Calculation No. H-1-ZZ-MDC-1929, Revision 0
Fuel Handling Accident Radiological Consequences

(NC.DE-AP.ZZ-0002(Q), Rev. 12, Form 1)	CALCULATION COVER SHEET	Page 1 of	24
CALCULATION NUMBER:	H-1-ZZ-MDC-1929	REVISION:	0
TITLE:	Fuel Handling Accident Radiological Consequences		

#SHTS (CALC):	24	#ATT/#SHTS:	1/1	#IDV/50.59/72.48 SHTS:	5/4/0	#TOTAL SHTS:	34

CHECK ONE:

☒ FINAL ☐ INTERIM (Proposed Plant Change) ☐ VOID

☐ FINAL (Future Confirmation Req'd, enter tracking Notification number:)

SALEM OR HOPE CREEK: ☐ Q – LIST ☒ IMPORTANT TO SAFETY ☐ NON-SAFETY RELATED

HOPE CREEK ONLY: ☒Q ☐Qs ☐Qsh ☐F ☐R

ISFSI: ☐ IMPORTANT TO SAFETY ☐ NOT IMPORTANT TO SAFETY

☐ ARE STATION PROCEDURES IMPACTED? YES ☐ NO ☒

IF "YES", INTERFACE WITH THE SYSTEM ENGINEER & PROCEDURE SPONSOR. ALL IMPACTED PROCEDURES SHOULD BE IDENTIFIED IN A SECTION IN THE CALCULATION BODY [CRCA 70038194-0280]. INCLUDE AN SAP OPERATION FOR UPDATE AND LIST THE SAP ORDERS HERE AND WITHIN THE BODY OF THIS CALCULATION.

☐ CP and ADs INCORPORATED (IF ANY):

DESCRIPTION OF CALCULATION REVISION (IF APPL.):

See Revision ~~1~~ 0 History on next page. GBP 8/2/06

PURPOSE:

The purpose this calculation is determine the doses at Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room (CR) due to a Fuel Handling Accident (FHA) occurring in the reactor building (RB) with equipment hatch C-9 and the truck bay door open and the CR Emergency Filtration (CREF) system remains in a normal mode operation for the entire duration of accident. The FHA analysis is performed using the Alternative Source term (AST), the guidance provided in the Regulatory Guide 1.183, Appendix B, and the TEDE dose criteria. The core thermal power level is assumed to be at 4,031 MW_t including the instrument uncertainty.

CONCLUSIONS:

The analysis results presented in Section 7.1 indicate that the EAB, LPZ, and CR doses are within their allowable limits for a FHA occurring in the reactor building without secondary containment integrity (that is, with the containment hatch C-9 and RB truck bay door open). The results demonstrate that HCGS Technical Specification 3.6.5.1 applicability can be removed when irradiated fuel is being handled in the secondary containment and during core alterations.

The comparisons in Section 7.2 confirm that the proposed increases in the EAB, LPZ, & CR doses are less than the minimal dose increase regulatory limits, and that the proposed total doses are less than the allowable regulatory limits. Therefore, pursuant to 10 CFR 50.59 guidance as defined in References 9.25 and 9.26, the proposed increase in the core thermal power level and resulting post-FHA dose consequences can be adopted as current licensing and design bases for the HCGS.

	Printed Name / Signature	Date
ORIGINATOR/COMPANY NAME:	Gopal J. Patel/NUCORE	05/10/06
REVIEWER/COMPANY NAME:	N/A	N/A
VERIFIER/COMPANY NAME:	Mark Drucker/NUCORE	05/15/06
CONTRACTOR SUPERVISOR (If applicable)	N/A	
PSEG SUPERVISOR APPROVAL: (Always required)	Emin B. Ortalan/PSEG	8/7/06

REVISION HISTORY

Revision	Issue Date	Revision Description
0		Original Issue. 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, LPZ, 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.

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1	0	15	0
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1.0 PURPOSE:

The purpose this calculation is determine the doses at Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room (CR) due to a Fuel Handling Accident (FHA) occurring in the reactor building (RB) with equipment hatch C-9 and the truck bay door open and the CR Emergency Filtration (CREF) system remains in a normal mode operation for the entire duration of accident. The FHA analysis is performed using the Alternative Source term (AST), the guidance provided in the Regulatory Guide 1.183, Appendix B, and the TEDE dose criteria. The core thermal power level is assumed to be at 4,031 MW_t including the instrument uncertainty.

This analysis provides a basis for removing HCGS Technical Specification 3.6.5.1 applicability when irradiated fuel is being handled in the secondary containment and during core alterations and operations with a potential for draining the reactor vessel.

2.0 BACKGROUND:

PSE&G Nuclear proposed that the requirement for maintaining secondary containment per Technical Specification Surveillance Requirement 4.6.5.1 (Ref. 10.6.1) be relaxed. Therefore, an FHA is postulated such that radioactive material is released to the environment at the ground level through an open hatch C-9 and a truck bay door to demonstrate that the secondary containment integrity is not required during a FHA.

The following technical specification requirements are addressed in the FHA analysis:

<u>TS 3/4.9.8 WATER LEVEL – REACTOR VESSEL</u>

At least 22 feet 2 inches of water shall be maintained over the top of the reactor pressure vessel flange (Ref. 10.6.1).

<u>TS 3/4.9.9 WATER LEVEL – SPENT FUEL STORAGE POOL</u>

At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks (Ref. 10.6.2).

<u>TS 5.2.3 - SECONDARY CONTIANMENT</u>

The secondary containment consists of the Reactor Building, and a portion of the main steam tunnel and has a free volume of 4,000,000 cubic feet (Ref. 10.6.3).

<u>TS 5.3.1 – REACTOR CORE</u>

The reactor core shall contain 764 fuel assemblies and shall be limited to those assemblies that have been approved for use in BWRs.

3.0 ANALYTICAL APPROACH:

This analysis uses Version 3.02 of the RADTRAD computer code to calculate the potential radiological consequences of the FHA. 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 (Reference 9.2). The RADTRAD code is maintained as Software ID Number A-0-ZZ-MCS-0225 (Reference 9.18).

The EPU core inventory is obtained from Reference 9.3, and is calculated based on a thermal power level of 4,031 MW_t including a 2% instrument uncertainty (Section 6.1). The Ci/MW_t in the RADTRAD3.02 default nuclide inventory file Bwr_def_NIF is dependent on the plant-specific core thermal power level, reload design, fuel burnup, and fuel cycle. Therefore, the plant-specific isotopic Ci/MW_t information is developed in Table 1. The total number of fuel rods in the core is calculated by the product of rods per assembly (Ref. 9.11) and assemblies per core (Ref. 9.6.5). The post-FHA noble gas and iodine isotopic activities released in the reactor building are calculated using the non-LOCA fraction fission product inventory in gap (Ref. 9.1, Table 3), number of damaged fuel rods of 124 (Ref. 9.11), a conservative radial peaking factor of 1.75 (Refs. 9.14 & 9.15), and iodine DF of 200 (Ref. 9.1, Appendix B, Section 2). The post-FHA activity released in the RB is normalized based on the core thermal power level to obtain the Ci/MW_t. The RADTRAD3.02 Nuclide Inventory File (NIF) Bwr_def.NIF is modified based on the Ci/MW_t information developed in Table 1. The newly developed NIF HEPUFHA_def.txt is used in the analysis. In Table 1, the KR-85 and I-131 isotopic core inventories are multiplied by factors 2 and 1.6 respectively to make Table 3 of RG 1.183 gap fractions consistent with the noble gases and halogens release fraction 0.05.

The plant-specific NIF HEPUFHA_def.txt is further modified to include Kr-83m, Xe-131m, Xe-133m, Xe-135m, and Xe-138 isotopes. The RADTRAD3.02 dose conversion factor (DCF) File Fgr11&12.inp is modified to include the DCFs for the added noble gas isotopes using data obtained from References 9.7 & 9.8. The modified DCF file HCGS_FG11&12.txt is used in the FHA analysis. The RADTRAD3.02 code is verified and validated as an accepted software in Reference 9.18.

Post-FHA activity can be potentially released to the environment through the south plant vent, Filtration Recirculation and Ventilation System (FRVS) vent, or RB truck bay door at ground level (Refs. 9.20 & 9.21). The χ/Qs for these release paths are obtained from Reference 9.5, Sections 8.1, 8.2 and 8.4, and provided in the following table:

Time Interval (hr)	HCGS Control Room 95% Atmospheric Dispersion Factors (X/Qs) (s/m^3)		
	FRVS Vent	South Plant Vent	RB Truck Bay
0-2	1.25E-03	5.75E-04	1.39E-03
2-8	8.09E-04	3.84E-04	1.17E-03
8-24	3.04E-04	1.40E-04	4.76E-04
24-96	2.10E-04	9.08E-05	3.20E-04
96-720	1.59E-04	7.01E-05	2.60E-04

Comparison of χ/Qs in the above table indicates that the RB truck bay release path is the most limiting release path for the post-FHA release. Therefore, the CR dose is calculated using the post-FHA release through the RB truck bay door. Almost all of the radioactive material released from the damaged fuel pins is assumed to be released to the environment over a 2-hour period. (Ref. 9.1, Appendix B, Section B5.3). The resulting doses at the EAB, LPZ, and CR locations are compared with the dose acceptance criteria in Section 7.0.The RADTRAD computer run HEPU3300FHA00 is modified to incorporate the uprated thermal power level and post-FHA activity released to the environment through the reactor building truck bay door at ground level. There is no specific ESF function credited in the analysis except for the scrubbing of the activity in the spent fuel pool (SFP), which is limited by 23 feet height of water over the top of irradiated fuel assemblies seated in the SFP racks (Ref. 9.6.3).

The Control Room Emergency Filtration (CREF) system is not credited in the analysis. The CR is conservatively assumed to be 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.

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. 9.27) 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. 9.25 and 9.26), 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-GU-MDC-1775, Rev 2. The increases in the proposed EAB, LPZ, & CR doses are compared with the 10 CFR 50.59 allowable minimal dose increases in Section 7.2. Similarly, the proposed calculated total doses are compared with the allowable regulatory guide dose limits. The comparison in Section 7.2 confirms that the increase in the EAB, LPZ, & CR doses and the total calculated doses are less than the corresponding minimal dose increases and allowable regulatory guide limits. Therefore, pursuant to 10 CFR 50.59 guidance as defined in References 9.17 and 9.18, the proposed increase in the core thermal power level and resulting post-FHA doses can be adopted as current design and licensing bases for the HCGS.

4.0 <u>ASSUMPTIONS:</u>

Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident (FHA)

The assumptions in these sections are acceptable for evaluating the radiological consequences of a FHA. These assumptions supplement the guidance provided in Regulatory Guide 1.183, Appendix B (Ref. 9.1).

Source Term Assumptions

4.1 Per Reference 9.1, Section 3.2, for non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3 of RG 1.183. The release fractions from Table 3 are incorporated in Design Input 5.3.1.3 in conjunction with the core fission product inventory in Design Input 5.3.1.2 with an assumed maximum core radial peaking factor of 1.75, which is conservatively greater than the 1.5 value recommended in Safety Guide 25 (Ref. 9.15) and bounding for reactor core design (Ref. 9.14, Item 4). The cycle-specific reload physics analysis should calculate this number. The bromines are neglected from thyroid dose consideration due to their low thyroid dose conversion factors, relatively short half lives, and their decay into insignificant daughters.

4.2 Per Reference 9.1, Appendix B, Section B.1.1, the number of fuel rods damaged during the accident is based on a conservative analysis that considers the most limiting case. Per Reference 9.14, 124 fuel rods are assumed to be damaged (see Design Input 5.3.1.5).

4.3 Per Reference 9.1, Appendix B, Section B.1.2, the fission product release from the breached fuel is based on the fraction of fission product inventory in gap and the estimate of the number of fuel rods breached (See Table 1).

Core Inventory

The inventory of fission products in the reactor core and available for gap release from damaged fuel is based on the maximum power level of 4,031 MWt corresponding to current fuel enrichment and fuel burnup. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides included are xenons, kryptons, iodines, cesiums, and rubidiums. The fraction of fission product in gap activity is shown in Design Input 5.3.1.3. It is further assumed that irradiated fuel shall not be removed from the reactor until the unit has been sub-critical for at least 24 hours (Design Input 5.3.1.7) (Ref. 9.17, page 13, Section 5.1.13).

4.4 Timing of Release Phase

Per Reference 9.1, Section 3.3, for non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and the fuel pellet is assumed to occur instantaneously with the onset of the projected damage.

4.5 Chemical Form

Per Reference 9.1, Appendix B, Section B.1.3, The chemical form of radioiodine released from the fuel to the surrounding water is assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel is assumed to completely dissociate in the pool water. Because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This is assumed to occur instantaneously.

4.6 Water Depth

Since the depth of water above the damaged fuel is 23 feet or greater (Design Input 5.3.2.1), the decontamination factors for the elemental and organic species are 500 and 1 (Design Input 5.3.2.3), respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water) (Design Input 5.3.2.4). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species (Ref. 9.1, Appendix B, Section B.2) (Design Input 5.3.2.5).

4.7 Noble Gases

The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor) (Ref. 9.1, Appendix B, Section B.3).

Fuel Handling Accidents Within Containment

For fuel handling accidents postulated to occur within the containment, the following assumptions are acceptable to the NRC staff (Ref. 9.1, Appendix B, Section B.5).

4.8 Since the containment is open during fuel handling operations (i.e., containment hatch C-9 and RB truck bay door are open) the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period (Design Input 5.3.2.7).

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),
- 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, containment shine dose, and CR filter shine dose due to a FHA 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 FHA.

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 present in the control room are incorporated in design inputs 5.3.3.3 & 5.3.3.4 (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. Many of the design input parameter values used in the analysis are those specified in the technical specifications (Ref. 9.6).

5.1.5 Meteorology Considerations

The control room atmospheric dispersion factors (χ/Qs) for the RB truck bay 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 (Ref. 9.9). 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 Section 3 and Appendix B 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 FHA Occurring In Reactor Building With Hatch Doors Opened Design Input 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 for Discharge Bundle				9.3, Appendix C	
Isotope	**Ci/MW_t**	**Isotope**	**Ci/MW_t**	**Isotope**	**Ci/MW_t**
KR-83M	2.981E+03	I-132	3.991E+04	XE-133	5.306E+04
KR-85	4.711E+02	I-133	5.454E+04	XE-135	1.482E+04
KR-85M	5.908E+03	I-134	5.937E+04	XE-135M	1.118E+04
KR-87	1.097E+04	I-135	5.117E+04	XE-138	4.322E+04
KR-88	1.539E+04	XE-131M	3.129E+02		
I-131	2.779E+04	XE-133M	1.743E+03		

5.3.1.3 Radionuclide Release Fractions.		
Fraction of Fission Product Inventory in Gap		
Group	**Fraction**	9.1, Section 3.2, Table 3
I-131	0.08	
Kr-85	0.10	
Other Noble Gases	0.05	
Other Halogens	0.05	
Alkali Metals	0.12	
5.3.1.4 Radionuclide Composition		
Group	**Elements**	9.1, Section 3.4, Table 5
Noble Gases	Xe, Kr	
Halogens	I, Br	
Alkali Metals	Cs, Rb	
5.3.1.5 Damaged Fuel Rods	124	9.11 & 9.14
5.3.1.6 Number of Fuel Assemblies In Core	764	9.6.4
5.3.1.7 Irradiated Fuel Decay	24 hrs	9.17, page 13, Section 5.1.13
5.3.1.8 Radial Peaking Factor	1.5 1.75	9.15, & 9.24, Page C33 conservatively assumed
5.3.1.9 Number of Fuel Rods per Fuel Assembly	62	9.11 & 9.19
5.3.2 Activity Transport in Reactor Building		
5.3.2.1 SFP Water Depth	23 feet	9.6.1 & 9.6.2
5.3.2.2 Reactor Building Volume	4.00E+06 ft^3	9.6.3
5.3.2.3 Decontamination Factors (DFs) of Iodine		
Elemental	500	9.1, Appendix B, Section 2
Organic	1	

Design Input Parameter	Value Assigned	Reference
5.3.2.4 Overall Effective Decontamination Factors (DFs)		
Total Iodine	200	9.1, Appendix B, Section 2
5.3.2.5 Chemical Form of Iodine Released From Pool Water		
Elemental	57%	9.1, Appendix B, Section 2
Organic	43%	
5.3.2.6 DF of Noble Gas	1	9.1, Appendix B, Section 3
5.3.2.7 Duration of Release (Hr)	2	9.1, Appendix B, Section 5.3
5.3.2.8 FRVS Vent/Recirc Filter Efficiencies		
Species	**Efficiency (%)**	Not Credited In This Analysis
Aerosol	0	
Elemental	0	
Organic	0	
5.3.2.9 Fuel Peak Burnup	58 GWD/MTU < 62,000 GWD/MTU	9.3 & 9.1, Footnote 11
5.3.2.10 Maximum Linear Heat Generation Rate For Peak Rod	< 6.3 kw/ft	Assumed (Ref. 9.1, Footnote 11 and 9.16)
5.3.2.11 Reactor Building contaminated air release rate	1.535E+05 cfm	See Section 6.2
5.3.3 Control Room Model Parameters		
5.3.3.1 CR Volume	85,000 ft^3	9.10, Page 10
5.3.3.2 CR Normal Air In-Flow Rate During FHA	3,000 ± 10% cfm for 0-720 hrs (conservatively modeled as 3,300 cfm)	9.12 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 (m^3/sec)	3.5E-04	9.1, Section 4.2.6
5.3.3.5 CR Atmospheric Dispersion Factors for Reactor Building Truck Bay Release (X/Qs)		
Time (Hr)	**X/Q (sec/m^3)**	
0-2	1.39E-03	9.5, Section 8.2
2-8	1.17E-03	
8-24	4.76E-04	
24-96	3.20E-04	
96-720	2.60E-04	
5.3.4 Site Boundary Release Model Parameters		
5.3.4.1 EAB Atmospheric Dispersion Factor (X/Q) (sec/m^3)	1.9E-04	9.9, Pages 5 & 9

Design Input Parameter	Value Assigned	Reference
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 (m^3/sec)	3.5E-04	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. 9.22)

Proposed Power Level Increase = 20%

Instrument Uncertainty = 2% (Ref. 9.23)

Extended Uprated Power Level = 3,293 MW_t x 1.20 x 1.02 ≈ 4,031 MW_t

6.2 Post-FHA Release Rates

The release rate from the source node – Pool to Reactor Building – is calculated such that 99% of the activity released into the reactor building is released to the environment in two hours.

$A = A_0\, e^{-\lambda t}$

Where;

A_0 = Initial Activity in Source Node

A = Final Activity in Source Node

λ = Removal Rate (vol/hr)

t = Removal Time (hr) = 2.0 hr

Assuming that 99% of activity is released into the environment,

$A/A_0 = 0.01$

Therefore,

$A / A_0 = e^{-\lambda t}$

$0.01 = e^{-2\lambda}$

$\ln (0.01) = -2\lambda \ln(e)$

$-4.605 = -2\,\lambda$

$\lambda = -4.605/-2 = 2.303$ volume/hr

The Release Rate from the 4,000,000 cubic foot Reactor Building (Design Input 5.3.2.2) is:

= 2.303 volume/hr × 4,000,000 ft^3 = 9,212,000 ft^3/hr × 1/60 hr/min = 1.535E+05 ft^3/min

7.0 RESULTS SUMMARY:

7.1 The results of the FHA analysis, which establishes the licensing basis for relaxing the secondary containment integrity, are summarized in the following table:

	Fuel Handling Accident Occurring In Reactor Building TEDE Dose (rem)		
	Receptor Location		
	Control Room	EAB	LPZ
Calculated Dose	3.31E+00	5.27E-01 (0.0 hr)	5.27E-02
Allowable TEDE Limit	**5.0E+00**	**6.3E+00**	**6.3E+00**
	RADTRAD Computer Run No.		
	HEPU3300FHA00	HEPU3300FHA00	HEPU3300FHA00

Significant assumptions used in this analysis:

- Radial peaking factor = 1.75
- Containment hatch C-9 and RB truck bay door remain open for the duration of the accident
- Secondary containment integrity is not credited during the accident
- Essentially all activity (i.e., 99 percent) is released to the environment within 2 hours
- CREF system is not initiated.
- 124 fuel rods damaged
- 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
Fuel Handling Accident (FHA)	H-1-GU-MDC-1775, R2			H-1-ZZ-MDC-1929, Rev 0				
Control Room	N/A	N/A	3.21	3.31	5.00	0.10	0.18	5.00
Exclusion Area Boundary	N/A	N/A	0.515	0.527	25.00	0.012	2.45	6.30
Low Population Zone	N/A	N/A	0.0515	0.0527	25.00	0.001	2.49	6.30

E From 10 CFR 50.67 (Ref. 9.4)
H From RG 1.183, Table 6 (Ref. 9.1)

8.0 CONCLUSIONS:

The analysis results presented in Section 7.1 indicate that the EAB, LPZ, and CR doses are within their allowable limits for a FHA occurring in the reactor building without secondary containment integrity (that is, with the containment hatch C-9 and RB truck bay door open). The results demonstrate that HCGS Technical Specification 3.6.5.1 applicability can be removed when irradiated fuel is being handled in the secondary containment and during core alterations.

The comparisons in Section 7.2 confirm that the proposed increases in the EAB, LPZ, & CR doses are less than the minimal dose increase regulatory limits, and that the proposed total doses are less than the allowable regulatory limits. Therefore, pursuant to 10 CFR 50.59 guidance as defined in References 9.25 and 9.26, the proposed increase in the core thermal power level and resulting post-FHA dose consequences can be adopted as current licensing and design 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. HCGS Technical Specifications:

6.1 Specification 3/4.9.8, Water Level - Reactor Vessel

6.2 Specification 3/4.9.9, Water Level - Spent Fuel Pool Storage Pool

6.3 Specification 5.2.3, Secondary Containment

6.4 Specification 5.3.1, Reactor Core

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. Nuclear Fuel Section Design Input File, T03.5-043, Revised Refueling Accident (Bundle Drop) Analysis
12. HCGS Air Flow Diagram No. M-78-1, Rev 21, Auxiliary Building Control Area
13. NUREG-1433, Volume 2, Rev. 3, April 2001, Standard Technical Specifications General Electric Plants, BWR/4, Bases
14. Response For Nuclear Unit Technical Support (NUTS) Order No. TS980812184
15. U.S. NRC Safety Guide 25, 3/23/72, 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
16. Response For NUTS Order No. 80037558
17. HCGS Procedure No. HC.OP-IO.ZZ-0009(Q), Rev 30, Refueling Operations
18. Critical Software Package Identification No. A-0-ZZ-MCS-0225, Rev 2, RADTRAD Computer Code.
19. Nuclear Fuel Section Design Input File, HCA.5-0004, Hope Creek Thermal Hydraulic Data

20. HCGS Air Flow Diagram No. M-76-1, Rev 19, Reactor Building
21. P&ID M-84-1, Sheet 1 of 2, Rev 29, Reactor Building Exhaust Control Diagram
22. NRC Safety Evaluation Report NUREG-1048, October 1984, Operation of Hope Creek Generating Station
23. U.S. NRC Regulatory Guide 1.49, Rev 1, Power Levels of Nuclear Power Plants
24. HCGS Cycle 11 Core Design Report NFVD-AB-2001-007
25. PSEG Procedure No. NC.NA-AS.ZZ-0059(Q), Rev 11, 10CFR50.59 Program Guidance.
26. Nuclear Energy Institute Report No. NEI 96-07, Rev 1, Guidelines for 10 CFR 50.59 Implementation.
27. 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
Post-FHA Activity Released In Reactor Building Used In RADTRAD Nuclide Inventory File

Isotope	Core Initial Inventory (Ci/MWt) A	Radial Peaking Factor B	Total Number of Fuel Rod In Core C	Number of Fuel Rod Damaged D	Core Inventory In Damaged Fuel (Ci/MWt) E=A*B*D/C	DF F	Activity In RB Bldg For RADTRAD Code Nuclide Inventory File (Ci/MWt) G=E/F	Activity In RB Bldg For RADTRAD Code Nuclide Inventory File RADTRAD (Ci/MWt) I=G*1
KR 83M	2.981E+03	1.75	47368	124	1.366E+01	1.0	1.366E+01	.1366E+02
KR 85*	9.422E+02	1.75	47368	124	4.316E+00	1.0	4.316E+00	.4316E+01
KR 85M	5.908E+03	1.75	47368	124	2.707E+01	1.0	2.707E+01	.2707E+02
KR 87	1.097E+04	1.75	47368	124	5.026E+01	1.0	5.026E+01	.5026E+02
KR-88	1.539E+04	1.75	47368	124	7.050E+01	1.0	7.050E+01	.7050E+02
I131**	4.446E+04	1.75	47368	124	2.037E+02	200.0	1.018E+00	.1018E+01
I132	3.991E+04	1.75	47368	124	1.828E+02	200.0	9.142E-01	.9142E+00
I133	5.454E+04	1.75	47368	124	2.499E+02	200.0	1.249E+00	.1249E+01
I134	5.937E+04	1.75	47368	124	2.720E+02	200.0	1.360E+00	.1360E+01
I135	5.117E+04	1.75	47368	124	2.344E+02	200.0	1.172E+00	.1172E+01
XE-131M	3.129E+02	1.75	47368	124	1.433E+00	1.0	1.433E+00	.1433E+01
XE133	5.306E+04	1.75	47368	124	2.431E+02	1.0	2.431E+02	.2431E+03
XE-133M	1.743E+03	1.75	47368	124	7.985E+00	1.0	7.985E+00	.7985E+01
XE135	1.482E+04	1.75	47368	124	6.789E+01	1.0	6.789E+01	.6789E+02
XE135M	1.118E+04	1.75	47368	124	5.122E+01	1.0	5.122E+01	.5122E+02
XE-138	4.322E+04	1.75	47368	124	1.980E+02	1.0	1.980E+02	.1980E+03

A from Reference 10.3 except noted as follows

* KR-85 activity is multiplied by a factor 2 (0.1/0.05) to account for additional fractional release.

** I-131 activity is multiplied by a factor 1.6 (0.08/0.05) to account for additional fractional release.

11.0 FIGURES:

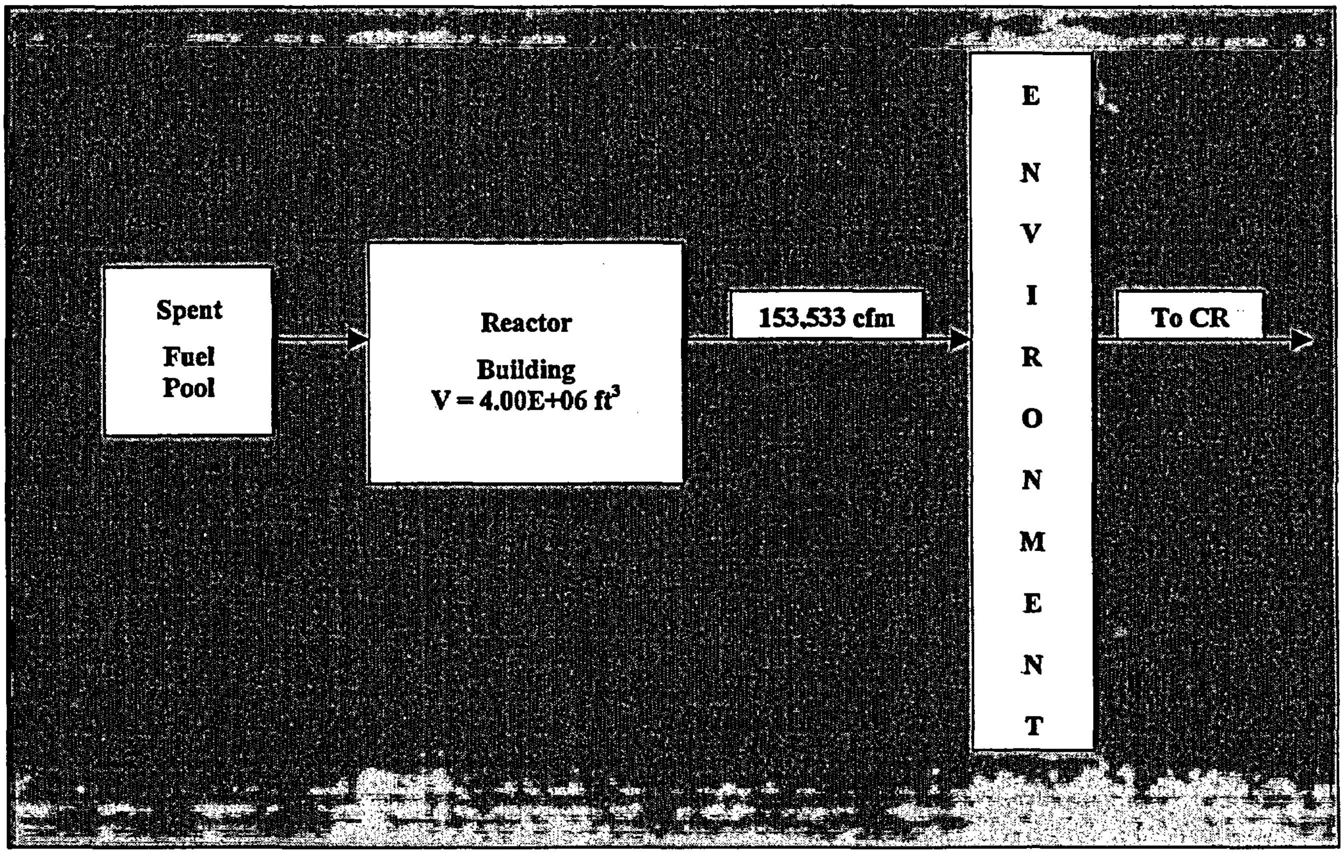

Figure 1: RADTRAD Nodalization For FHA Occurring In Reactor Building With Hatch Opened

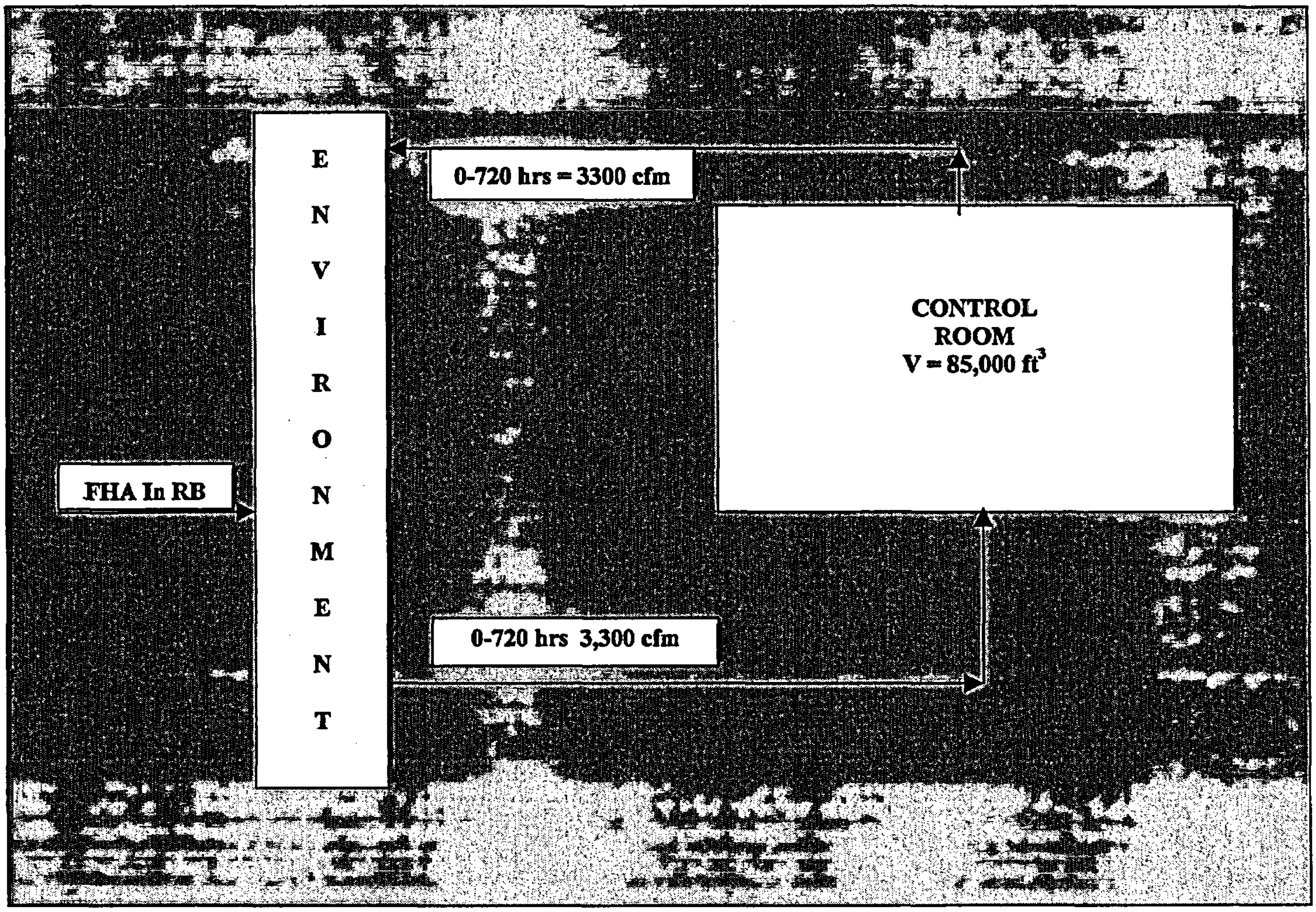

Figure 2 – HCGS Control Room RADTRAD Nodalization

12.0 AFFECTED DOCUMENTS:

The following documents will be revised:

HCGS UFSAR Section 15.7.9, Radiological Consequences
HCGS UFSAR Section 15A.10, Fuel Handling Accident (Section 15.7.4)
HCGS UFSAR Table 15.7.5, Fuel Handling Accident - Parameters Tabulated For Postulated Accident Analyses
HCGS UFSAR Table 15.7.8, Fuel Handling Accident Radiological Effects

The following document will be superseded:

Calculation No. H-1-GU-MDC-1775, Rev 2, Fuel Handling Accident Radiological Consequences

13.0 ATTACHMENTS:

Attachment A : 1 Diskette with the following electronic files:

Calculation No: H-1-ZZ-MDC-1929, Rev 0.
Comment Resolution Form 2 – Mark Drucker
Certification for Design Verification Form-1
RCPD Form-1