was submitted to the NRC Staff for review/approval in January 2002, with a proposed December 2002 implementation date. If this LCR has been approved, add its SER as a reference, and refer to the SER in the Section 2.0 Background text.	SER is not issued yet	MD 5-15-06

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FORM-2

COMMENT / RESOLUTION FORM FOR DESIGN DOCUMENT REVIEW/CHECKING OR DESIGN VERIFICATION

5. Per Section 5.3.1.2 the I-135 core inventory is 6.235e4 Ci/MWt. This is actually the combined I-135 plus Xe-135m core inventory. Using the combined inventory is appropriate if Xe-135m is not explicitly considered in the RADTRAD analysis (as would occur when using the default RADTRAD isotope release profile). However, Section 3.0 of this calculation added Xe-135m to the RADTRAD analysis. Therefore, this calculation conservatively double counts the Xe-135m activity. I suggest removing the Xe-135m contribution from the I-135 core inventory (i.e., revise Table 1 and rerun RADTRAD). Alternatively, since the dose impact is conservative and most likely negligible, Section 5.3.1.2 and/or Table 1 should be footnoted to document that the listed I-135 core inventory is actually the combined I-135 plus Xe-135m core inventory.	Incorporated	MD 5-15-06
6. Per Section 5.3.1.2 the Cs-137 core inventory is 1.096E+04 Ci/MWt. This is actually the combined Cs-137 plus Ba-137m core inventory. Using the combined inventory is appropriate since Ba-137m is not explicitly considered in the RADTRAD analysis. Section 5.3.1.2 and/or Table 1 should be footnoted to document that the listed Cs-137 core inventory is actually the combined Cs-137 plus Ba-137m core inventory.	A footnote is added to Table 1	MD 5-15-06
7. In Table 2: -- Column A Title should list units of "(Ci)" -- Column B Title should not list the units of curies; the data are unitless release fractions. -- Column C Title should not list the units of curies; the data are unitless release fractions. -- Column D Title should not list the units of curies; the data are unitless release fractions.	Incorporated	MD 5-15-06
8. EDITORIAL: In dose conversion factor (DCF) File HCRDA_FG11&12.txt the first line states "added 5 nuclides". In actuality, seven (7) nuclides have been added. If you should ever need to rerun the CRDA case, consider revising "5" to "7" in the first line of the DCF file.	Incorporated	MD 5-15-06

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FORM-2

COMMENT / RESOLUTION FORM FOR DESIGN DOCUMENT REVIEW/CHECKING OR DESIGN VERIFICATION

9. Dose conversion factor (DCF) File HCRDA_FG11&12.txt lists for Rb-88 an ingestion "effective" committed dose equivalent per unit intake value of 4.700E-11 Sv/Bq. Per FGR-11 page 160, the Rb-88 ingestion "effective" committed dose equivalent per unit intake value is actually 4.710E-11 Sv/Bq. This is a minor error that will have absolutely no impact on the analysis results. If you should ever need to rerun the CRDA case, revise this DCF entry.	Incorporated	MD 5-15-06
10. EDITORIAL: Input File HEPU3300CRDA00.psf spells "condenser" incorrectly in the Pathway 1 description. This is a minor error that will have absolutely no impact on the analysis results. If you should ever need to rerun the CRDA case, revise this pathway description.	Incorporated	MD 5-15-06
END		
Mark Drucker (MD) Mark Drucker 05/09/2006 SUBMITTED BY DATE	Gopal J. Patel 05/10/2006 RESOLVED BY DATE	Acceptance of Resolution

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FORM-2
COMMENT / RESOLUTION FORM
FOR DESIGN DOCUMENT
REVIEW/CHECKING OR DESIGN VERIFICATION
(SAP Standard Text Key "NR/CDV2")

REFERENCE DOCUMENT NO. /REV. H-1-CG-MDC-1795 / Rev. 4

Control Rod Drop Accident Radiological Consequences

COMMENTS

1. Calculation continuation sheets wrong revision—everything in the calc should have Rev. 12 on the bottom
2. Coversheet: HOPE CREEK ONLY – Q box should be checked
3. Section 5.1.1: is reference to TID-14844 correct for AST? Aren't the Iodine DCFs from ICRP 30?
4. Reference 9.2: should include reference to V3.02 of Radtrad
5. Reference 9.3: add DCRMS number (VTD 430058, Sheet 2, Rev. 1)
6. Reference 9.1 is still Rev. 0 in DCRMS
7. Reference 9.11 is Rev. 32
8. References 9.13 (on Sheets 5, 14, of 24), 9.14 (on Sheet 16 of 24), and 9.15 (on Sheet 6 of 24) are missing in Reference Section.
9. Table 1: spot check of Reference values—cannot find CS-138 core inventory value listed in Reference 9.3
10. Comment / Resolution Form-2: wrong Revision # 4IR0 verses 4? Wrong procedure. IDV under CC-AP.10 and not DE-AP.10.
11. RCPD: Form-1: wrong Revision. Now Rev. 11 Changes item 7 only. Need new dates for signatures.
12. Background Section 2.0 of calculation states that LCR H01-03 creates MVP trip that will ensure CRDA release will be through condenser. The LCR has been approved in NRC license amendment 143. Should reference T/S 3/4.3.10 rather than LCR H01-03. Also see Section 12.0
13. Section 13.0 Attachments: why does the disk have Calc H-1-ZZ-MDC-1930, Rev. 0? That is listed in DCRMS as the FHA Analysis for EPU, but its status is listed as Preliminary.

RESOLUTION

1. Incorporated.
2. Incorporated.
3. TID establishes source term basis pre-AST.
4. Incorporated.
5. Incorporated.
6. Rev 1 will in DCRM before this calc is issued.
7. Incorporated.
8. It looks like that the reference page was missing, which is included.
9. Cs-138 is deleted. Cesium remains water borne, therefore not included in the analysis.
10. Incorporated.
11. Incorporated.
12. Incorporated.

FORM-2
COMMENT / RESOLUTION FORM
FOR DESIGN DOCUMENT
REVIEW/CHECKING OR DESIGN VERIFICATION
(SAP Standard Text Key "NR/CDV2")

ACCEPTANCE OF RESOLUTION	
1-12: ALL COMMENTS HAVE BEEN SATISFACTORILY INCORPORATED & RESOLVED. MEC 6/20/2006	
Michael E. Crawford 05/06/06 SUBMITTED BY DATE	Gopal J. Patel 05/10/06 RESOLVED BY DATE

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FORM-1
REGULATORY CHANGE PROCESS DETERMINATION

Document I.D.: H-1-CG-MDC-1795 **Revision:** 4

Title: Control Rod Drop Accident Radiological Consequences

Activity Description:

Issuing the design calculation, which determines the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room (CR) doses due to a Control Rod Drop Accident (CRDA) using the Alternative Source Term (AST) and core thermal power level of 4,031 MW_t, including the instrument uncertainty.

Note that more than one process may apply. If unsure of any answer, contact the cognizant department for guidance.

Activities Affected	Yes	No	Action
1. Does the proposed activity involve a change to the Technical Specifications or the Operating License?	☐	☒	If Yes, contact Licensing. **See NOTE in Section 4.1.1 LCR No.** _____
2. Does the proposed activity involve a change to the Quality Assurance Plan? Example: • Changes to Chapter 17.2 of UFSAR	☐	☒	If Yes, contact Quality Assessment.
3. Does the proposed activity involve a change to the Security Plan? Examples: • Change program in NC.NA-AP.ZZ-0033(Q) • Change indoor/outdoor security lighting • Placement of component or structure (permanent or temporary) within 20 feet of perimeter fence • Obstruct field of view from any manned post • Interfere with security monitoring device capability • Change access to any protected or vital area • Modify safeguards systems or equipment	☐	☒	If Yes, contact Security Department.
4. Does the proposed activity involve a change to the Emergency Plan? Examples: • Change ODCM/accident source term • Change liquid or gaseous effluent release path • Affect radiation monitoring instrumentation or EOP/AOP setpoints used in classifying accident severity • Affect emergency response facilities or personnel, including control room • Affect communications, computers, information systems or Met tower	☐	☒	If Yes, contact Emergency Preparedness

FORM-1
REGULATORY CHANGE PROCESS DETERMINATION

Document I.D.: H-1-CG-MDC-1795 **Revision:** 4

Title: Control Rod Drop Accident Radiological Consequences

Activities Affected	Yes	No	Action
6. Does the proposed activity involve a change to the IST Program Plan? Example: • Affect the design or operating parameters of a Nuclear Class 1, 2, or 3 Pump or Valve (Guidance in NC.CC-AP.ZZ-0007(Q))	☐	☒	If Yes, contact Engineering Programs ISI/IST.
7. Does the proposed activity involve a change to the Fire Protection Program? Examples: • Change program in NC.DE-PS.ZZ-0001(Q) • Change combustible loading of safety related space • Change or affect fire detection system • Change or affect fire suppression system/component • Change fire doors, dampers, penetration seal or barriers • See NC.CC-AP.ZZ-0007 for details • Change or affect FPP compensatory measures	☐	☒	If Yes, contact Design Engineering.
8. Does the proposed activity involve Maintenance which restores SSCs to their original design and configuration? Examples: • CM or PM activity • Implements an approved Design Change? • Troubleshooting (which does not require 50.59 screen per SH.MD-AP.ZZ-0002)	☐	☒	If Yes, process in accordance with NC.WM-AP.ZZ-0001(Q)
9. Is the proposed activity a temporary change (T-Mod) which ***meets all the following conditions***? • Directly supports maintenance and is NOT a compensatory measure to ensure SSC operability. • Will be in effect at power operation **less than 90 days.** • Plant will be restored to design configuration upon completion. • SSCs will NOT be operated in a manner that could impact the function or operability of a safety related or Important-to-Safety system.	☐	☒	If Yes, contact Engineering.
10. Does the proposed activity consist of changes to maintenance procedures which do NOT affect SSC design, performance, operation or control? **Note:** Procedure information affecting SSC design, performance, operation or control, including Tech Spec required surveillance and inspection, ***requires 50.59 screening***. Examples include acceptance criteria for valve stroke times or other SSC function, torque values, and types of materials (e.g., gaskets, elastomers, lubricants, etc.)	☐	☒	If Yes, process in accordance with NC.NA-AP.ZZ-0001(Q)

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Activities Affected	Yes	No	Action
11. Does the proposed activity involve a *minor* UFSAR change (including documents incorporated by reference)? Examples: • Reformatting, simplification or clarifications that do not change the meaning or substance of information • Removes obsolete or redundant information or excessive detail • Corrects inconsistencies within the UFSAR • Minor correction of drawings (such as mislabeled ID)	☐	☒	If Yes, process in accordance with NC.NA-AP.ZZ-0035(Q)
12. Does the proposed activity involve a change to an Administrative Procedure (NAP, SAP or DAP) governing the conduct of station operations? Examples: • Organization changes/position titles • Work control/ modification processes	☐	☒	If Yes, process in accordance with NC.NA-AP.ZZ-0001(Q) and NC.DM-AP.ZZ-0001(Q)
13. Does the proposed activity involve a change to a regulatory commitment?	☐	☒	If Yes, contact Licensing.
14. Does the activity impact other programs controlled by regulations, operating license or Tech Spec? Examples: • Chemical Controls Program • NJ "Right-to-know" regulations • OSHA regulations • NJPDES Permit conditions • State and/or local building, electrical, plumbing, storm water management or "other" codes and standards • 10CFR20 occupational exposure	☐	☒	If Yes, process in accordance with applicable procedures such as: NC.NA-AP.ZZ-0038(Q) NC.LR-AP.ZZ-0037(Q)
15. Does the proposed activity affect the **Independent Spent Fuel Storage Installation (ISFSI)** or the **Dry Cask Storage System (DCSS)** or their analyses? Examples: • Affect the spent fuel canisters or casks • Affect the method of lifting, rigging or transporting DCSS • Challenge Spent Fuel Pool level limits or reactivity limits • Affect fire hazard analyses for the Heavy Haul Path • Affect procedures for DCSS operation or ISFSI activities	☐	☒	If Yes, contact Licensing and initiate the 10CFR72.48 screening process per NC.NA-AS.ZZ-0041 (NAS-41).
16. Has the activity already received a **10CFR50.59 Screen or Evaluation** under another process? Examples: • Calculation • Design Change Package or OWD change • Procedure for a Test or Experiment • DR/Nonconformance • Incorporation of previously approved UFSAR change	☒	☐	Take credit for **10CFR50.59 Screen or Evaluation** already performed. ID: H-1-ZZ-MDC-1880, Rev 2
17. Is the proposed change a change to a Chemistry procedure as described in paragraph 4.1.7?	☐	☒	If YES, no 50.59 Screen is required.

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If any other program or regulation *may be* affected by the proposed activity, contact the department indicated for **further** review in accordance with the governing procedure. If responsible department determines their program is not affected, attach a written explanation.

If ALL of the answers on the previous pages are "No," then check A below:

A. [ ] None of the activity is controlled by any of the processes above, therefore a 10CFR50.59 review **IS** required. Complete a 10CFR50.59 screen.

If one or more of the answers on the previous pages are "Yes," then check either B or C below as appropriate and explain the regulatory processes which govern the change:

B. [X] All aspects of the activity are controlled by one or more of the processes above, therefore a 10CFR50.59 review **IS NOT** required.

C. [ ] Only part of the activity is controlled by the processes above, therefore a 10CFR50.59 review **IS** required. Complete a 50.59 screen.

Explanation:

PREPARER (SIGN)	05/12/2006 DATE	Gopal J. Patel NAME (PRINT)	04/04/2007 QUAL EXPIRES
REVIEWER (SIGN)	06/09/2006 DATE	Michael E. Crawford NAME (PRINT)	12/13/07 QUAL EXPIRES