Attachment 6

Calculation S-C-ZZ-MDC-1920, Rev. 4IR0

Fuel Handling Accident Radiological Consequences Evaluation

(NC.DE-AP.ZZ-0002(Q), Rev. 12, Form 1)	CALCULATION COVER SHEET	Page 1 of	45

CALCULATION NUMBER:	S-C-ZZ-MDC-1920	REVISION:	4IR0
TITLE:	Fuel Handling Accidents Radiological Consequences		

#SHTS (CALC):	45	#ATT/#SHTS:	2/3	#IDV/50.59/72.48 SHTS:	6 /4/0	#TOTAL SHTS:	58

CHECK ONE: JD 5/18/06

☒ 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 70038184-0280]. 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): N/A

DESCRIPTION OF CALCULATION REVISION (IF APPL.):

The analysis is revised to calculate doses at various decay times in support of an anticipated submittal for a Technical Specification change. The nature of revision is such that the entire calculation is revised.

PURPOSE:

The purpose of this analysis is to determine the Exclusion Area Boundary (EAB), Low Population Zone (LPZ) and Control Room (CR) doses due to a fuel handling accident (FHA) occurring in the containment building with the containment equipment hatch (CEH) open and in the fuel handling building. The FHA analyses are performed using the Alternative Source Term (AST), guidance in the Regulatory Guide 1.183, Appendix B, TEDE dose criteria, and various fuel decay times.

CONCLUSIONS:

The Sections 8.1 and 8.2 results indicate that the EAB, LPZ, and CR doses are within their respective allowable limits for the FHAs occurring in the containment building and fuel handling building. The FHA occurring in the containment provides basis for changing the following SNGS Technical Specification requirements:

1. The irradiated fuel assemblies can be handled in the reactor pressure vessel (RPV) after the reactor has been sub-critical for at least 24 hours. This provides a basis for changing the reactor minimum sub-critical time from 168 hours to 24 hours (Technical Specification Limiting Condition for Operation (LCO) 3.9.3)
2. The irradiated fuel assemblies can be moved with the containment equipment hatch and personnel locks opened, and all containment penetrations opened in the piping penetration areas without containment integrity (operability) (Technical Specification LCO 3.9.4)
3. The core alterations can be performed without containment integrity (Technical Specification LCO 3.9.4).

The FHA occurring in the FHB provides basis for relaxing the SNGS Technical Specification Surveillance requirements 4.9.12.b and 4.9.12.c.

	Printed Name / Signature	Date
ORIGINATOR/COMPANY NAME:	Gopal J. Patel/NUCORE	05/17/2006
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CONTRACTOR SUPERVISOR (If applicable)	N/A	
PSEG SUPERVISOR APPROVAL: (Always required)	Alan A. Johnson/PSEG JD	5/18/06

REVISION HISTORY

Revision	Description
0	Original Issue
1	Editorial changes to various sections
2	Revised EAB χ/Qs and changes to various sections
3	Revised to simplify the calculation title, correct a typographical error in Section 4.8a identified in Notification 20104610 IAW NUTS Order 80048072 and correct a typographic error in the heading for Section 6.0. Additionally, revised Section 9.0 to limit the discussion to conclusion and added Section 12.0, identifying affected documents (there are none relating to the revision).
4	The analysis is revised to calculate doses at various decay times in support of an anticipated submittal for a Technical Specification change. The nature of revision is such that the entire calculation is revised.

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1.0 PURPOSE

The purpose of this analysis is to determine the Exclusion Area Boundary (EAB), Low Population Zone (LPZ) and Control Room (CR) doses due to a fuel handling accident (FHA) occurring with the reactor being subcritical for various times in:

1. The containment building (CB) with the containment equipment hatch (CEH), personnel air locks, and other containment penetrations open or
2. The fuel handling building (FHB)

The analyses are performed using the Alternative Source Term (AST), guidance in Regulatory Guide 1.183, Appendix B, and TEDE dose criteria with the different fuel decay times.

2.0 BACKGROUND

PSEG Nuclear is expected to change the minimum fuel decay time requirement for the reactor to be subcritical prior to the movement of irradiated fuel assemblies (Ref. 10.6.2). Fuel handling accidents are postulated in the RB and FHB with the reactor being subcritical for various times. Activity is released to the environment through the opened CEH or the plant vent (PV). The releases are modeled as ground-level releases.

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

- 3.9.3 DECAY TIME

  The reactor shall be subcritical for at least 100 hours prior to movement of irradiated fuel in the reactor pressure vessel (Ref. 10.6.2). This requirement for the subcritical time is expected to change.
- 3.9.4 CONTAINMENT BUILDING PENETRATION

  The containment building penetrations shall be operable during CORE ALTERATIONS or movement of irradiated fuel within containment (Ref. 10.6.1).
- 3.9.10 WATER LEVEL – REACTOR VESSEL

  At least 23 feet of water shall be maintained over the top of the reactor pressure vessel (RPV) flange (Ref. 10.6.3).

- 3.9.11 STORAGE POOL WATER LEVEL

  At least 23 feet of water shall be maintained over the top of the irradiated fuel assembly seated in the storage racks (Ref. 10.6.9).

- 1.25 RATED THERMAL POWER (RTP)

  RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3459 MWt (Ref. 10.6.4).

- 5.3.1 FUEL ASSEMBLIES

  The reactor shall contain 193 fuel assemblies (Ref. 10.6.5).

- 3.3.3.1 RADIATION MONITORING INSTRUMENTATION

  The radiation monitoring instrumentation channels shown in Technical Specification Table 3.3-6 shall be operable with their alarm/trip setpoints with the specified limits (Ref. 10.6.6).

- TABLE 3.3-6 RADIATION MONITOR INSTRUMENTATION

  The control room normal intake radiation monitors must be operable during fuel movement (Ref. 10.6.7).

- 3.9.12 Fuel Handling Area Ventilation System

  The fuel handling area ventilation system shall be operable (Ref. 10.6.10).

3.0 ANALYTICAL APPROACH

This analysis uses Version 3.02 of the RADTRAD computer code (Ref. 10.2) to calculate the potential radiological consequences of an FHA. The RADTRAD code is documented in NUREG/CR-6604 (Ref. 10.2). The RADTRAD code is maintained as Software ID Number A-0-ZZ-MCS-0225 (Ref. 10.33).

The FHA is analyzed using the plant specific design inputs. The design inputs are compatible to the AST and TEDE dose criteria.

The scrubbing of the iodine activity in the reactor cavity and spent fuel storage pool are credited in the analyses. The scrubbing effects are limited by 23 feet height of water over the top of the RPV flange (Ref. 10.6.3) and over the top of the irradiated fuel assemblies in the spent fuel pool storage racks (Ref. 10.6.9).

The core inventory is obtained from Reference 10.3 (page 33, Table 2), which is calculated based on a thermal power level of 3,600 MW_t. The radial peaking factor of 1.7 is conservatively used instead of the 1.65 value

recommended in Reference 10.19. The thermal power level of 3,632 MW_t, which is 105% of the rated thermal power level of 3,459 MW_t (Ref. 10.6.4), is used in the analysis to provide a margin for future power uprate. The core activity obtained from Reference 10.3 is listed in Table 1 and normalized in Tables 1, 2 & 3 based on the core thermal power level, the gap fission product release fractions in Design Input 5.3.1.3, peaking factor, and one fuel assembly failed during the FHA (Ref. 10.19, page 5). The maximum linear heat generation rate is limited to less than 6.3 kw/ft peak rod average power (Ref. 10.1, Table 3, Note 11). The high power density of cores in Pressurized Water Reactors (PWRs), increased fuel burnup, and extended fuel cycle potentially may increase the maximum heat generation rate to a value exceeding the limit of 6.3 kw/hr peak rod average power for burnups exceeding 54 GWD/MTU at the end of the fuel cycle. Many PWR core design loading analyses have reported fuel assemblies that have exceeded the maximum heat generation rate of 6.3 kw/ft. Therefore, to establish a conservative basis for those fuel assemblies that may in future cycle operations exceed the maximum heat generation rate of 6.3 kw/hr, the gap fission product fractions in Table 3 of RG 1.183 are doubled to the values shown in Section 5.3.1.3 for use in this FHA dose analysis (Table 2). The RADTRAD V3.02 code default nuclide inventory file (NIF) Bwr_def. NIF is modified based on the normalized Ci/MW_t in Table 3. The plant-specific NIF SNGSFHA_def is further modified to include Kr-83m, Xe-131m, Xe-133m, Xe-135m, and Xe-138 isotopes. The RADTRAD V3.02 dose conversion factor (DCF) File Fgr11&12 (based on Refs. 10.7 and 10.8) is modified to include the DCFs for the added noble gas isotopes. The modified DCF file SALEMFHA_FG11&12 is used in the FHA analyses.

3.1 FHA Occurring In Containment Building

There are one CEH, two personnel air locks, and containment piping penetrations in the containment boundary (Ref. 10.17). The CEH provides a direct release path to the environment (Refs. 10.17.a, 10.17.b, 10.17.g). The personnel air locks and penetrations provide release paths to the environment through the plant vent via piping penetration areas (Refs. 10.17 & 10.18). The most limiting atmospheric dispersion factors for these release paths are obtained from Reference 10.5 and compared in the following table.

Time Interval (hr)	Salem 1 CR Intake χ/Qs (s/m^3)	
	Unit 1 Equip Hatch Unit 1 CR Intake	Unit 1 Plant Vent Unit 1 CR Intake
0-2	2.86E-03	1.78E-03
2-8	2.22E-03	1.31E-03
8-24	9.15E-04	5.22E-04
24-96	6.60E-04	3.77E-04
96-720	5.62E-04	3.17E-04

The comparison of χ/Qs in the above table indicates that the CEH provides a conservative release path for the FHA occurring in the containment. Therefore, the EAB, LPZ, and CR doses are calculated using the post-FHA release through the CEH. The activity release rate from the CEH is calculated in Section 7.2 based on the removal of 99% of radioactive material released from the damaged fuel to the environment over a 2-hour period. (Ref. 10.1, Appendix B, Regulatory Position B.5.3). The resulting doses at the EAB, LPZ, and CR locations are compared with the regulatory allowable limits in Section 8.1.

3.2 FHA Occurring In Fuel Handling Building

A parametric study is performed to determine a conservative release model using either a post-FHA release rate based on a 0-2 hour release, or a rapid release rate based on one FHB volume per minute. The results of the parametric study shown in Sections 8.2 & 8.3 indicate that a release based on the rapid release rate of one FHB volume per minute yields a higher CR dose. The puff release yields a higher CR dose because it results in a larger amount of unfiltered iodine activity entering the CR volume prior to the one minute start of the Control Room Emergency Air Conditioning System (CREACS) outside air inflow filtration.

Should a FHA occur in the FHB, the activity can be either released through the plant vent (Ref. 10.18) or the FHB rollup door at ground level (Ref.10.23). However, the following post-FHA release paths are identified in Reference 10.21 during the FHB pressurization due to a single failure of one FHB exhaust fan:

1. Release through the plant vent, via one operational FHB exhaust fan, at a rate of 15,300 cfm
2. Leakage through truck bay roll-up door at a rate of 3,883 cfm
3. Leakage through gravity damper (that replaced the truck bay exhaust fan) at a rate of 256 cfm

The atmospheric dispersion factors (χ/Qs) for the plant vent and FHB rollup doors are calculated in Reference 10.5, Sections 8.2 & 8.3, respectively, using the ARCON96 computer code. The χ/Qs for the gravity damper release are conservatively assumed to be same as those for a smoke hatch. The smoke hatch χ/Qs are developed in Reference 10.9, Section 8.4 using the ARCON96 computer code. Since the FHA in the FHB release duration is two hours (Ref. 10.1, Appendix B, RGP B.4.1), the plant vent, FHB rollup doors and smoke hatch 0-2 χ/Q values are used to calculate the equivalent 0 to 2 hr χ/Q in Section 7.5 for a combined post-FHA release path. The equivalent χ/Q is used with the post-FHA unfiltered release from the FHB to calculate the EAB, LPZ, and CR doses. Activity from the FHB is assumed to be released to the environment at a rate of 21,439 cfm (design flow rate + 10%). The resulting doses at the EAB, LPZ, and CR locations are compared with the regulatory allowable limits in Section 8.2.

3.3 Post-FHA Technical Support Center (TSC) Habitability

The TSC habitability is additionally evaluated to fulfill the PSEG Licensing request to evaluate the post-FHA TSC dose. The TSC is located in the Clean Facilities Building (CFB) at the second and third floors (Refs. 10.27.b & 10.27.c). The CFB is located southeast of the Unit 1 containment building (Ref. 10.28). As discussed in Section 3.1 above, the CEH and PV are the release points for the FHA occurring in the containment. As discussed in Section 3.2 above, the plant vent, FHB rollup doors and gravity damper (modeled as the smoke hatch) are the release points for the FHA occurring in the FHB. The TSC emergency air intake is in the Mechanical Equipment Room located on the roof of CFB (Refs. 10.26, 10.27, & 10.29). The TSC is located closer to Unit 1 containment compared to Unit 2 containment, therefore, the distances between the Unit 1 CEH & PV and TSC intake are calculated in Section 7.6. These distances are compared with the corresponding distances to the Unit 1 CR intake in Section 7.6. The CR doses are considered bounding for TSC for the FHA occurring in the containment and FHB because:

1. The TSC intake is located farther from the subject release points in comparison to the CR intakes. Therefore, the values of corresponding TSC intake χ/Qs will be lower than CR intake χ/Qs and the resulting post-FHA TSC doses will be lower in the same proportion of χ/Qs values.
2. The comparison of CR χ/Qs in Reference 10.5, Section 8.1, indicates that the variation of χ/Qs due to change in wind direction is insignificant. Therefore, the TSC χ/Qs will not be impacted by the differences in wind direction for 0-2 hr period.

3. Manning the TSC occurs some time after initiation of the postulated accident. Therefore, at first there will be a period with no occupancy during the initial phase of the accident.

3.4 CR Intake Monitor Response

There are two radiation monitors in each normal CR air intake duct having the alarm/trip set point of 2.48×10^3 cpm (Refs. 10.6.7 & 10.13). These monitors are classified as safety related (Ref. 10.13), are required to be operable in all modes and during movement of irradiated fuel assemblies and during CORE ALTERATION (Refs. 10.6.6 & 10.6.7), are powered by emergency power sources (Ref. 10.22), and are instantaneously actuated by exceeding a predetermined setpoint (Ref. 10.6.7 & Section 7.4). The post-FHA activity at the CR air intake will instantaneously reach the Alert/Trip setpoint (Section 7.4) and actuate the monitors. Therefore, these monitors are credited for automatic initiation CR Emergency Air Conditioning System (CREACS). The CR intake monitor preferential alignment of less contaminated air intake is conservatively not credited. The delay associated with the CR intake damper closure time (20 seconds) (Ref. 10.14, page 8), diesel generator speedup time (13 seconds) (Ref. 10.6.8) if the loss of offsite power is assumed to occur at the time of damper closure, and over-all monitor response time (4 seconds) (Ref. 10.14, Appendix A). The total delay time is less than 1.0 minute. A delay of 1 minute is assumed in the analysis for the initiation of the Control Room Emergency Air Conditioning System (CREACS) and the control room envelope isolation.

4.0 ASSUMPTIONS

The regulatory requirements in the Regulatory Guide 1.183, Appendix B (Ref. 10.1) are adopted as assumptions in the following section, which are incorporated as design inputs in Section 5.3 along with other plant-specific as-built design parameters. The assumptions in this section are acceptable by the Staff for evaluating the radiological consequences of FHA occurring in the containment building.

Source Term Assumptions

4.1 Per Reference 10.1, Regulatory Position 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 the Design Input 5.3.1.3 in conjunction with the core fission product inventory in Design Input 5.3.1.2 with the maximum core radial peaking factor of 1.70 (Ref. 10.19) and the core inventory at 3,632 MWt power level. The bromines are neglected from thyroid dose consideration due to their low thyroid dose conversion factors, relatively short half-lives, and decay into insignificant daughters.

4.2 Per Reference 10.1, Appendix B, Regulatory Position B.1.1, the number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. One spent fuel assembly is assumed to be damaged (see Design Input 5.3.1.5). Reference 10.31, Section 3.1.3, Risk Significance, indicates that there have been several occasions when fuel bundles have been dropped during fuel handling. In each case, the actual releases from fuel have been minimal or nonexistent. This evidence shows that the assumption of damage of one fuel assembly in the radiological analysis for a FHA is conservative.

4.3 Per Reference 10.1, Appendix B, Regulatory Position B.1.2, the fission product release from the breached fuel is based on fraction of fission product inventory in gap (RGP 3.2) and the estimate of the number of fuel rods breached (See Table 3).

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 3,632 MWt corresponding to current fuel enrichment and fuel burnup. All the gap activity in the damaged rods is assumed to be instantaneously released. The radionuclides included are xenons, kryptons, and iodines. 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).

4.4 Timing of Release Phase

Per Reference 10.1, Regulatory Position 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 10.1, Appendix B, Regulatory Position B.1.3, The chemical form of radioiodine released from the fuel to the surrounding water should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodine. 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

If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, 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). 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. 10.1, Appendix B, RGP B.2).

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. 10.1, Appendix B, RGP 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. 10.1, Appendix B, RGP B.5).

4.8a If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is 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 (Ref. 10.1, Appendix B, RGP B.5.3). The activity release from the damaged fuel is postulated to mix in the RB volume and release to the environment at a rate such that 99% of post-FHA activity is removed from the RB volume (Section 7.2) (Figure 1).

Fuel Handling Accidents Within The Fuel Building

For fuel handling accidents postulated to occur within the fuel building, the following assumptions are acceptable to the NRC staff.

4.8b The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period (Ref 10.1, Appendix B, RGP B.4.1). The activity released from the damaged fuel is postulated to mix in the FHB volume and be released to the environment over a two hour period at a rate of 21,439 cfm per Design Input 5.3.3.3 (See Figure 2).

A reduction in the amount of radioactive material released from the fuel pool by engineered safety feature (ESF) filter systems is not accounted for in the radioactivity release analyses.

<u>Offsite Dose Consequences</u>

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 and used in determining compliance with the dose acceptance criterion in Reference 10.1, Appendix B, RGP 4.4 and RGP Table 6.

EAB Dose Acceptance Criteria: 6.3 Rem TEDE

4.10 The breathing rates for persons at offsite locations are given in Reference 10.1, RGP 4.1.3, which are incorporated in Design Input 5.3.5.4.

4.11 TEDE is determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and is used in determining compliance with the dose acceptance criterion in Reference 10.1, RGP 4.4 and RGP Table 6.

LPZ Dose Acceptance Criteria: 6.3 Rem TEDE

4.12 No correction is made for depletion of the effluent plume by deposition on the ground (Ref 10.1, RGP 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 10.1, RGP 4.2.1):

- Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the post-accident radioactive plume released from the facility (via CR air intake),
- Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope (via CR unfiltered inleakage),
- Radiation shine from the external radioactive plume released from the facility (external airborne cloud),
- Radiation shine from radioactive material in the reactor containment (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 (CR filter shine dose).

Note: The external airborne cloud dose, containment shine dose, and CR filter shine dose due to 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 10.1), therefore, these direct dose contributions are considered to be insignificant and are not evaluated for a FHA.

4.14 The radioactivity releases and radiation levels used in the control room dose is 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 10.1, RGP 4.2.2).

4.15 The occupancy and breathing rate of the maximum exposed individuals present in the control room are incorporated in design inputs 5.3.4.8 & 5.3.5.3 (Ref. 10.1, RGP 4.2.6).

4.16 10 CFR 50.67 (Ref 10.4) establishes the following radiological criterion for the control room.

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

4.17 Credit for engineered safety features that mitigate airborne activity within the control room may be assumed including control room isolation or pressurization, intake or recirculation filtration (Ref. 10.1, RGP 4.2.4). The control room pressurization as a result of CREACS actuation following CR intake monitor response to a FHA (Ref. 10.6.6 & Sections 3.4 & 7.4) is assumed. No credit is taken for the preferential alignment of the outside air emergency intake dampers.

4.18 No credit is taken for KI pills or respirators (Ref. 10.1, RGP 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 SNGS plant specific design inputs and assumptions used in the current facility's design basis FHA analysis 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 comply with RG 1.183, Appendix B requirements.

5.1.2 Credit for Engineered Safeguard 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 normal CR air intake monitors are required to be operable by TS 3.3.3.1 in ALL MODES and during movement of irradiated fuel assemblies and during CORE ALTERATIONS. The normal CR air intake monitor's function of preferential alignment of the less contaminated outside air emergency intake is conservatively not credited (Ref. 10.10, page 49). The CREACS charcoal filtration operation is credited (Ref. 10.6.15) with a 1-minute system response delay. The FHB safety related charcoal filtration system is conservatively not credited in the analysis.

5.1.3 Meteorology Considerations

The control room atmospheric dispersion factors (χ/Qs) for the CEH, PV, and FHB rollup door release point are developed (Ref. 10.5) using the NRC sponsored computer code ARCON96 and guidance provided for the use of ARCON96 in the Regulatory Guide 1.194. The EAB and LPZ χ/Qs are calculated using the SNGS plant specific meteorology and appropriate regulatory guidance (Ref. 10.16). The site boundary χ/Qs in Reference 10.16 were accepted by the staff in the previous licensing proceedings.

5.2 Accident-Specific Design Inputs/Assumptions

The design inputs/assumptions utilized in the post-FHA 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 Regulatory Position 3 and Appendix B of RG 1.183 (Ref. 10.1). 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 Source Term and Transport Parameters		
5.3.1 Source Term		
5.3.1.1 Core Power Level	3,459 MW_t 3,632 MW_t (3,459 MW_t x 1.05)	10.6.4 Used in the analysis
5.3.1.2 Isotopic Core Inventory @ 3,600 MW_t		10.3, Table 2

Core Inventory (Ci)

Isotope	Activity	Isotope	Activity	Isotope	Activity
KR-83M	1.20E+07	I-132	1.40E+08	XE-133	2.00E+08
KR-85M	2.60E+07	I-133	2.00E+08	XE-135	5.00E+07
KR-85	1.10E+06	I-134	2.20E+08	XE-135M	4.00E+07
KR-87	4.70E+07	I-135	1.90E+08	XE-138	1.60E+08
KR-88	6.70E+07	XE-131M	7.00E+05		
I-131	9.90E+07	XE-133M	2.90E+07		

5.3.1.3 Radionuclide Release Fractions (10.1, RGP 3.2, Table 3)

Group	Fraction	Fraction Used in Analysis
I-131	0.08	0.16
Kr-85	0.10	0.20
Other Noble Gases	0.05	0.10
Other Halogens	0.05	0.10
Alkali Metals	0.12	0.24

5.3.1.4 Radionuclide Composition

Group	Elements	Reference
Noble Gases	Xe, Kr	10.1, RGP 3.4, Table 5
Halogens	I, Br	
Alkali Metals	Cs, Rb	

Design Input Parameter	Value Assigned	Reference
5.3.1.5 Number of Damaged Fuel Assembly	1	Assumed per Assumption 4.2
5.3.1.6 Number of Fuel Assemblies In Core	193	10.6.5
5.3.1.7 Irradiated Fuel Decay Time	96 Hrs used in the analysis 72 Hrs 60 Hrs 48 Hrs 24 Hrs	Assumed
5.3.1.8 Radial Peaking Factor	1.65 (1.70 used in the analysis)	10.19

Design Input Parameter	Value Assigned	Reference
5.3.2 Activity Transport in Containment Building		
5.3.2.1 Refueling Cavity Water Depth	23 feet	10.6.3
5.3.2.2 Containment Building Free Air Volume	2.62E+06 ft^3	10.11
5.3.2.3 Iodine Decontamination Factors (DFs)		
Elemental	500	10.1, Appendix B, Section 2
Organic	1	
5.3.2.4 Overall Effective Decontamination Factor (DFs) for Iodine		
Total Iodine	200	10.1, Appendix B, Section 2
5.3.2.5 Chemical Form of Iodine Released From Pool Water		
Elemental	57%	10.1, Appendix B, Section 2
Organic	43%	
5.3.2.6 DF of Noble Gas	1	10.1, Appendix B, Section 3
5.3.2.7 Duration of Release (hr)	2	10.1, Appendix B, Section 5.3
5.3.2.8 Containment Exhaust From Ring Header	35,000 cfm	10.18.g & 10.18.h
5.3.2.9 Activity release rate	100,600 cfm	See Section 7.2.1
5.3.3 Activity Transport in Fuel Handling Building		
5.3.3.1 Spent Fuel Pool Storage Water Depth	23 feet	10.6.9
5.3.3.2 Fuel Handling Building Volume	558,550 ft^3	Section 7.2.2
5.3.3.3 Activity release rate	21,439 cfm (19,490 x 1.1 = 21,439 cfm)	10.18.a, 10.18.d, & 10.21
5.3.3.4 FHB Charcoal Filter Efficiencies	Not credited in the analysis	10.6.10
The remaining FHA occurring in the FHB source term and activity transport design input parameters are the same as those for a FHA occurring in the containment (see design inputs 5.3.1 and 5.3.2)		
5.3.4 Control Room Model Parameters		
5.3.4.1 CR Volume	81,420 ft^3	10.12, page 33
5.3.4.2 CR Normal Flow Rate	1,320 cfm	Section 7.3
5.3.4.3 CREACS Design Makeup Flow Rate	2,200 cfm	10.6.13
5.3.4.4 CREACS Ventilation Flow Rate	8,000 cfm ± 10% cfm 5,000 cfm (used in analysis)	10.6.12 Section 7.3
5.3.4.5 CREACS Charcoal Filter Efficiency	95%	Section 7.7

Design Input Parameter	Value Assigned	Reference
5.3.4.6 CREACS HEPA Filter Efficiency	99% 95%	Section 7.7 Used in Analysis
5.3.4.7 CR Unfiltered Inleakage	150 cfm (nominal value measured is less than 100 cfm)	10.32, Table 1
5.3.4.8 CR Occupancy Factors		
Time (Hr)	%	10.1, RGP 4.2.6
0-24	100	
24-96	60	
96-720	40	
5.3.4.9 CR Breathing Rate)	3.5E-04 m^3/sec	10.1, RGP 4.2.6
5.3.4.10 Unit 1 CR χ/Qs – Post-FHA Release From Unit 1 CEH		
Time (Hr)	**χ/Q (sec/m^3)**	
0-2	2.86E-03	10.5, page 33
2-8	2.22E-03	
8-24	9.15E-04	
24-96	6.60E-04	
96-720	5.62E-04	
5.3.4.11 FHB 0-2 hr Equivalent χ/Q	1.85E-03 s/m^3	Section 7.5
5.3.4.12 Unit 1 CR χ/Qs – Post-FHA Release From Unit 1 Plant Vent		
Time (Hr)	**χ/Q (sec/m^3)**	
0-2	1.78E-03	10.5, page 34
2-8	1.31E-03	
8-24	5.22E-04	
24-96	3.77E-04	
96-720	3.17E-04	
5.3.4.13 Unit 1 CR χ/Qs – Post-FHA Release From FHB Rollup Door		
Time (Hr)	**χ/Q (sec/m^3)**	
0-2	1.50E-03	10.5, page 35
2-8	1.20E-03	
8-24	4.48E-04	
24-96	3.22E-04	
96-720	2.50E-04	

Design Input Parameter	Value Assigned	Reference
5.3.4.14 Unit 1 CR χ/Qs – Post-FHA Release From Smoke Hatch		
Time (Hr)	**χ/Q (sec/m^3)**	
0-2	1.15E-02	10.9, Section 8.4
2-8	9.28E-03	
8-24	3.50E-03	
24-96	2.49E-03	
96-720	2.02E-03	
5.3.5 Site Boundary Release Model Parameters		
5.3.5.1 EAB Atmospheric Dispersion Factor (X/Q) (sec/m^3)	1.30E-04	10.16, Table 5
5.3.5.2 LPZ Atmospheric Dispersion Factors (X/Qs)		
Time (Hr)	**X/Q (sec/m^3)**	10.16 Table 5
0-2	1.86E-05	
2-8	7.76E-06	
8-24	5.01E-06	
24-96	1.94E-06	
96-720	4.96E-07	
5.3.5.3 CR Breathing Rate (m^3/sec)	3.5E-04	10.1, RGP 4.2.6
5.3.5.4 Offsite Breathing Rate (m^3/sec)		
Time (Hr)	**(m^3/sec)**	10.1, RGP 4.1.3
0-8	3.5E-04	
8-24	1.8E-04	
24-720	2.3E-04	
5.3.5.5 CR Intake Monitor Xe-133 Sensitivity	6.2×10^7 cpm/µCi/cc	10.13, page 12
5.3.5.6 CR Intake Monitor Alert/Trip Setpoint	2.48×10^3 cpm	10.6.7

6.0 METHODOLOGY

6.1 Post-FHA Activity Release Rates

Activity released from the reactor cavity is uniformly distributed in the entire volume of containment building and released to environment over a two hour time period such that 99% of the activity released from the damaged spent fuel assembly is released to the environment. The post-FHA activity release rate from the containment is calculated in Section 7.2.1.

The FHB volume is back calculated in Section 7.2.2 knowing the FHB exhaust rate of 21,439 cfm and the requirement to remove 99% of the activity in a two hour period.

6.2 Fuel Handling Accident in the FHB with a Failure of an Exhaust Fan

The post-FHA activity releases through three different release paths due to pressurization of the FHB are discussed in Section 3.2 and a composite 0-2 hr χ/Q is calculated for a combined release path is calculated in Section 7.5.

7.0 CALCULATIONS

7.1 SNGS Plant Specific Nuclide Inventory File (NIF) For RADTRAD V3.02 Input

The parameter Ci/MW_t in the RADTRAD V3.02 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 NIF is modified based on the plant-specific isotopic Ci/MW_t information developed in Table 3. The RADTRAD nuclide inventory file SNGSFHA_def.txt is used in the analysis.

7.2 Release Rates

7.2.1 Containment Building

The release rate from the source node – reactor cavity to containment – is calculated such that 99% of the activity released into the containment is released to the environment in two hours. The 1% of the activity remaining in the containment is insignificant.

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

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

Containment Building Release Rate = 2.303 1/hr x 2,620,000 ft^3 x 1 hr/60 min ≅ 100,600 ft^3/min

7.2.2 Fuel Building

Fuel building exhaust flow rate = 19,439 cfm (Ref. 10.18.a & 10.18.d) x 1.10 = 21,439 cfm

A removal rate of 2.303 volume/hr (calculated in the above section) corresponds to removal of 99% of activity from the FHB volume over a two-hour period.

Therefore, the FHB volume can be arbitrarily calculated as follows:

2.303 vol/hr x 1 hr/60 min = 3.838E-02 vol/min

$$\text{FHB Volume} = \frac{21{,}439\ ft^3/min}{3.838E\text{-}02\ vol/min} = 558{,}550\ ft^3$$

This volume is used in the RADTRAD model.

For the scenario of a FHA occurring in the fuel handling building with a rapid release of one volume per minute, a FHB release rate is 558,550 ft^3/min is used in the RADTRAD model.

7.3 Control Room Flow Rates

Normal Flow Rate

Reference 10.20, Note "S" provides the outside air flow rates to Zone 1 from the Unit 1 and Unit 2 air intakes. Zone 1 is the combined control room envelop. The control area air conditioning system (CAACS) normal airflow rate is calculated as follows:

Total CAACS Air Flow Rate = 32,600 cfm (2,200 cfm outside air + 30,400 cfm recirc air)

Zone 1 (control room pressure boundary) Supply Air Flow Rate = 8,000 cfm

Amount of Outside Air To Zone 1

= (Fraction of Total CAACS Air Flow Rate to Zone 1) x (2,200 cfm outside air inflow rate)

= (8,000 cfm / 32,600 cfm) x 2,200 cfm = 0.2454 x 2,200 cfm = 540 cfm

Use 600 cfm for Zone 1 During Normal Plant Operation

Total Amount of Outside Air Flow Rate From Both Intakes = 2 x 600 cfm = 1,200 cfm

Maximum Amount of Outside Air Flow Rate = 1.1 x 1,200 cfm = 1,320 cfm

CREACS Recirculation Flow Rate

CREACS ventilation flow rate = 8,000 cfm ± 10% cfm (Ref. 10.6.12)

Minimum CREACS flow rate = 8,000 cfm – 0.10 x 8,000 cfm = 8,000 cfm – 800 cfm = 7,200 cfm

Net CREACS recirculation flow rate = Minimum CREACS flow rate – CREACS makeup flow rate

7,200 cfm – 2,200 cfm (Ref. 10.6.13) = 5,000 cfm

7.4 CR Intake Monitor Setpoint

FHA In Containment Building

Minimum Xe-133 Concentration at CR Intake

= total Xe-133 release (Table 3) divided between two CR intakes = 3.555E+06 Ci / 2 = 1.778E+06 Ci

$$= 1.778\text{E+06 Ci} \times \frac{1}{2.62\text{E+06 ft}^3} \times 35{,}000\ \text{ft}^3/\text{min (Ref. 10.18.c \& e)} \times \frac{1}{60\ \text{sec/min}} \times 2.86\text{E-03 sec/m}^3$$

= 1.132 Ci/m^3 = 1.132 μCi/cc

CR Intake Monitor Xe-133 Sensitivity = 6.2 x 10^7 cpm/μCi/cc (Ref. 10.13, page 12)

CR Intake Monitor Alarm/Trip Setpoint = 2.48 x 10^3 cpm (Ref. 10.6.7)

CR Monitor Count Rate Due Post-FHA Activity Concentration At CR Intake

$= 1.132\ \mu Ci/cc \times 6.2 \times 10^7\ cpm/\mu Ci/cc = 7.02 \times 10^7 cpm >> 2.48 \times 10^3$ cpm

<u>FHA In Fuel Handling Building</u>

Minimum Xe-133 Concentration at CR Intake

= total Xe-133 release (Table 3) divided between two CR intakes = 3.555E+06 Ci / 2 = 1.778E+06 Ci

$$= 1.778E{+}06\ Ci \times \frac{1}{558{,}550\ ft^3} \times 21{,}439\ ft^3/min\ (\text{Section } 7.2.2) \times \frac{1}{60\ sec/min} \times 1.85E{-}03\ sec/m^3$$

$= 2.104\ Ci/m^3 = 2.104\ \mu Ci/cc$

CR Intake Monitor Xe-133 Sensitivity = 6.0×10^7 cpm/μCi/cc (Ref. 10.13, page 12)

CR Intake Monitor Alarm/Trip Setpoint = 2.48×10^3 cpm (Ref. 10.6.7)

CR Monitor Count Rate Due Post-FHA Activity Concentration At CR Intake

$= 2.104\ \mu Ci/cc \times 6.2 \times 10^7\ cpm/\mu Ci/cc = 1.30 \times 10^7\ cpm >> 2.48 \times 10^3$ cpm

It is clear that the CR intake monitor will instantaneously reach its Alarm/Trip setpoint following a FHA occurring in the containment or fuel handling building.

7.5 Equivalent 0-2 hr χ/Q For FHB Release Path

Plant Vent 0-2 $\chi/Q = 1.78E{-}03\ s/m^3$ (Ref. 10.5, page 34)

FHB Rollup Door 0-2 $\chi/Q = 1.50E{-}03\ s/m^3$ (Ref. 10.5, page 35)

Smoke Hatch 0-2 $\chi/Q = 1.15E{-}02\ s/m^3$ (Ref. 10.9, Section 8.4)

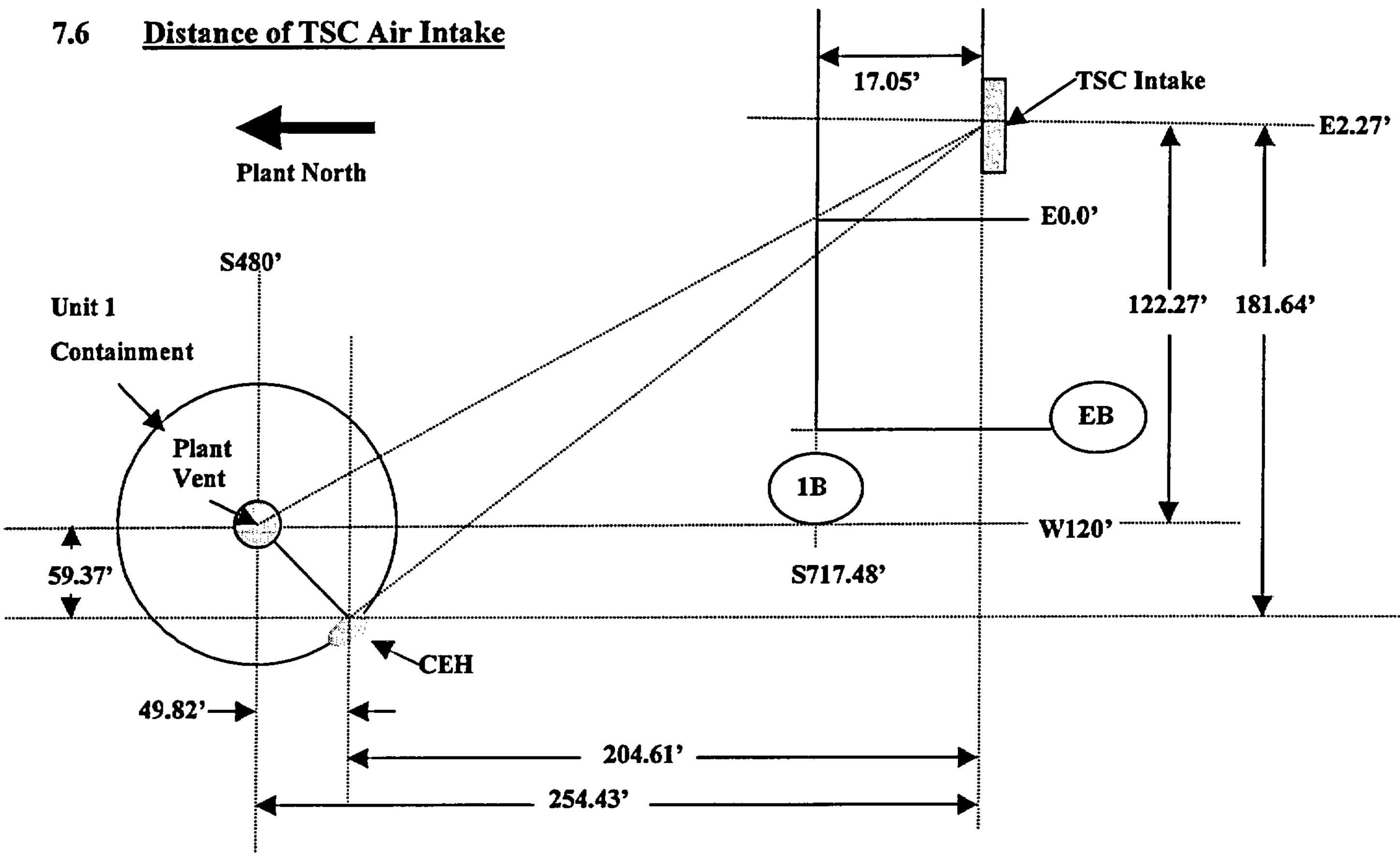
1. Release through the plant vent at a rate of 15,300 cfm (Ref. 10.21)
2. Leakage through truck bay roll-up door at a rate of 3,883 cfm (Ref. 10.21)
3. Leakage through gravity damper 256 cfm (Ref. 10.21)

0-2 hr FHB χ/Q

$$= \frac{15{,}300\ cfm \times 1.78E{-}03\ s/m^3 + 3{,}883\ cfm \times 1.50E{-}03\ s/m^3 + 256\ cfm \times 1.15E{-}02\ s/m^3}{(15{,}300\ cfm + 3{,}883\ cfm + 256\ cfm)}$$

$$= \frac{36.00\ cfm.s/m^3}{19{,}439\ cfm} = 1.85E{-}03\ s/m^3$$

7.6 <u>Distance of TSC Air Intake</u>



South Coordinate of Unit 1 Containment = South Coordinate of Plant + Distance between Centerlines of Plant and Unit 1 Containment

= S320.0' (Ref. 10.23.a) + 160'-0" (Ref.10.23.b) = S480.0'

South Coordinate of Column 1B of Clean Facility Building (CFB)

= South Coordinate of CFB + Distance between South Coordinate and Column 1B

= S715.88' (Ref. 10.28) +1'-6" (Ref. 10.28) = S717.38'

Distance between Column 1B and TSC Air Intake

Distance between Columns 1B and 2B – Distance between 2B and TSC Air Intake

= 22'-3-1/2" (Ref. 10.27.a) – (4'-8-3/4" + 6-1/8") (Ref. 10.29) = 22'-3-1/2" – 5'-2-7/8" = 17.05'

South Distance between Centerline of Unit 1 Containment and TSC Air Intake

= S717.38 – S480.0' + 17.05' = 237.38' + 17.05' = 254.43'

Distance between Centerlines Unit 1 Containment and CEH = 49.82 (Ref. 10.5, page 26)

South Distance between Centerline Unit 1 CEH and TSC Air Intake

= 254.43'- 49.82' = 204.61'

West Coordinate of Containment Centerline = W120.0' (Ref. 10.23.a)

East Coordinate of Centerline of TSC Air Intake

= East Coordinate of East Wall of CFB – (Distance between East Wall of CFB and Row AB + Distance between Centerlines of Rows AB and BB) + Distance between Row BB and Centerline of TSC Air Intake

= (E30.79' (Ref. 10.28) – 1-6" (Ref. 10.28) - 28'-10-1/4" (Ref 10.28)) + (1'-0" + (1'-8")/2) (Ref. 10.29.f)

= E2.27'

Distance between Centerlines of Unit 1 Containment and CEH = 59.37' (Ref. 10.5, page 26)

East-west Distance between Centerlines of Unit 1 Containment and TSC Air Intake

= E2.27' + W120.0' = 122.27'

East-west Distance between Unit 1 CEH and TSC Air Intake

= E2.27' + W120.0' + = 122.27'

East-west Distance between Centerlines of Unit 1 CEH and TSC Air Intake

East-west Distance between Centerlines of Unit 1 Containment and TSC Air Intake + Distance between Centerlines of Unit 1 Containment and CEH

= 122.27' + 59.37' = 181.64'

Slant Distance between Centerlines of Unit 1 Containment (Plant Vent) and TSC Air Intake

= $[(254.43)^2 + (122.27)^2]^{1/2}$ = 282.28' = 86.06 m

Slant Distance between Centerlines of Unit 1 CEH and TSC Air Intake

= $[(204.61)^2 + (181.64)^2]^{1/2}$ = 273.60' = 83.42 m

The distance between the source locations (Plant Vent and CEH) and receptor locations (Unit 1 CR and Unit 1 & 2 TSC) are compared in the following table:

Comparison of Distance Between Source & Receptor
Control Room Intake Vs Technical Support Center Intake

Slant Distance Between Source and Receptor			
Unit 1 Plant Vent and Unit 1 CR Intake (Meter)	**Unit 1 Plant Vent and TSC Intake (Meter)**	**Unit 1 CEH and Unit 1 CR Intake (Meter)**	**Unit 1 CEH and TSC Intake (Meter)**
30.25	86.06	46.62	83.42

7.7 CREACS Charcoal/HEPA Filter Efficiencies

Charcoal Filter

In-place penetration testing acceptance criteria for the safety related Charcoal filters are as follows:

CREACS Charcoal Filter – in-laboratory testing methyl iodide penetration < 2.5% (Ref. 10.6.11)

GL 99-02 (Ref 10.30) requires a safety factor of at least 2 should be used to determine the filter efficiencies to be credited in the design basis accident.

Testing methyl iodide penetration (%) = (100% - η)/safety factor = (100% - η)/2

Where η = charcoal filter efficiency to be credited in the analysis

CREACS Charcoal Filter

$2.5\% = (100\% - \eta)/2$

$5\% = (100\% - \eta)$

$\eta = 100\% - 5\% = 95\%$

HEPA Filter

HEPA filter efficiency = 99% (Ref. 10.6.14). HEPA filter efficiency of 95% is used in the analysis

Safety Grade Filter	Filter Efficiency Credited (%)		
	Aerosol	Elemental	Organic
CREACS	95	95	95

8.0 RESULTS SUMMARY

8.1 The EAB, LPZ, & CR doses due to a FHA occurring in the containment building with the CEH, personnel air locks, and containment penetrations open are summarized in the following table for different fuel decay times:

Fuel Decay Time (hr) Computer Run Number	Fuel Handling Accident Occurring In Containment Building TEDE Dose (rem)		
	Receptor Location		
	Control Room	EAB	LPZ
24 S24FHA150.o0	1.13	1.26	0.18
48 S48FHA150.o0	0.95	1.05	0.15
60 S60FHA150.o0	0.89	0.99	0.14
72 S72FHA150.o0	0.84	0.93	0.13
96 S96FHA150.o0	0.76	0.84	0.12
Allowable TEDE Limits	5.0	6.3	6.3

Significant assumptions used in this analysis:

- CEH, personnel air locks, and other containment penetrations remain open for the duration of the accident
- Containment integrity is not credited in the analysis
- Gap fission product fractions doubled
- Activity is released to the environment at a rate of 100,600 cfm
- CR envelope is pressurized with actuation of the CREACS following a FHA
- CR monitors' preferential alignment to less contaminated CR intake is not credited
- Worst χ/Qs are used for entire duration of the accident
- CR unfiltered inleakage of 150 cfm is assumed
- All fuel rods in one spent fuel assembly are damaged
- Reactor cavity overall effective DF = 200
- Core thermal power = 3,632 MW_t
- Radial Peaking Factor = 1.70

8.2 The EAB, LPZ, & CR doses due to a FHA occurring in the fuel handling building with a failure of an exhaust fan, are summarized in the following table for different fuel decay times:

Fuel Decay Time (hr) Computer Run Number	Fuel Handling Accident Occurring In Fuel Handling Building TEDE Dose (rem)		
	Receptor Location		
	Control Room	EAB	LPZ
24 FB24FHA150.o0	0.73	1.26	0.18
48 FB48FHA150.o0	0.62	1.05	0.15
60 FB60FHA150.o0	0.58	0.99	0.14
72 FB72FHA150.o0	0.55	0.93	0.13
96 FB96FHA150.o0	0.49	0.84	0.12
Allowable TEDE Limits	5.00	6.3	6.3

Significant assumptions used in this analysis:

- FHB charcoal filtration is not credited
- Gap fission product fractions doubled
- Activity is released to the environment at a rate of 21,439 cfm
- CR envelope is pressurized with actuation of the CREACS following a FHA
- CR monitors' preferential alignment to less contaminated CR intake is not credited
- Worst χ/Qs are used for entire duration of the accident
- CR unfiltered inleakage of 150 cfm is assumed
- All fuel rods in one spent fuel assembly are damaged
- Spent fuel pool overall effective DF = 200
- Core thermal power = 3,632 MW_t
- Radial Peaking Factor = 1.70

8.3 The EAB, LPZ, & CR doses due to a FHA occurring in the fuel handling building with a rapid release of one volume per minute are summarized in the following table for different fuel decay times:

Fuel Decay Time (hr) Computer Run Number	Fuel Handling Accident Occurring In Fuel Handling Building TEDE Dose (rem)		
	Receptor Location		
	Control Room	EAB	LPZ
24 FB24PUFF150.o0	2.06	1.27	0.18
48 FB48PUFF150.o0	1.78	1.06	0.15
60 FB60PUFF150.o0	1.67	1.00	0.14
72 FB72PUFF150.o0	1.58	0.94	0.13
96 FB96PUFF150.o0	1.43	0.85	0.12
Allowable TEDE Limits	5.00	6.3	6.3

Significant assumptions used in this analysis:

- FHB charcoal filtration is not credited
- Post-FHA activity is released to the environment at a rate of one volume/minute (558,550 cfm)
- Gap fission product fractions doubled
- CR envelope is pressurized with actuation of the CREACS following a FHA
- CR monitors' preferential alignment to less contaminated CR intake is not credited
- Worst χ/Qs are used for entire duration of the accident
- CR unfiltered inleakage of 150 cfm is assumed
- All fuel rods in one spent fuel assembly are damaged
- Spent fuel pool overall effective DF = 200
- Core thermal power = 3,632 MW_t
- Radial Peaking Factor = 1.70

9.0 CONCLUSIONS

9.1 FHA Occurring In Containment

The Section 8.1 results indicate that the EAB, LPZ, and CR doses are within allowable limits for a FHA occurring in the Containment building without containment integrity (with the CEH, personnel locks, and containment penetrations in the piping penetration areas opened) with a minimum fuel decay time of 24 hours. The results demonstrate that the following Salem 1 & 2 Technical Specification requirements can be relaxed:

1. The irradiated fuel can be moved in the reactor pressure vessel after the reactor has been sub-critical for at least 24 hours (relaxation to Technical Specification LCO 3.9.3)
2. Irradiated fuel assemblies can be moved without containment integrity (relaxation to Technical Specification LCO 3.9.4)
3. Core alterations can be performed without containment integrity (relaxation to Technical Specification LCO 3.9.4)

9.2 FHA Occurring In Fuel Handling Building

The Sections 8.2 and 8.3 results indicate that the EAB, LPZ, and CR doses are within allowable limits for a FHA occurring in the fuel handling building without crediting the charcoal filtration in the fuel handling ventilation system with a minimum fuel decay time of 24 hours. The results demonstrate that the Salem 1 & 2 Technical Specification Surveillance requirements 4.9.12.b and 4.9.12.c can be relaxed.

10.0 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: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," NUREG/CR-6604, USNRC, April 1998

3. Westinghouse Calculation No. CN-CRA-93-144, Rev 0, Salem LOCA Dose Analysis

4. 10 CFR 50.67, "Accident Source Term."

5. Calculation No. S-C-ZZ-MDC-1912, Rev 0, Control Room χ/Qs Using ARCON96 Code – Equipment Hatch & Plant Vent Releases

6. SNGS Technical Specifications:

6.1 Specification 3.9.4, Containment Building Penetrations

6.2 Specification 3.9.3, Decay Time

6.3 Specification 3.9.10, Water Level – Reactor Vessel

6.4 Specification 1.25, Rated Thermal Power

6.5 Specification 5.3.1, Fuel Assemblies

6.6 Specification 3.3.3.1, Radiation Monitoring Instrumentation LCO

6.7 Table 3.3-6, Radiation Monitoring Instrumentation

6.8 Specification Surveillance Requirement 4.8.1.1.2, Each diesel generator shall be demonstrated to be operable

6.9 Specification 3.9.11, Storage Pool Water Level

6.10 Specification 3.9.12, Fuel Handling Area Ventilation System

6.11 Specification Surveillance Requirement 4.7.6.1.b.3 and 4.7.6.1.c, CREACS Methyl Iodide Penetration

6.12 Specification Surveillance Requirement 4.7.6.1.d.1, CREACS Ventilation Flow Rate

6.13 Specification Surveillance Requirement 4.7.6.1.d.3, CREACS Design Makeup Flow Rate

6.14 Specification Surveillance Requirement 4.7.6.1.e, HEPA Filter DOP

6.15 Specification 3.7.6.1, Control Room Emergency Air Conditioning System (CREACS)

7. Federal Guidance Report 11, EPA-5201/1-88-020, Environmental Protection Agency

8. Federal Guidance Report 12, EPA-402- R-93-081, Environmental Protection Agency

9. Calculation No. S-C-ZZ-MDC-1959, Rev 0, CR χ/Qs Using ARCON96 Code – Non-LOCA Releases.

10. Design Change Package (DCP) No. 1EC-3505, CP Rev 2, Package No. 3, Control Area Ventilation – Radiation Monitoring Mod

11. Specification 5.2.1, Salem Unit 1/Unit 2 Containment Configuration

12. CD P534 of Design Change Package (DCP) No. 1EC-3505, Rev 7, Package No. 1, Control Area Air Conditioning System Upgrade
13. SNGS Calculation No. SC-RM005-01, Rev 2, R1B Radiation Monitors
14. Vendor Technical Document No. 322265-4, Rev 2, Fuel Handling Accident In Containment (Non-Design Basis)
15. Not Used.
16. Vendor Technical Document No. 321035, Rev 3, Accident X/Q Values At the Salem Generating Station Control Room Fresh Air Intakes, Exclusion Area Boundary And Low Population Zone
17. SNGS Architectural Drawings:
    a. 207069, Rev 12, Unit 1 Reactor Containment Floor Plan EL 130'-0"
    b. 207070, Rev 14, Unit 2 Reactor Containment Floor Plan EL 130'-0"
    c. 207080, Rev 23, Unit 1 Auxiliary Building Floor Plan EL 100'-0"
    d. 207081, Rev 29, Unit 2 Auxiliary Building Floor Plan EL 100'-0"
    e. 207084, Rev 13, Unit 1 Auxiliary Building Roof Plan EL 140'-0" & 141'-0"
    f. 207085, Rev 10, Unit 2 Auxiliary Building Roof Plan EL 140'-0" & 141'-0"
    g. 204803, Rev 10, Auxiliary Building EL 122', Reactor Cont & Fuel Building Area EL 130'
18. SNGS Mechanical P&IDs:
    a. 205321, Rev 21, Sheet 1 of 3, Unit 1 – Auxiliary Building Diesel Generator & Fuel Handling Area Ventilation
    b. 205237, Rev 42, Sheet 1 of 3, Unit 1 – Auxiliary Building - Ventilation
    c. 205237, Rev 30, Sheet 2, Unit 1 – Auxiliary Building - Ventilation
    d. 205322, Rev 23, Sheet 1 of 3, Unit 2 – Auxiliary Building Diesel Generator & Fuel Handling Area Ventilation
    e. 205337, Rev 36, Sheet 1 of 3, Unit 2 – Auxiliary Building - Ventilation
    f. 205337, Rev 22, Sheet 2, Unit 2 – Auxiliary Building - Ventilation
    g. 205238, Rev 33, Sheet 2, Reactor Containment – Ventilation
    h. 205338, Rev 27, Sheet 2, Reactor Containment – Ventilation
19. Core Operating Limits Reports for Salem 1 & 2:
    a. NFS-0190, Rev 0, Cycle 15, February 20001
    b. NFS-0209, Rev 0, Cycle 13, January 2002
20. SNGS Mechanical P&IDs:
    a. 205248, Rev 43, Sheet 2, Unit 1 Aux Bldg Control Area Air Conditioning & Ventilation
    b. 205348, Rev 34, Sheet 2, Unit 2 Aux Bldg Control Area Air Conditioning & Ventilation

21. Memorandum From Paul Wood To John Duffy, dated 12/15/96, Subject: Estimate of Unfiltered Inleakage from FHB with One Exhaust Fan and the Supply Fan Operating (Attached)

22. SNGS Wiring Diagram No. 220813, Rev 22, No. 2 Unit–Control Area No. 2 B 115 V AC Vital Instrument Bus

23. SNGS General Arrangement Drawings:

    a. 204805, Rev 5, Aux Bldg El. 84', Reactor Cont. 78' & 81', Fuel Handling Area El. 85'& 89'-6"

    b. 204808, Rev 1, Auxiliary Building & Reactor Containment Section A-A

24. Not Used.

25. Not Used.

26. SNGS Mechanical P&IDs:

    a. 602513, Sheet 1 of 3, Rev 0, No. 1 & 2 Units Technical Support Center - Ventilation

    b. 602513, Sheet 2 of 3, Rev 0, No. 1 & 2 Units Technical Support Center - Ventilation

27. SNGS Mechanical Arrangement Drawings:

    a. 602511, Rev 0, Clean Facilities Bldg, – Technical Support Center/Computer Room HVAC Systems El. 132'-6"

    b. 602512, Rev 0, Clean Facilities Bldg – Technical Support Center HVAC Equipment Room – Elevation 147'-4/12"

    c. 602514, Rev 0, Clean Facilities Bldg – Technical Support Center Technical Document & Annex Room HVAC Systems EL 119'-0"

28. SNGS Concrete Structural Drawing No. 242914, Rev 3, Clean Facilities Building Foundation Plan

29. SNGS Architectural Drawing No. 245685, Rev 2, Clean Facilities Bldg, Technical Support Center Floor, Roof Plans & Sections

30. USNRC, "Laboratory Testing of Nuclear-Grade Activated Charcoal", NRC Generic Letter 99-02, June 3, 1999

31. NRC Safety Evaluation for Calvert Cliffs Nuclear Power Plant Unit Nos. 1 and 2, Docket Nos. 50-317 and 50-318, License Amendment Nos. 242 and 216, dated March 12, 2001

32. Vendor Technical Document No. 326043, Control Room Envelope Inleakage Testing At Salem Nuclear Generating Station 2003.

33. Critical Software Package Identification No. A-0-ZZ-MCS-0225, Rev.2, RADTRAD Computer Code, Version 3.02

11.0 TABLES

Table 1
Salem 1 & 2 Noble Gas & Iodine Normalized Core Inventory

Isotope	Core Inventory At 3600 MWt (Ci) A	Core Power Normalizing Factor B	Normalized Core Inventory (Ci) C=AxB
KR-83M	1.200E+07	1.009	1.211E+07
KR-85	1.100E+06	1.009	1.110E+06
KR-85M	2.600E+07	1.009	2.623E+07
KR-87	4.700E+07	1.009	4.742E+07
KR-88	6.700E+07	1.009	6.760E+07
I-131	9.900E+07	1.009	9.988E+07
I-132	1.400E+08	1.009	1.412E+08
I-133	2.000E+08	1.009	2.018E+08
I-134	2.200E+08	1.009	2.220E+08
I-135	1.900E+08	1.009	1.917E+08
XE-131M	7.000E+05	1.009	7.062E+05
XE-133M	2.900E+07	1.009	2.926E+07
XE-133	2.000E+08	1.009	2.018E+08
XE-135	5.000E+07	1.009	5.044E+07
XE-135M	4.000E+07	1.009	4.036E+07
XE-138	1.600E+08	1.009	1.614E+08

A From Reference 10.3, Table 2

B = (3459 MWt x 1.05)/3600 MWt = (3632/3600) = 1.009

Table 2
Normalized Core Inventory Used In FHA Analysis

Isotope	Normalized Core Inventory (Ci) A	Gap Release Fraction IN RFT File B	Gap Release Fraction Used In Analysis C	Normalized Core Inventory Used In FHA D=(A*C)/B
KR-83M	1.211E+07	0.05	0.10	2.421E+07
KR-85	1.110E+06	0.05	0.20	4.439E+06
KR-85M	2.623E+07	0.05	0.10	5.246E+07
KR-87	4.742E+07	0.05	0.10	9.484E+07
KR-88	6.760E+07	0.05	0.10	1.352E+08
I-131	9.988E+07	0.05	0.16	3.196E+08
I-132	1.412E+08	0.05	0.10	2.825E+08
I-133	2.018E+08	0.05	0.10	4.036E+08
I-134	2.220E+08	0.05	0.10	4.439E+08
I-135	1.917E+08	0.05	0.10	3.834E+08
XE-131M	7.062E+05	0.05	0.10	1.412E+06
XE-133M	2.926E+07	0.05	0.10	5.852E+07
XE-133	2.018E+08	0.05	0.10	4.036E+08
XE-135	5.044E+07	0.05	0.10	1.009E+08
XE-135M	4.036E+07	0.05	0.10	8.071E+07
XE-138	1.614E+08	0.05	0.10	3.228E+08

A From Table 1
C From Design Input 5.3.1.3

Table 3
Post-FHA Activity Released In Containment Building Used In RADTRAD Nuclide Inventory File

Isotope	Core Initial Inventory (Ci) A	Radial Peaking Factor B	Total Number of Fuel Assembly In Core C	Number of Fuel Assembly Damaged D	Activity In Damaged Fuel Rods (Ci) E=A*B*D/C	DF F	Post-FHA Activity In RB Bldg For RADTRAD Code Nuclide Inventory File (Ci) G=E/F	(Ci/MWt) H=G/3632	RADTRAD (Ci/MWt) I=H*1
KR-83M	2.421E+07	1.70	193	1	2.133E+05	1.0	2.133E+05	5.872E+01	.5872E+02
KR-85	4.439E+06	1.70	193	1	3.910E+04	1.0	3.910E+04	1.077E+01	.1077E+02
KR-85M	5.246E+07	1.70	193	1	4.621E+05	1.0	4.621E+05	1.272E+02	.1272E+03
KR-87	9.484E+07	1.70	193	1	8.353E+05	1.0	8.353E+05	2.300E+02	.2300E+03
KR-88	1.352E+08	1.70	193	1	1.191E+06	1.0	1.191E+06	3.279E+02	.3279E+03
I-131	3.196E+08	1.70	193	1	2.815E+06	200.0	1.408E+04	3.876E+00	.3876E+01
I-132	2.825E+08	1.70	193	1	2.488E+06	200.0	1.244E+04	3.425E+00	.3425E+01
I-133	4.036E+08	1.70	193	1	3.555E+06	200.0	1.777E+04	4.893E+00	.4893E+01
I-134	4.439E+08	1.70	193	1	3.910E+06	200.0	1.955E+04	5.383E+00	.5383E+01
I-135	3.834E+08	1.70	193	1	3.377E+06	200.0	1.688E+04	4.649E+00	.4649E+01
XE-131M	1.412E+06	1.70	193	1	1.244E+04	1.0	1.244E+04	3.425E+00	.3425E+01
XE-133M	5.852E+07	1.70	193	1	5.154E+05	1.0	5.154E+05	1.419E+02	.1419E+03
XE-133	4.036E+08	1.70	193	1	3.555E+06	1.0	3.555E+06	9.787E+02	.9787E+03
XE-135	1.009E+08	1.70	193	1	8.887E+05	1.0	8.887E+05	2.447E+02	.2447E+03
XE-135M	8.071E+07	1.70	193	1	7.109E+05	1.0	7.109E+05	1.957E+02	.1957E+03
XE-138	3.228E+08	1.70	193	1	2.844E+06	1.0	2.844E+06	7.830E+02	.7830E+03

A From Table 2

12.0 FIGURES

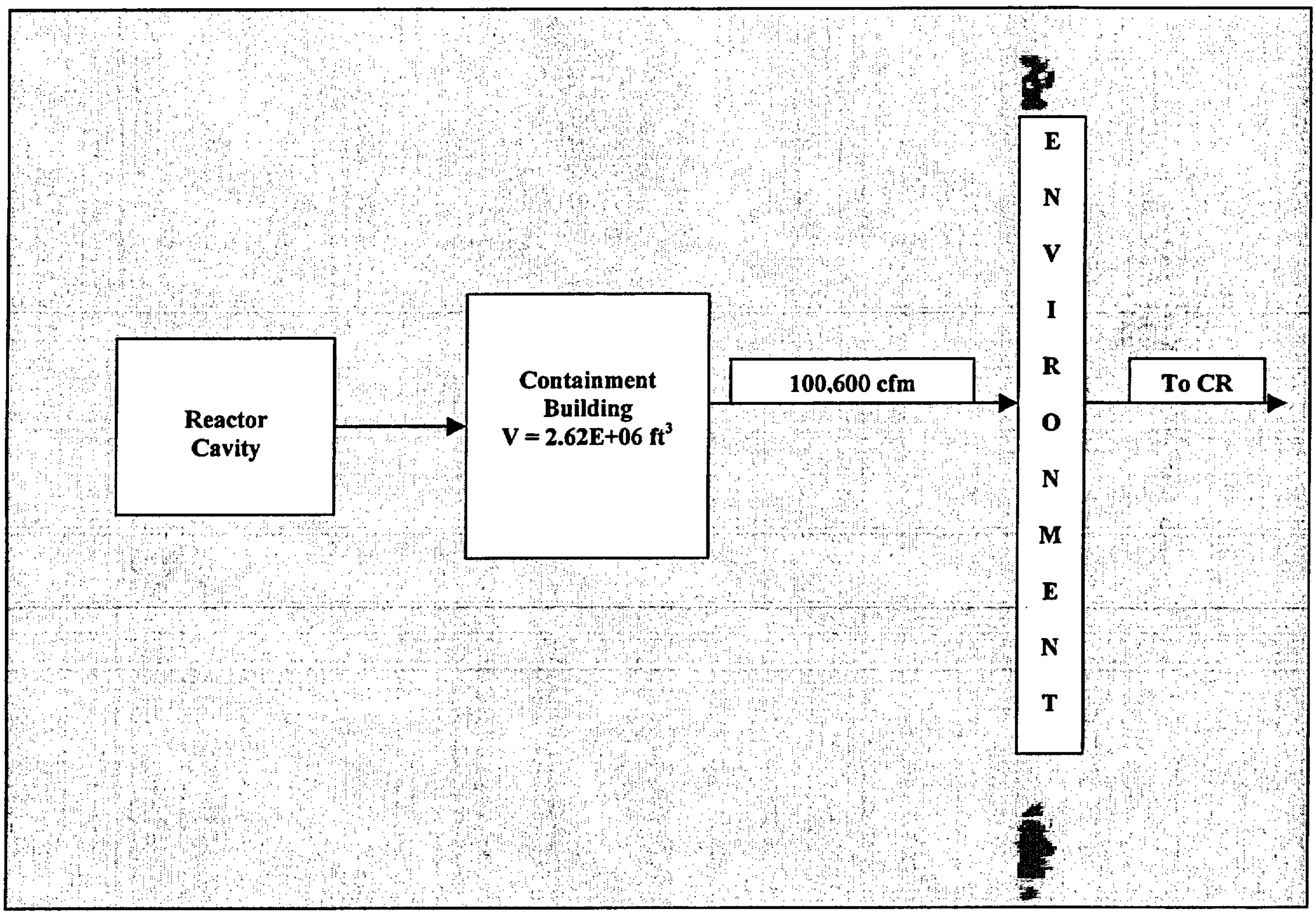


Figure 1: FHA In Containment Building With Equipment Hatch Open RADTRAD Nodalization

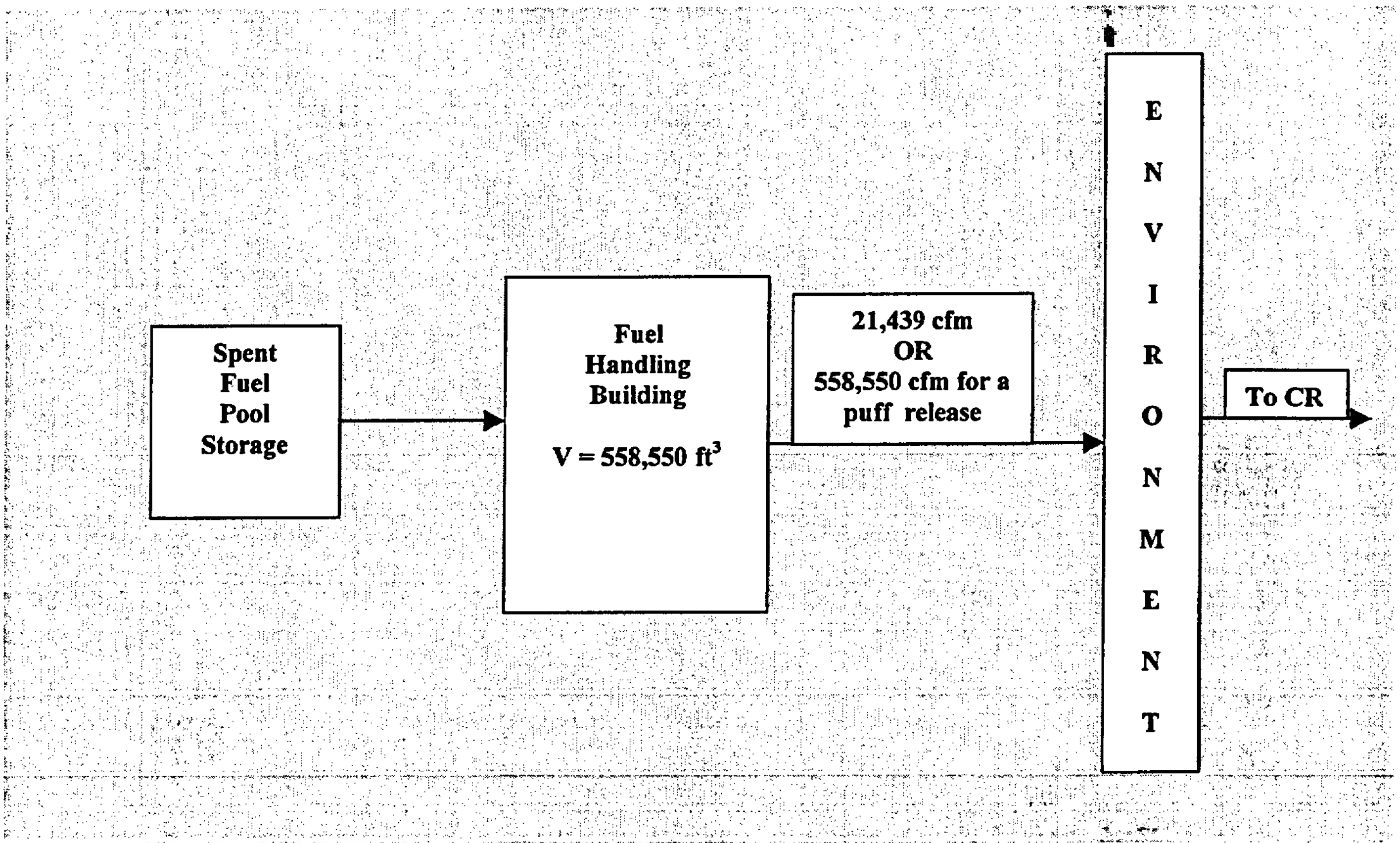


Figure 2: FHA Occurring In Fuel Handling Building RADTRAD Nodalization

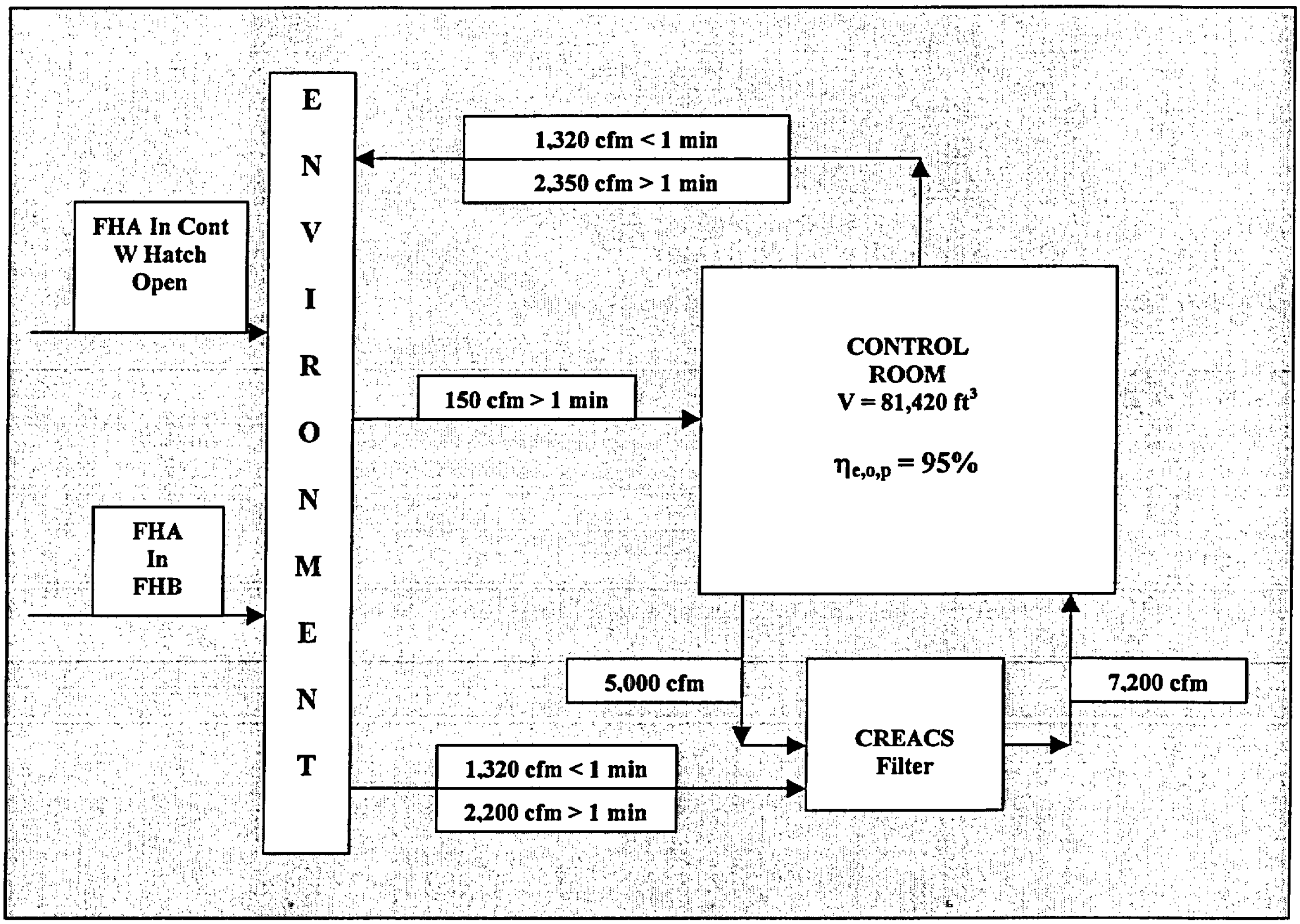


Figure 3: Salem Control Room RADTRAD Nodalization

Figure 4: Post-FHA CR TEDE Dose Vs Fuel Decay Time (hr) (CB)

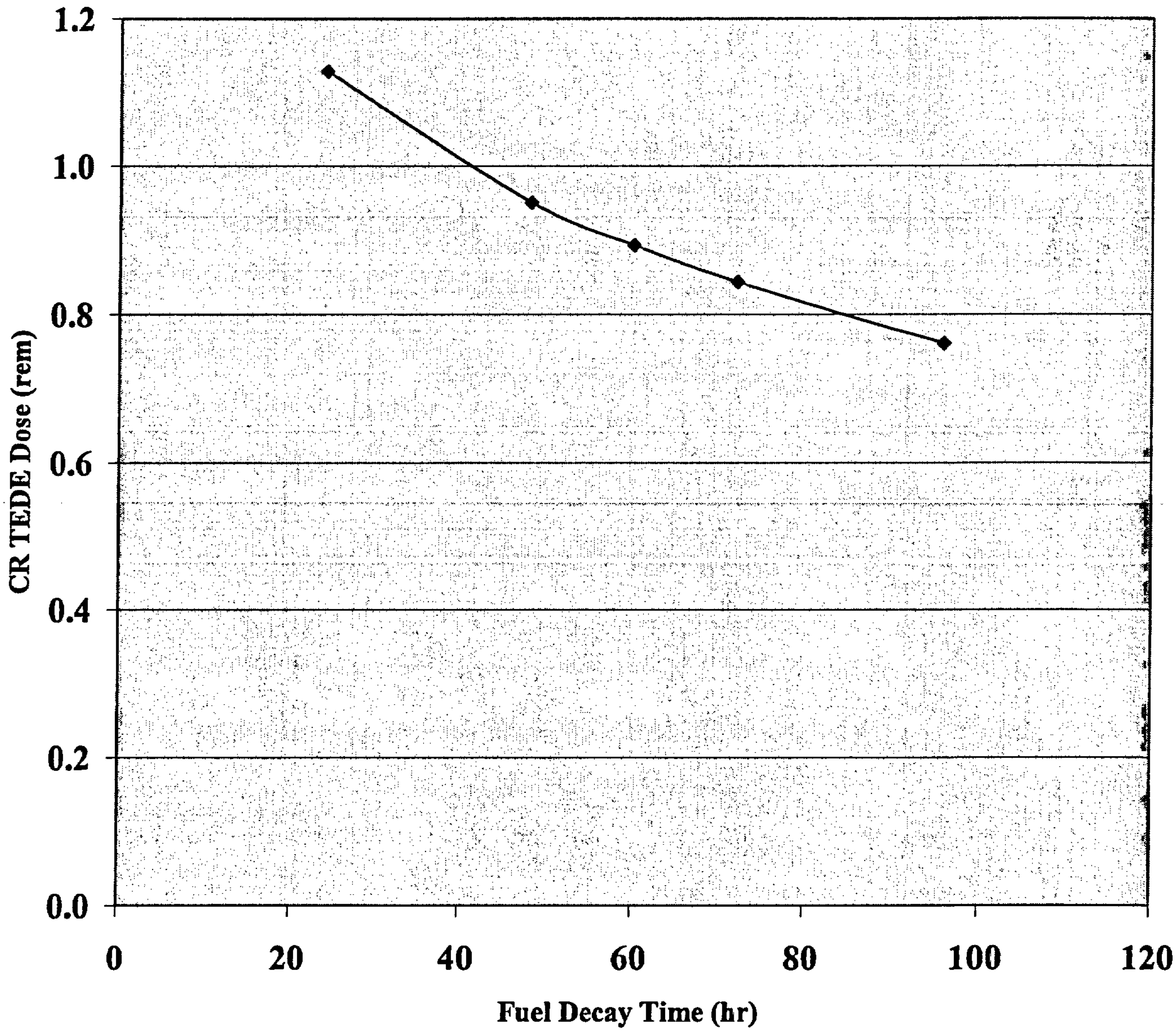
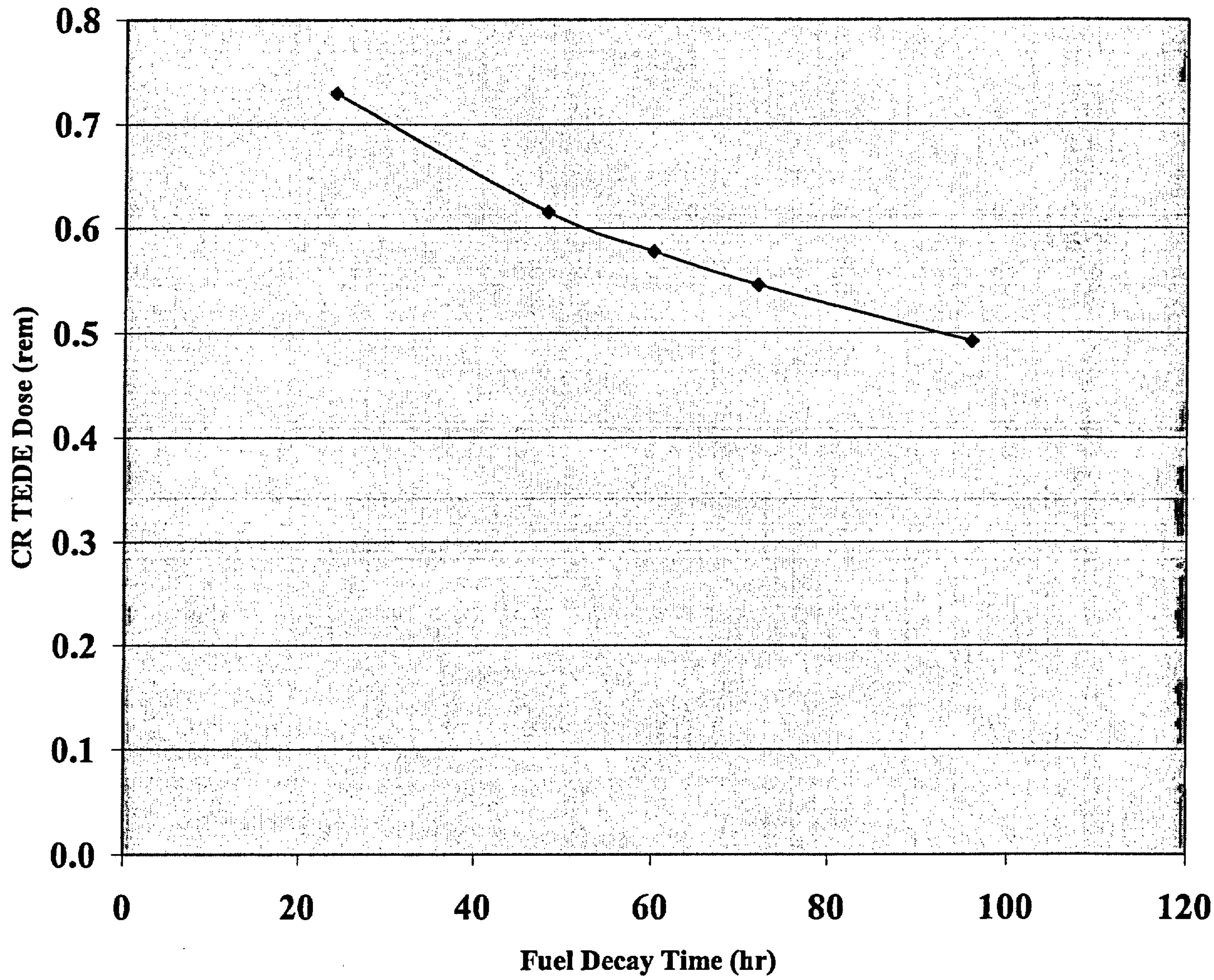


Figure 5: Post-FHA CR TEDE Dose Vs Fuel Decay Time (FHB)



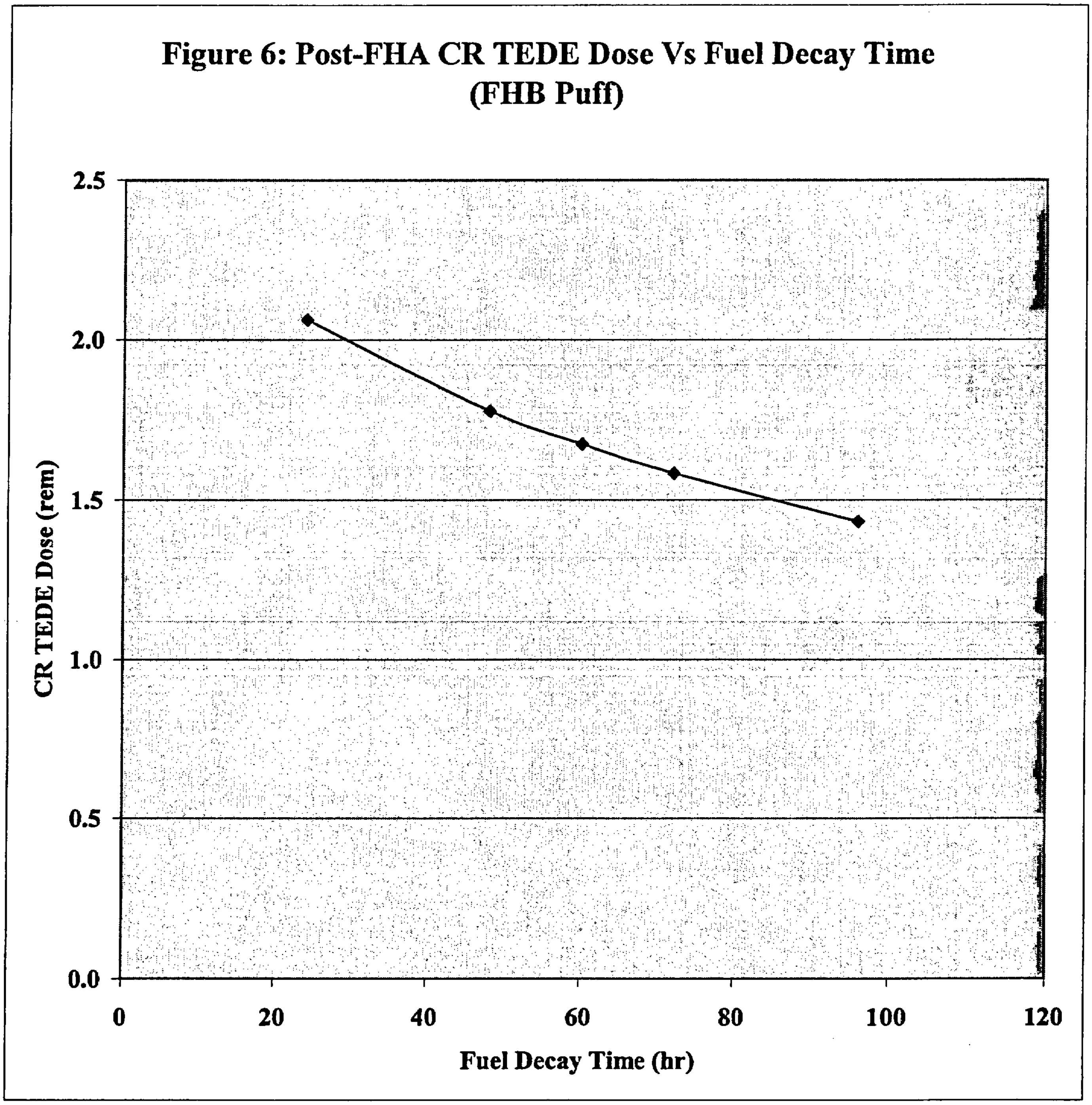


Figure 6: Post-FHA CR TEDE Dose Vs Fuel Decay Time
(FHB Puff)
CR TEDE Dose (rem)
0.0
0.5
1.0
1.5
2.0
2.5
0
20
40
60
80
100
120
Fuel Decay Time (hr)

13.0 ATTACHMENTS

13.1 CD containing the following electronic files

Design Calculation S-C-ZZ-MDC-1920, Rev 4

Nuclide Inventory File SNGSFHA_def.txt

Nuclide Release Fraction & Timing File SNGSFHA_rft.txt

FGR Dose Conversion File SALEMFHA_FG11&12.txt

RADTRAD Input and Output Files for FHA Inside Containment:

S24FHA150.psf and S24FHA150.o0

S48FHA150.psf and S48FHA150.o0

S60FHA150.psf and S60FHA150.o0

S72FHA150.psf and S72FHA150.o0

S96FHA150.psf and S96FHA150.o0

RADTRAD Input and Output Files for FHA Inside Fuel Handling Building:

FB24FHA150.psf and FB24FHA150.o0

FB48FHA150.psf and FB48FHA150.o0

FB60FHA150.psf and FB60FHA150.o0

FB72FHA150.psf and FB72FHA150.o0

FB96FHA150.psf and FB96FHA150.o0

RADTRAD Input and Output Files for FHA Inside Fuel Handling Building (Puff Release):

FB24PUFF150.psf and FB24PUFF150.o0

FB48PUFF150.psf and FB48PUFF150.o0

FB60PUFF150.psf and FB60PUFF150.o0

FB72PUFF150.psf and FB72PUFF150.o0

FB96PUFF150.psf and FB96PUFF150.o0

13.2 Copy of Reference 10.21 (2 pages)

14.0 AFFECTED DOCUMENTS

S-C-ZZ-MDC-1920, Revision 3 will be superseded.

CD With Various Electronic Files

MEMORANDUM From the Desk of Paul Woods

Paul Woods

TO: John Duffy

DATE: 12/16/96

SUBJECT: Estimate of Unfiltered Leakage from FHB with One Exhaust Fan and the Supply Fan Operating.

Initially, the FHV system will be in the normal alignment for fuel handling, 2-exhaust fans and the supply fan operating. Upon loss of one exhaust fan the building pressure controller will attempt to modulate open to maintain building negative pressure. Eventually the maximum travel stop will be reached and no further exhaust flow increase is possible. Building pressure will continue to increase due to the imbalance between the exhaust flow and the supply flow. when the building pressure reaches the alarm setpoint (approx. 0.16 inches of water negative, WRT the outside) the control room annunciator will alarm. Per the Alarm Response procedure the operator is directed to shut down the operating supply fan when the building pressure alarm is received. During the period when the operator is evaluating the alarm, and taking action, the fuel handling building may go positive.

Potential release points are the truck bay roll-up door on the west end of the Fuel Handling Building, leakage through the closed gravity damper that replaced the truck bay exhaust fan (shown on Dwg 207647 and located in the south wall at elevation 124'-6"; 8' west of the N-N grid location line), and leakage through the 2FHV8 supply air handling unit damper located in the north wall at the 100' elevation; approximately 10' east of grid location line R-R, also in the truck bay area (also shown on Dwg 207647).

During normal operation, the supply fan is set to approximately 2000 cfm less than the exhaust flow rate. The normal exhaust flow rate is approximately 19,490 cfm. Therefore the normal supply is around 17,500 cfm.

The FHV exhaust fans are identical fans operated in parallel, with back draft dampers on the exhaust to permit the full flow from one fan to be exhausted if the other fan is stopped. The pressure vs. flow response for this arrangement can be modeled by plotting the equivalent fan curve for both fans from the fan curve for a single fan by doubling the flow rate for each constant total pressure point. When one fan is lost, the result is that the system resistance curve is followed down to the intersection of the system curve with the single fan curve. The 1 and 2 fan curves can be plotted using the vendor supplied fan curve. The equation of the system curve is given by $\Delta P = RV^2$. Where ΔP is the fan total pressure, R is a constant, and V is the volumetric flow. The operating point on the 2-fan curve is 19,490 cfm. From the combined 2 Fan curve this corresponds to 8.2 in W.G. total pressure. Solving for R

$R = \Delta P / V^2$
$R = 8.2/(19490)^2$
$R = 2.16(10)^{-8}$ in W.G./cfm

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To S. Ferguson	Date 12/17/96 · # of pages ▶ 9
Co./Dept. SWEC	From J. Duffy
Phone # (617) 589-8033	Co. PSE&G
Fax # (617) 589-1315	Phone # (609) 339-1022
	Fax # (609) 339-1218

MEMORANDUM From the Desk of Paul Woods

This constant is then used to plot the system curve, which intersects the 1-fan curve at 14,500 cfm (refer to the attached curves).

Therefore, the flow through the normal exhaust path (filtered) is 14,500 cfm. A conservative assumption is that the supply fan continues to operate at 17,500 cfm with 14,500 cfm exhausted to the stack. and the remainder or 3,000 cfm leaking unfiltered to the environment. The methodology for this type of evaluation is provided in Section 19.4 of *Handbook of Air Conditioning and Refrigeration* by Shan K. Wang; McGraw-Hill, 1993.

The gravity damper identified above is Class II leakage per ASME AG-1 (Refer to PSBP 317245). Class II dampers are rated for 8 cfm per sq. ft of face are at 1 in WG. For the 48" x 48" gravity relief damper, this is equal to 128 cfm @ 1 in. The maximum pressure from the supply fan is approximately 3 in W.G. therefore to be conservative the leakage from the damper is estimated to be 2 times 128 or 256 cfm.

Using a similar method to the above, the +10% and -10% system curves were plotted, and the unfiltered leakage for each case was evaluated, the results are presented below.

	Design Airflow	Design +10%	Design -10%
2-Exhaust Fans	19,490	21,439	17,541
Supply Fan	17,490	19,439	15,541
1-Exhaust Fan	14,500	15,300	13,600
Delta (Supply - single exhaust)	2,990	4,139	1,941
Leakage at Truck Bay	2,734	3,883	1,685
Leakage Through Gravity Damper	256	256	256

There are several conservatisms built into the above estimate. First, The estimated system curve ignores the effect of the supply fan and only looks at the effect of the two exhaust fans in parallel. The pressurization effect of the supply fan will tend to push more air through the exhaust filters. Second calculating the leakage through the gravity damper ignores the manual damper recently placed in series with this damper. The manual damper should be closed during fuel handling and will provide additional resistance to leakage at this location. Third, the leakage through the gravity damper is assumed to double at a pressure of 3 in W.G. although this is square root relationship, and the leakage would actually only increase by a factor of 1.7. Fourth, the building pressure is assumed to be 3 in W.G. positive, i.e. the maximum attainable by the supply fan, in fact the building pressure will be less than this maximum, and may be nearly neutral.

Reviewed By: [signature] Date: 12/6/96

FORM-1

CERTIFICATION FOR DESIGN VERIFICATION

Reference No. S-C-ZZ-MDC-1920, Rev. 4IR0

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 S-C-ZZ-MDC-1920, Rev 3 to perform a sensitivity study to determine the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room (CR) doses for various fuel decay times for the FHA occurring in the reactor building and fuel handling building. The analysis is revised to calculate doses at various decay times in support of an anticipated license change request LCR No. S06-07 for a Technical Specification change.

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.

Alan Johnson	Mark F Drucker Mark Drucker 5-18-2006
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.

1 of 6

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

REFERENCE DOCUMENT NO. /REV. S-C-ZZ-MDC-1920, Revision 4IR0

	COMMENTS	RESOLUTION	ACCEPTANCE OF RESOLUTION
1	**General** Editorial Comments are being provided separately in the form of a redline/strikeout mark-up. The Originator may determine which editorial comments should be incorporated.	Incorporated.	MD 5-18-06
2	**Section 3.0** It is recommended that an introductory paragraph be added to explicitly state that this analysis uses Version 3.02 of the RADTRAD computer code (Ref. 10.2) to calculate the potential radiological consequences of an FHA. The RADTRAD code is documented in NUREG/CR-6604 (Ref. 10.2). The RADTRAD code is maintained as Software ID Number A-0-ZZ-MCS-0225 (Ref. 10.33).	Incorporated.	MD 5-18-06
3	**Section 3.2** The text erroneously states that the results of the parametric study shown in Sections 8.2 & 8.3 indicate that a release over a two-hour period yields a higher CR dose due to a larger amount of activity entering the CR volume. In fact, the results of the parametric study indicate that a release based on the rapid release rate of one FHB volume per minute yields a higher CR dose. The puff release yields a higher CR dose because it results in a larger amount of unfiltered iodine activity entering the CR volume prior to the one minute start of the CREACS outside air inflow filtration.	Typo corrected.	MD 5-18-06
4	**Section 3.2** The text discussion of the Reference 10.9 ARCON95 analysis of the smoke hatch is not necessary. Reference 10.9 used the ARCON96 code. This should also allow for the deletion of Reference 10.24 from Section 10.	Information deleted.	MD 5-18-06
5	**Design Input 5.3.4** Please add a new design input section (5.3.4.14) to document the χ/Qs for the smoke hatch release taken from Reference 10.9 page 42.	Incorporated.	MD 5-18-06

2

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

REFERENCE DOCUMENT NO. /REV. S-C-ZZ-MDC-1920, Revision 4IR0

	COMMENTS	RESOLUTION	ACCEPTANCE OF RESOLUTION
6	**Design Inputs 5.3.5.1 & 5.3.5.2 and Ref. 10.9** Design Inputs 5.3.5.1 and 5.3.5.2 present the EAB and LPZ X/Q values. In previous calculation revisions the data was taken from Reference 10.9 (which was probably Vendor Technical Document No. 321035, Rev. 3). The current analysis has replaced Reference 10.9 with Calculation SC-ZZ-MDC-1959 which provides the smoke hatch X/Q values, but which does not provide the offsite X/Q values. Please add a reference for offsite X/Q values.	New Reference 10.16 added.	MD 5-18-06
7	**Section 7.2.1, Section 5.3.2.9, and FHA Inside Containment RADTRAD runs** The containment building release rate of 99,800 cfm modeled in the RADTRAD runs is calculated in Section 7.2.1 for a containment volume of 2.6E6 cf. Design Input 5.3.2.2 revised this volume to 2.62E6 cf. When recalculated, the release rate will increase to approximately 100,600 cfm.	New release rate is 100,600 cfm used in the analysis and the RADTRAD runs for the FH occurring in the containment building are revised.	MD 5-18-06
8	**Section 7.4 and Design Input 5.3.5.5** The Section 7.4 calculations model a CR intake monitor Xe-133 sensitivity of 6.0E7 rather than the 6.2E7 value shown in Design Input 5.3.5.5 (which cites Ref. 10.13 page 12). Please revise as necessary to ensure consistency with Reference 10.13.	Information is made consistent.	MD 5-18-06
9	**Section 7.5 and Design Input 5.3.4** Section 7.5 models a 0-2 hr smoke hatch X/Q of 1.14E-2. Per Reference 10.9 Section 8.4 the maximum X/Q is 1.15E-2 for the U1 smoke hatch to U1 CR intake path. Please revise Section 7.5. In addition, please add a new design input section (perhaps 5.3.4.14) to document the χ/Qs for the smoke hatch release taken from Reference 10.9.	Incorporated.	MD 5-18-06
10	**Sections 8.1 through 8.3** The doses are reported to the ten-thousandths rem (i.e., tenths of a millirem). Consider rounding the doses reported in the results with the same level of accuracy that they will be reported in the UFSAR.	Incorporated.	MD 5-18-06
11	**RADTRAD runs for FHA in containment** In each run, the CREACS recirculation filter is turned on at 2 minutes, instead of the 1 minute value specified in Section 3.4	Incorporated.	MD 5-18-06
12	**RADTRAD runs for FHA in FHB (2 hr release)** 1) In each run, the CREACS recirculation filter is turned on at 2 minutes, instead of the 1 minute value specified in Section 3.4 2) The FB24FHA24 output file name has an extra dot before its extension: "..o0"	Incorporated.	MD 5-18-06

COMMENT / RESOLUTION FORM
FOR DESIGN DOCUMENT
OWNER'S REVIEW

REFERENCE DOCUMENT NO. /REV. S-C-ZZ-MDC-1920, Rev. 4IR0

COMMENTS

1. Cover Sheet: The original plan was to retain our licensing basis analysis and add a sensitivity study of dose vs. decay time. However, the revised calculation eliminates the current analysis-of-record. The revision should be identified as interim rather than final.

2. Cover Sheet: The description of the revision indicates that the only parameter changed is decay time. The analysis is revised to calculate doses at various decay times in support of an anticipated submittal for a Technical Specification change. However, the other parameters that were also changed should be identified as well as why it was necessary to change other parameters.

3. General Comment: Much of the analysis is not changed. Revision bars should be used to identify the changes.

4. Design Input Parameter 5.3.4.7 unfiltered control room inleakage is changed from 4000 cfm to 150 cfm. The original plan was to retain the current analysis-of-record parameter values and add a sensitivity study of dose vs. decay time. The 4000 cfm value should be retained. Additionally, Table 1 in Reference 10.32 does not show a value of 150 cfm. 150 cfm bounds the nominal inleakage results except for the isolation mode. 4000 cfm bounds all the results even if uncertainty is included. The higher value should be retained.

5. Regulatory Change Process Determination: The original plan was to retain our licensing basis analysis and add a sensitivity study of dose vs. decay time. In that case the calculation revision would not be controlled by any of the processes identified. However, with the elimination of the current analysis-of-record, the calculation revision should be associated with a planned change to the Technical Specifications and an LCR should be identified and all aspects of the calculation revision would be controlled by the license change submittal. The RCPD should be revised to include an explanation that the calculation is revised to support the submittal.

6. 10CFR50.59 Screening: A screening is not required if all aspects of the calculation revision support the license change submittal.

J. Duffy	05/17/2006
SUBMITTED BY	DATE

4

COMMENT / RESOLUTION FORM
FOR DESIGN DOCUMENT
OWNER'S REVIEW

RESOLUTION

1. Incorporated.
2. The statement is revised.
3. Since the various sections are re-organized to standardize the calculation format, the entire calculation is considered revised.
4. The CR unfiltered inleakage licensing basis was established in the LOCA analysis, which is 150 cfm. The use of additional unfiltered inleakage makes the CR dose unnecessary conservative. The use of a very high CR unfiltered inleakage of 4,000 cfm was acceptable in absence of the tracer gas test result.
5. Incorporated.
6. 50.59 Screening is deleted.

Gopal J. Patel 05/17/2006
RESOLVED BY DATE

ACCEPTANCE OF RESOLUTION

J. Duffy

J. Duffy 05/18/2006
SUBMITTED BY DATE

5

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

REFERENCE DOCUMENT NO. /REV. S-C-ZZ-MDC-1920, Revision 4IR0

	COMMENTS	RESOLUTION	ACCEPTANCE OF RESOLUTION
13	**RADTRAD runs for FHA in FHB (puff)** 1) In each run, the CREACS recirculation filter is turned on at 2 minutes, instead of the 1 minute value specified in Section 3.4 2) In each run, the CREACS filtered and unfiltered inflow rates are initiated at 2 minutes, instead of the 1 minute value specified in Section 3.4 3) In each run, the CR unfiltered inleakage rate is modeled as 4000 cfm (not 150 cfm); and consequently the CR outflow rate is also high. 4) In each run, the CR X/Q is modeled as 1.78E-3 (not 1.85E-3). 5) The run titles state that a puff release rate of 350,000 cfm is modeled. The actual modeled puff release rate is 558,550 cfm.	Incorporated.	MD 5-18-06
END			

Mark F Drucker (MD) Mark Drucker 05/14/2006 SUBMITTED BY DATE	Gopal J. Patel 05/17/2006 RESOLVED BY DATE

FORM-1
REGULATORY CHANGE PROCESS DETERMINATION

Document I.D.:	**S-C-ZZ-MDC-1920**	**Revision:**	**4IR0**
Title:	**Fuel Handling Accidents Radiological Consequences**		

Activity Description:

Issuing the design calculation, which performs a sensitivity study to determine the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room (CR) doses for various fuel decay times for the FHA occurring in the reactor building and fuel handling building. The analysis is revised to calculate doses at various decay times in support of an anticipated license change request LCR No. S06-07 for a Technical Specification change.

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. S06-07**
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.:	S-C-ZZ-MDC-1920	Revision:	4IR0
Title:	Fuel Handling Accidents 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)

FORM-1
REGULATORY CHANGE PROCESS DETERMINATION

Document I.D.:	S-C-ZZ-MDC-1920	Revision:	4IR0
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Activities Affected	Yes	No	Action
11. Does the proposed activity involve a *minor* UFSAR change (including documents incorporated by reference)? <u>Examples</u>: • 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? <u>Examples</u>: • 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? <u>Examples</u>: • 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? <u>Examples</u>: • 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? <u>Examples</u>: • 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: ______________
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.

FORM-1
REGULATORY CHANGE PROCESS DETERMINATION

Document I.D.:	S-C-ZZ-MDC-1920	Revision:	4IR0
Title:	Fuel Handling Accidents Radiological Consequences		

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:

The analysis is revised to support a planned licensing change request LCR S06-07 to reduce the fuel decay time in the Salem 1 & 2 Technical Specification.

[signature]	05/18/2006	Gopal J. Patel	04/04/2007
PREPARER (SIGN)	DATE	NAME (PRINT)	QUAL EXPIRES
J. Duffy	5/18/06	John F. Duffy	10/31/2007
REVIEWER (SIGN)	DATE	NAME (PRINT)	QUAL EXPIRES