ADDITIONAL ATTACHMENTS TO
10-10-05 Letter: Supplement to Request for LAR Application of AST

Attachment 009 AST – LM-0645 Rev 1 FHA.

ATTACHMENT 1
Design Analysis Cover Sheet

Design Analysis (Major Revision)		**Last Page No.** [6] 18 / Att. G-3	
Analysis No.: [1]	LM-0645	**Revision:** [2]	1
Title: [3]	Re-analysis Of Fuel Handling Accident (FHA) Using Alternative Source Terms		
EC/ECR No.: [4]	04-00003	**Revision:** [5]	0

Station(s): [7]	Limerick	**Component(s):** [14]	
Unit No.: [8]	1 & 2	N/A	
Discipline: [9]	MEDC		
Descrip. Code/Keyword: [10]	H84 /AST, FHA		
Safety/QA Class: [11]	SR		
System Code: [12]	912		
Structure: [13]	N/A		

CONTROLLED DOCUMENT REFERENCES [15]

Document No.:	From/To	Document No.:	From/To
LGS UFSAR	From/To	Design Analysis LM-0312	From
Design Analysis LM-0641	From	Tech. Spec. 3/4.9.4	From
Calculation M-78-01	From	Dwg No.M-102, M-107	From
Dwg No. SIM-M-76, Sheet 2	From		

Is this Design Analysis Safeguards Information? [16] Yes ☐ No ☒ If yes, see SY-AA-101-106

Does this Design Analysis contain Unverified Assumptions? [17] Yes ☐ No ☒ If yes, ATI/AR#:

This Design Analysis SUPERCEDES: [18] LM-0645, Rev. 0 **In its entirety.**

Description of Revision (list affected pages for partials): [19]

This revision incorporates responses to pertinent NRC Requests for Additional Information (RAIs) with respect to all Exelon Nuclear Station Alternative Source Term License Amendment Applications The fuel damage assessment utilized was revised to agree with that currently in the UFSAR, except for use of a more conservative peaking factor to provide bounding results. Bounding results for other previously considered FHA scenarios are also included. Finally, additional assumptions from Regulatory Guide 1.183 are included to directly indicate conformance with this Regulatory Guide.

	Print Name	Sign Name	Date
Preparer: [20]	Harold Rothstein	Harold Rothstein	9/27/2005

Method of Review: [21] **Detailed Review** ☒ **Alternate Calculations (attached)** ☐ **Testing** ☐

	Print Name	Sign Name	Date
Reviewer: [22]	Paul Reichert	Paul Reichert	9/27/2005

Review Notes: [23] Hu AA/212 Independent review ☒ Peer review ☐
Third Party Review by R. Hess (Limerick), D. Strawson (MPR)

	Print Name	Sign Name	Date
(For External Analyses Only) **External Approver:** [24]	Harold Rothstein	Harold Rothstein	9/27/05
Exelon Reviewer: [25]	T. J. Mscisz	[signature]	9/27/05

Is a Supplemental Review Required? [26] Yes ☐ No ☒ based on 3rd Party reviews. If yes, complete Attachment 3

	Print Name	Sign Name	Date
Exelon Approver: [27]	E. Flick	Elliott Flick	9/27/05

Table of Contents

1. PURPOSE/OBJECTIVE

The purpose of this calculation is to apply Alternative Source Term (AST) methodology to the analysis of the Fuel Handling Accident (FHA) for Limerick Generating Station (LGS) Units 1 & 2. The calculation is based on normal Reactor Building unfiltered exhaust through the South Stack with no Control Room Emergency Filtration (CREF). Therefore, this calculation supports changes to the current LGS 1 & 2 Technical Specifications (TS) to consider that maintenance of the secondary containment integrity and the operability of emergency filtration systems and subsystems previously required to mitigate the radiological consequences of fuel handling accidents may not be necessary. Not having to consider secondary containment and control room integrity and filtration requirements in support of refueling activities has the potential to significantly improve the flexibility and duration of scheduled plant outage activities. Based on LGS Technical Specification 3/4.9.4 on Decay Time for movement of irradiated fuel in the reactor pressure vessel, in agreement with the discussion in UFSAR [Ref. 1] Sections 15.7.4.5 and especially 15.7.4.5.2.1, movement of irradiated fuel will not occur less than 24 hours after the associated reactor is subcritical, and therefore, a 24-hour delay period is used. This value continues to be a very conservative assumption for BWRs, given the operations necessary before commencing fuel movement.

A "recently irradiated fuel" parameter is considered as the point in time after shutdown when secondary containment integrity features are not required. Therefore, this calculation also justifies 24 hours as this time after shutdown.

Regulatory Guide 1.183 [Ref. 2] is the basis for these evaluations. Concerning the FHA, this AST guidance has the advantage of smaller gap fractions, a larger pool decontamination factor [DF], and dose criteria that replace both the whole body and thyroid dose limits with a limit on Total Effective Dose Equivalent [TEDE].

Other changes from the current UFSAR calculation are listed below.

- An offsite dose limit of 6.3 rem TEDE (Reg. Guide 1.183 & 10 CFR 50.67(b)(2)(iii)) is applied instead of the Standard Review Plan (SRP) 15.7.4 values of 25% of the 10CFR100 limits.
- A control room dose limit of 5 rem TEDE (10 CFR 50 Appendix B (II) GDC 19) is applied instead of the SRP 15.7.4 values of 5 rem whole body, or its equivalent.
- Analysis is based on NRC Regulatory Guide 1.183.
- Secondary Containment automatic isolation and filtration are not credited.
- The Control Room and offsite $^{\chi}/_{Q}$'s were recalculated for release from the Unit 1 & 2 Reactor Building (RB) South Stack as the worst case release location with respect to the Control Room intake for no Standby Gas Treatment System (SGTS) filtration, and the new limiting $^{\chi}/_{Q}$'s were applied to this analysis.
- CREF operation is not credited.
- Dose Conversion Factors (DCFs) for Immersion and Inhalation are taken from Federal Guidance Reports (FGRs) 12 [Ref. 5] and 11 [Ref. 4], respectively. Regulatory Guide 1.183 cites these DCFs as acceptable current estimates for evaluating the radiological impact of nuclear plant accidents.

This calculation also documents the development of core source terms to be used in this and other accident analyses that involve postulated fuel damage and are being reanalyzed using alternative source terms.

2. METHODOLOGY AND ACCEPTANCE CRITERIA

Analyses of radiological consequences resulting from a Fuel Handling Accident (FHA) are performed using the guidance for application of Alternative Source Terms to this event in Regulatory Guide (RG) 1.183.

Analyses of radiation transport and dose assessment are performed using RADTRAD v. 3.03. RADTRAD is a simplified model of RADionuclide Transport and Removal And Dose Estimation developed for the NRC and endorsed by the NRC as an acceptable methodology for reanalysis of the radiological consequences of design basis accidents. The technical basis for the RADTRAD code is documented in NUREG/CR-6604 [Ref. 3]. The methodologies significant to this analysis are the dose consequence analysis (NUREG Section 2.3) and the Radioactive Decay Calculations (NUREG Section 2.4). This version of RADTRAD has been pre-qualified for safety related design analysis by Washington Group International per its 10CFR50 Appendix B Quality Assurance program.

2.1. Fuel Source Term Model

The dose assessments in this calculation use the UFSAR (Ref. 1) historical fuel damage assumptions of a total of 212 failed rods based on an 8x8 fuel design containing 62 fuel rods, but with a conservatively higher radial peaking factor (PF) of 1.7 instead of 1.5. As per UFSAR Section 15.7.4, the analytical methodology and licensing bases for determination of fuel damage in a FHA are provided in GESTAR II, and compliance with these bases is verified for each new fuel design. The dose analysis in this calculation applies the UFSAR Section 15.7.4 additional conservative assumptions so that this calculation continues to provide margin.

The assumed accident, per UFSAR Section 15.7.4, is an assembly and mast drop from the maximum height allowed by the refueling platform (a height of 32 feet for the fuel assembly, and 47 feet for the mast) over the reactor well onto fuel in the reactor. Based on fuel damage assessments in Ref. 9 and as shown below, this bounds the damage assessments for various 8x8 and 7x7 array fuel types with 60 and 49 fuel pins per bundle, respectively, and 111 failed pins and a 1.5 PF, as well as GE11 or GE13 9x9 array fuel types with 74 fuel pins per bundle,140 failed pins and a 1.5 PF, and GE12 or GE14 10x10 array fuel types with 172 failed pins per bundle and a 1.7 PF.

Bundle Type	Fuel Array	Pins in Bundle	Failed Pins	Damaged Core Fraction[B]	PF	Damaged Core Fraction multiplied by PF
Previous UFSAR Basis with 1.7 PF **[New Dose Analysis Basis]**	8x8	62	212	0.004476	1.7	**0.007608**
Previous UFSAR Basis	8x8	62	212	0.004476	1.5	**0.006713**
Various	8x8	60	111	0.002421	1.5	**0.003632**
Various	7x7	49	111	0.002965	1.5	**0.004448**
GE11&GE13	9x9	74	140	0.002476	1.5	**0.003714**
GE12&GE14[A]	10x10	87.33	172	0.002578	1.7	**0.004382**

A. Bounding Assembly type in current use, with higher peaking factor commensurate with full core application for 10x10 fuel
B. Damaged Core Fraction = Failed Pines / Pins in Bundle / 764 bundles in core

The associated power of the UFSAR Basis damaged fuel = 3527 MWth * 0.007608 = 26.83 MWth.

For bounding dose assessment purposes, the fuel source term is based on the reactor core source terms described in Attachment A. These source terms are bounding for LGS fuel cycle designs as documented on the last two pages of Attachment A.

2.2. Gap Activity

This calculation is applicable to fuel whose burnup and power limits are bounded by those specified in RG 1.183, footnote 11. This allows application of the gap activity fractions for LOCA events per RG 1.183, Table 3, which are as follows:

5% of the noble gases (excluding Kr-85)
10% of the Kr-85
5% of the iodine inventory (excluding I-131)
8% of the I-131
12% of the Alkali metal inventory

Because RADTRAD does not allow for application of isotope specific release fractions, the "Limerick Generating Station AST Source Term.nif" file is modified to accommodate the differential gap activities among the halogen (I-131) and noble gas (Kr-85) gap fractions dictated by RG 1.183 [Ref. 2] shown above. Therefore, the initial activity of isotope I-131 and Kr-85 are multiplied by 1.6 and 2.0, respectively, in order to accommodate the respective 10% and 8% release fractions directed by regulatory guidance [Ref. 2].

2.3. Pool Decontamination Factor (DF)

Attachment E provides assessments of water coverage for FHAs over the reactor well and the spent fuel pool, and demonstrates that the drop over the reactor well is more limiting. This is due to the greater number of fuel rods damaged for the reactor well drop, and the fact that the lower iodine decontamination factor for a drop over the spent fuel pool is not significant enough to overcome the fuel damage difference.

As prescribed in RG 1.183, Appendix B, for the 23 feet or greater water depth, the overall effective DF of 200 is used, with

DF_{Inorg} = 285.29, which is the inorganic iodine DF that would yield an overall DF of 200 for a 23 foot water depth, when used in conjunction with the chemical fractions listed below. Conservatively, the 500 DF per RG 1.183, Appendix B is <u>not</u> credited.

fraction of inorganic iodine in fuel = 0.9985

fraction of organic iodine in fuel = 0.0015

fraction of elemental iodine above the water = 0.70

fraction of organic iodine above the water = 0.30

2.4. Release Model

Release modeling uses the RADTRAD computer program. As discussed in Section 2.2 above, the normal Nuclide Inventory File (NIF) representing a LGS core is artificially adjusted to account for the higher than average gap fractions for I-131 and Kr-85 provided by RG 1.183.

The compartments are the Secondary Containment Refuel Floor Air Space, the Environment, and the Control Room. The refuel floor exhaust rate is set artificially high at 6 air changes per hour. This results in 99.9994% of the contained radioactivity being exhausted within two hours.

The exhaust point under the assumed no filtration condition is the Reactor Building South Stack as per Ref. 12. This release point results in specific dispersion characteristics which are defined by unique dispersion factors, or χ/Q's, as derived in Ref. 6. The North Stack, which is used for releases filtered by the SGTS, is located closer to the Control Room intake and therefore has higher χ/Q's, as also derived in Ref. 6. However, the SGTS is designed to remove at least 99% of the iodine that would otherwise be released; this filtration more than overcomes the effect of the higher χ/Q's, as demonstrated herein, so the South Stack release unfiltered is bounding.

Site walkdowns and specific reviews of the LGS General Arrangement Drawings such as M-102, the Plan at El. 217' – 0" (one foot above Grade) [Ref. 15] and M-107 [Ref. 16], the Section showing the North and South Stack, confirmed that there are no potential release pathways that could be worse with respect to the Control Room intake than the analyzed stacks. In particular, there are no hatches or single-door Reactor Building openings leading directly to the outside, and grade openings are considered to have χ/Q's that are bounded by the South Stack release point χ/Q based on the greater distances of travel required for releases from them to the Control Room intake. This includes the large railroad doors at grade elevation, which could be postulated to be open at the same time as the equipment hatch cover on the refueling floor to support a future spent fuel cask move. Any other non-normal opening that could be postulated would be evaluated for its effects on this accident before such simultaneous opening would be allowed.

2.5. Control Room Model

The Control Room (CR), as analyzed for this FHA analysis, is unfiltered. Although the normal maximum flow into the CR is 2100 cfm, a Control Room changeover rate of 1 CR volume per minute is used for conservatism and to allow for any unfiltered inleakage. Flow into the CR is therefore assumed to be 126,000 cfm, and to balance the system for analytical purposes, an equal flow of clean air is considered to leave the CR.

2.6. Dose Modeling

Dose models for both onsite and offsite meet RG 1.183 requirements. Dose conversion factors are based on Federal Guidance Reports 11 and 12 [Ref. 4, 5]. RADTRAD uses the following formulations, integrated numerically over the accident duration:

2.6.1. EAB and LPZ

Doses at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) for the FHA are based on the following formulas:

$$\text{Dose}_{\text{CEDE}}\ (\text{rem}) = \text{Release (Curies)} * \frac{\chi}{Q}(\text{sec/m}^3) * \text{Breathing Rate (m}^3\text{/sec)} * \text{Inhalation DCF (rem}_{\text{CEDE}}\text{/Ci inhaled)}$$

and

$$\text{Dose}_{\text{EDE}}\ (\text{rem}) = \text{Release (Curies)} * \frac{\chi}{Q}(\text{sec/m}^3) * \text{Submersion DCF (rem}_{\text{EDE}} - \text{m}^3\text{/ Ci - sec)}$$

and finally,

$$Dose_{TEDE}\ (rem) = Dose_{CEDE}\ (rem) + Dose_{EDE}(rem)$$

2.6.2. Control Room

The formulas used by RADTRAD, by time increment, are:

$$\text{Dose}_{\text{CEDE}}\ (\text{rem}) = \text{Time Dependent CR Air Concentration (Ci/m}^3\text{)} * \text{Time Increment Duration (sec)} * \\ \text{Breathing Rate (m}^3\text{/sec)} * \text{Inhalation DCF (rem}_{\text{CEDE}}\text{/Ci inhaled)} * \text{Occupancy Factor of 1}$$

and

$$\text{Dose}_{\text{EDE}}\ (\text{rem}) = \text{Time Dependent CR Air Concentration (Ci/m}^3\text{)} * \text{Time Increment Duration (sec)} * \\ \text{Submersion DCF (rem}_{\text{EDE}} - \text{m}^3\text{/ Ci - sec)} * \text{Occupancy Factor of 1} * \text{CR Geometry Factor}$$

and finally,

$$\text{Dose}_{\text{TEDE}}\ (\text{rem}) = \text{Dose}_{\text{CEDE}}\ (\text{rem}) + \text{Dose}_{\text{EDE}}\ (\text{rem})$$

2.7. Acceptance Criteria

Dose acceptance criteria are per 10CFR50.67 and RG 1.183 guidance.

Table 2.1 lists the regulatory limits for accidental dose to 1) a control room operator, 2) a person at the EAB, and 3) a person at the LPZ boundary.

Table 2.1. Regulatory Dose Limits (Rem TEDE)*

CR (30 days)	EAB (2 hours)	LPZ (30 days
5	6.3	6.3

Direct conformance with the relevant sections of the body of Regulatory Guide 1.183 (such as the Acceptance Criteria provided above) and all of the Assumptions in its Appendix B "Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident" is provided by this analysis, as shown in the Conformance Matrix Table 2.2.

Table 2.2: Conformance with RG 1.183 Appendix B (Fuel Handling Accident)			
RG Section	RG Position	LGS Analysis	Comments
1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms	These assumptions are utilized; see Section 2 of this calculation
1.1	The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.	Conforms	A conservative fuel damage analysis has been performed; see Section 2.1 and Attachment E of this calculation.
1.2	The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.	Conforms	These assumptions are utilized; see Section 2.2 of this calculation.
1.3	The chemical form of radioiodine released from the fuel to the spent fuel pool should be 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. The NRC staff will consider, on a case-by-case basis, justifiable mechanistic treatment of the iodine release from the pool.	Conforms	All iodine added to pool is assumed to dissociate.
2	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. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-	Conforms, the 500 DF for elemental iodine is not used. A more conservative value of 285.29 is used since it	The decontamination factor was determined in a more conservative manner than prescribed in RG 1.183, as described in Section 2.3 of this calculation.

Table 2.2: Conformance with RG 1.183 Appendix B (Fuel Handling Accident)			
RG Section	**RG Position**	**LGS Analysis**	**Comments**
	by-case method.	is the value that yields an overall effective DF of 200 for 23 feet of water when combined with the stated initial iodine fractions.	
3	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).	Conforms	These assumptions are utilized.
4.1	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.	Conforms	This assumption is utilized. No credit is taken for the SGTS.
4.2	A reduction in the amount of radioactive material released from the fuel pool by engineered safety feature (ESF) filter systems may be taken into account provided these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02. Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.	Not Applicable	No credit is taken for filtration from the reactor building.
4.3	The radioactivity release from the fuel pool should be assumed to be drawn into the ESF filtration system without mixing or dilution in the fuel building. If mixing can be demonstrated, credit for mixing and dilution may be considered on a case-by-case basis. This evaluation should consider the magnitude of the building volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the pool, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the pool and the exhaust plenums.	Not Applicable	Two-hour release to the environment is assumed, without mixing or dilution.

Table 2.2: Conformance with RG 1.183 Appendix B (Fuel Handling Accident)

RG Section	RG Position	LGS Analysis	Comments
5.1	If the containment is isolated during fuel handling operations, no radiological consequences need to be analyzed.	Not Applicable	Containment is not isolated.
5.2	If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment, no radiological consequences need to be analyzed.	Not Applicable	Containment is not isolated. No credit is taken for "defense in depth " actions.
5.3	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.	Conforms	This 2-hour release assumption is utilized.
5.4	A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02. Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.	Not Applicable	No credit is taken for filtration of release from the reactor building.
5.5	Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50% of the containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the reactor cavity and the exhaust plenums.	Not Applicable	No credit is taken for dilution or mixing of the activity released from the reactor cavity. A 2-hour release assumption is utilized.

3. ASSUMPTIONS

Assumptions and analyzed conditions regarding the fuel handling accident scenarios are provided below.

1. Movement of recently irradiated fuel will not occur less than 24 hours after the associated reactor shutdown, per LGS Technical Specification 3/4.9.4.
2. Fuel bundle peak burnup will not exceed the RG 1.183 footnotes 10 and 11 limit of 62 GWD/MTU.
3. For fuel exceeding a 54 GWD/MTU burnup, the maximum linear heat generation rate will not exceed the RG 1.183 footnote 11 limit of 6.3 kW/ft rod average power.
4. As shown in Attachment E, the bounding fuel damage assessment scenario associated with a drop over the reactor core is used. For this event the RG 1.183 DF value of 200 is conservative.
5. Spent fuel source terms are based on reactor core source terms as discussed in Attachment A.
6. Activity reaching the refuel floor airspace will essentially all be exhausted within 2 hours by using an artificially high exhaust rate. This also provides an allowance for uneven mixing in the refuel floor airspace. (See Section 2.4)
7. The release pathway is unfiltered through the worst case Reactor Building South Stack to the Control Room normal intake. (See Section 2.4)
8. No credit is taken for the operation of the CREFAS system during the FHA. Conservatively high intake and outflow are considered. (See Section 2.4)
9. As the LGS Control Room has no exterior walls or overlying structures that are less than 2 feet thick concrete, this is considered sufficient to eliminate separate consideration of the radiation shine from the external radioactive plume release.

4. DESIGN INPUT

The design inputs used for this calculation are summarized in the following Table 4:

Parameters Applicable to AST Fuel Handling Accident Dose Considerations for Limerick Generating Station

TABLE 4: FHA AST Analysis Parameter or Method for Limerick Generating Station	AST Value	Source Documents
Reactor Power	3527 MWth	Calc. No. LM-312, Rev. 0
Damaged Rods	212	Ref. 1
Fuel Assembly Configuration	8x8 in a 62 fuel pin bundle	Ref. 1
Peaking Factor	1.7	New bounding analysis value compared to 1.5 in Ref. 1
Allowable Fuel Burnup and non-LOCA gap fractions	Peak burnup less than 62 GWD/MTU	RG 1.183, Table 3
FHA Radionuclide Inventory	From Attachment A of this Calc. for the 60 isotopes forming the standard RADTRAD library, with decay to 24 hours. Gap activities per R.G. 1.183.	See Attachment A 24 hours per Ref. 1 and Technical Specification 3 / 4.9.4, for fuel not recently irradiated
Underwater Decontamination Factor	Noble Gases: 1 Particulate (cesium and rubidium): infinity Iodine: 200, conservative value for drop over reactor well.	RG 1.183 RG 1.183 RG 1.183 (See Attachment E)
Dose Conversion Factors	EPA Federal Guidance Reports 11 and 12	Ref. 4 and 5
Offsite Dose Limit	6.3 rem TEDE for the duration of the accident	RG 1.183
Control Room Dose Limit	5 rem TEDE for the duration of the accident	10CFR50 App. A, GDC 19 and 10CFR50.67
Secondary Containment Automatic Isolation and Filtration	Not credited	RG 1.183
Mitigation by CREF System	Not credited	RG 1.183
Normal Control Room Fresh Air Make-up Rate and Volume	An artificial bounding value 1 CR volume per minute is used for conservatism and to allow for any unfiltered inleakage; therefore flow is assumed to be 126,000 cfm (actual design values are more than an order of magnitude lower).	

TABLE 4: FHA AST Analysis Parameter or Method for Limerick Generating Station	**AST Value**	**Source Documents**
	Volume 126,000 ft^3	Ref. 14
Refuel Floor Normal Ventilation rate	Approximately 6 air changes per hour and an artificial value of 100 ft^3 is used for simplicity. This evacuates 99.9994% of all activity within 2-hours.	Conservative value for calculation
CR Release Point Basis Dispersion Factors 0 – 2 hr	 Reactor Building South Stack 1.26E-03 sec/m^3	 Ref. 12 Ref. 6
EAB Release Point Basis and Distance to EAB Dispersion Factors 0 – 2 hr	 Reactor Building South Stack and 731 m 3.18E-04 sec/m^3	 Ref. 12 Ref. 6
LPZ Release Point Basis and Distance to LPZ Dispersion Factors 0 – 2 hr	 Reactor Building South Stack and 2043 m 1.15E-04 sec/m^3	 Ref. 12 Ref. 6

5. REFERENCES

1. Limerick Generating Station Units 1 & 2, UFSAR, Revision 12.

2. Regulatory Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors", July 2000.

3. NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation", April1998, and Supplements 1, June 1999, and 2, October 2002.

4. Federal Guidance Report (FGR) No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion", 1988.

5. FGR No. 12, "External Exposure to Radionuclides in Air, Water, and Soil", 1993.

6. LGS Design Analysis LM-0641, "Calculation of Alternative Source Terms Onsite and Offsite $^{\chi}/_{Q}$ Values", Rev. 0.

7. LGS Design Analysis LM-0312, "Impact of Power Rerate on Fuel Handling Accident Doses and Activities", Rev. 0.

8. G. Burley, "Evaluation of Fission Product Release and Transport", Staff Technical Paper, 1971.

9. NEDC-32868P, "GE14 Compliance With Amendment 22 of NEDE-24011-P-A (GESTAR II)", Rev. 1, September 2000.

10. NEDE–31152P, "General Electric Fuel Bundle Designs" February 1993.

11. GESTAR II, NEDE-24011-P-A-11-US, Refueling Accident Analysis.

12. Limerick Generating Station Drawing SIM-M-76, Sheet 2, "Reactor Enclosure and Refueling Area – HVAC (Unit 2)", Rev. A (applied to both Units).

13. NEDE-24011-P-A-14-US, General Electric Standard Application for Reactor Fuel, Licensing Topical Report, June 2000

14. LGS Calculation No. M-78-01, "Control Room Area – Room Volume", Rev. 6.

15. Drawing M-102, "General Arrangement Plan at El. 217'–0", Rev. 10.

16. Drawing M-107, "General Arrangement Section A-A & B-B", Rev. 8.

17. LGS Design Analysis LM-0656, "Determine Failed Fuel Rods From Collision of Refuel Bridges", Rev. 0.

18. LGS Design Analysis LM-0641, "Determine Impact of RCWP Hoist Drop on UFSAR Fuel Handling Accident", Rev. 0.

6. CALCULATIONS

6.1. New Dose Analysis Basis

This calculation evaluates the radiological dose to an operator in the Control Room and a person at the EAB and LPZ locations following an FHA involving irradiated fuel that has decayed for a minimum of 24 hours after shutdown. This analysis uses Alternative Source Term assumptions per guidance in RG 1.183. The RADTRAD v. 3.03 computer code was used for this Limerick Units 1 & 2 FHA calculation. Releases are treated as unfiltered (no SGTS) through the Reactor Building South Stack (a 99% SGTS filtered release through the North Stack is considered separately in another RADTRAD run to show it is not bounding).

The RADTRAD inputs are summarized below:

A. Compartments

1. Containment – This compartment represents the Reactor Building Air Space, into which fission products leaving the spent fuel pool are released.
   a. Compartment type – Other – since it is not the environment or control room.
   b. Volume – 1.000E+02 ft^3 – This is an artificial RB volume, used to simplify the evacuation of activity from the RB to the environment. An exhaust rate was tailored to this nominal volume in order to model 6 air changes per hour, which ensures that essentially all activity is released within the 2-hour period.
   c. Source term fraction – 1.0
   d. Compartment features – no compartment removal mechanisms selected.

2. Environment
   a. Compartment type – Environment

3. Control Room
   a. Compartment type – Control Room
   b. Volume – 126,000 ft^3 – Ventilated volume.
   c. Source term fraction – 0.0
   d. Compartment features –none selected

B. Transfer Pathways

1. Filtered Flow, Leak to the Environment
   a. From Compartment 1 – Containment
   b. To Compartment 2 – Environment
   c. Transfer mechanism – "Filter" selected
   d. Filter Efficiency Panel – Flow rate – 10 cfm – This is an arbitrary value that was set to ensure the release of 99.9994% of the activity from the Reactor Building within 2 hours, based on the nominal value as set above. This flow rate corresponds to 6 air changes per hour.
   e. Filter Efficiency Panel – The efficiency used is 0 to represent no SGTS filtation.

2. Environment to Control Room
   a. From Compartment 2 – Environment
   b. To Compartment 3 – Control Room
   c. Transfer mechanism – "Filter" selected
   d. Filter Efficiency Panel – 126,000 cfm – Artificially high CR intake flowrate of one air change per minute, to conservatively allow for any unfiltered inleakage, for the duration of the accident.

e. Filter Efficiency Panel – Filter efficiency is entered as 0.0% for all chemical forms of iodine, to show that no CREF is credited.

3. Control Room to Environment
   a. From Compartment 3 – Control Room
   b. To Compartment 2 – Environment
   c. Transfer mechanism – "Filter" selected
   d. Filter Efficiency Panel –Flow rate – Flow rate – 126,000 cfm for the duration of the accident
   e. Filter Efficiency Panel – Filter efficiency is entered as 100.0% iodine chemical for all time periods. This is the exit from the control room to the environment; the filtration prevents a double counting of the iodine release. Note that the noble gas release will still be re-circulated between the control room and the outside environment.

C. Dose Locations

1. Exclusion Area Boundary
   a. In Compartment 2 – Environment
   b. $\chi/_Q$ – 3.18E-04 sec/m^3 – this is the 0-2 hr accident $\chi/_Q$ for LGS Reactor Building South Ventilation Stack. This value is entered from time 24-hours to the end of the accident.
   c. Breathing Rate – 3.5E-04 m^3/sec.
2. Low Population Zone
   a. In Compartment 2 – Environment
   b. $\chi/_Q$ – 1.15E-04 sec/m^3 – this is the 0-2 hr accident $\chi/_Q$ for LGS Reactor Building South Ventilation Stack. This value is entered from time 24-hours to the end of the accident.
   c. Breathing Rate – 3.5E-04 m^3/sec.

3. Control Room
   a. In Compartment 3 – Control Room
   b. $\chi/_Q$ – 1.26E-03 sec/m^3 – this is the 0-2 hr accident $\chi/_Q$ for LGS Reactor Building South Ventilation Stack. This value is entered from time 24-hours to the end of the accident.
   c. Breathing Rate – 3.5E-04 m^3/sec
   d. Occupancy Factor – 1.0 – this the RG 1.183 value for the first day.

D. Source Term and Release Fraction Treatment
   a. The *"Limerick Generating Station AST Source Terms for FHA.nif"* file [Attachment C] reflects the LGS core activities with the I-131 value multiplied by 1.6 and the Kr-85 value multiplied by 2.0. These changes are made so that the Reg. Guide 1.183, Table 3 differentiation in release fraction can be made.
   b. The power level of 26.83 MW is per section 2.1 above and reflects the fraction of the core damaged and the radial peaking factor applied to that fuel.
   c. The 24-hour delay time reflects the minimum time after shutdown that fuel movement is expected.
   d. The file *"Limerick Generating Station AST FHA.rft"* [Attachment D] is designed to reflect gap activity fractions per Reg. Guide 1.183, Table 3, with the adjustment for the "nif" file described above.

E. Dose Conversion Factors

The default FGR-11 and FGR-12 dose conversion factors provided with RADTRAD are used.

6.2. *Margin Assessment for Other Previously Considered FHA Scenarios*

Additional FHA scenarios have been considered in Calculations LM-0656 and LM-0657 (Ref. 17 and 18). Ref. 17 considers a collision of 2 refuel bridges carrying 24 hour (since irradiation) spent fuel over the Spent Fuel Pool, causing both spent fuel bundles to drop over the spent fuel racks. Ref. 18 considers a drop of 24 hour (since irradiation) spent fuel over the open reactor vessel onto the core, combined with a simultaneous drop of a jib crane and its suspended load for a total drop load of 1000 pounds from the Reactor Cavity Work Platform (RCWP).

Both of these scenarios are addressed based on application of AST methodology using an assessment of available margin to the New Dose Analysis Basis (as described in Sections 2.1 and 7.0) in the Attachment G spreadsheet.

7. SUMMARY AND CONCLUSIONS

The RADTRAD code was used to examine the effect of the alternative source term release on offsite and CR doses. Shown below are the results, as provided in the first RADTRAD run in Attachment B, as well as the dose acceptance criteria.

Location	Dose (rem TEDE)
LIMITS	CR 5.0; EAB & LPZ 6.3
EAB	1.52
LPZ	0.549
CR	4.47

The alternative of treating releases 99% SGTS filtered through the North Stack, with a χ/Q to the Control Room Intake of 6.88E-03 sec/m^3, is considered in the second RADTRAD run in Attachment B, with non-bounding results 0.620 rem TEDE to the CR, 0.395 rem TEDE to the EAB, and 0.143 rem TEDE to the LPZ.

These results indicate that the calculated consequences of the postulated Fuel Handling Accident at or after 24 hours of shutdown will be within regulatory limits without the requirement of SGTS filtration. Furthermore, Control Room filtration is not required to maintain operator doses within regulatory limits. These results will bound any other fuel handling accident scenarios that are:

- based on the GESTAR II damage analysis assessment methodology of Attachment E,
- have a Damaged Core Fraction with PF (as derived in Section 2.1) up to 0.007608, for either:
    - o an overall effective DF of 200 (for 23 feet minimum water coverage), or
    - o for an assumed drop over the spent fuel storage racks with less than 23 feet water coverage, the calculated number of damaged rods is adjusted by the factor of [200 divided by the appropriate DF for the water coverage over the damaged rods]

      (pages E-5 through E-8 of Attachment E provides an example of such spent fuel pool DF consideration).

As examples, Attachment G shows that both of the scenarios addressed in Section 6.2 are bounded by the margin to the New Dose Analysis Basis described in Sections 2.1, and establishes the limiting number of damaged rods in the struck bundles for each scenario.

8. OWNER'S ACCEPTANCE REVIEW CHECKLIST FOR EXTERNAL DESIGN ANALYSIS

DESIGN ANALYSIS NO. LM-0645 REV: 1

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

EXELON REVIEWER: T. J. Mscisz / [signature] **DATE:** 9/27/05

Print / Sign

ATTACHMENT A
Source Terms

{The following Source Terms derivation is from the Peach Bottom Atomic Power Station (PBAPS) AST FHA Calculation PM-1059. The results are also applicable to LGS, as confirmed by Exelon on the last two pages of this Attachment}

Introduction

The following description and calculation input and results from Robert Jaffa of Exelon Nuclear derives the isotopic inventory (core source terms) to be used as inputs for the PBAPS AST Design Basis Accident analyses. The derivation uses an Exelon Nuclear controlled-version of ORIGEN 2.1, as noted. Although derived for PB Unit 3, they are considered as applicable to PB Unit 2 as well.

The core source terms are initially calculated based on a power level of 3514.9 MWt, which is approximately the Rated Thermal Power of 3514 MWt. The AST analyses utilize a Design Basis Accident power level of 3528 MWt. RADTRAD uses a NIF file in units of Curies/MWt and multiplies this by the specified core power, so the derived Curie quantities for the 60 RADTRAD isotopes in the following Table 2 are divided by 3514.9 to create the NIF file as presented in Attachment C, and 3528 MWt is used as the input specified core power for full core power analyses.

1.0 PURPOSE

To calculate the isotopic core inventory for PB Unit 3 using the ORIGEN2.1 code based on reactor operation at 3514.9 MWt and an equilibrium 711 EFPD two-year cycle design.

2.0 SUMMARY OF RESULTS

The bounding isotopic core inventory for a 711 EFPD two-year cycle design at PB Unit 3 is shown in terms of activity (Curies) and concentration (grams) in Tables 2 and 4, respectively. This bounding isotopic core inventory was determined for PB Unit 3's expected power uprate power level of 3514.9 MWt, which is approximately equal to the rated thermal power of 3514 MWt.

3.0 REFERENCES

3.1 RSIC Code Package CCC-371, "ORIGEN 2.1, Isotope Generation and Depletion Code Matrix Exponential Method," May 1999.

3.2 Exelon Nuclear memo NFM-MA:03-002 from J. Wolfrom to R. Jaffa, Re: "Peach Bottom Alternate Source Term Information," dated January 9, 2003.

3.3 Memo from R. Jaffa to R. Tropasso, Re: "Installation Verification of ORIGEN2.1 on the IBM PC Platform," dated July 7, 2000.

3.4 ORNL/TM-11018, "Standard- and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code," S. Ludwig, J. Renier, December 1989.

4.0 ASSUMPTIONS

4.1 The ORIGEN2.1 code [Ref 3.1] was used to calculate isotopic activities based on the cycle design described in Reference 3.2. The ORIGEN2.1 code was run on an IBM-PC and was confirmed to be controlled per Ref. 3.3.

4.2 Batch-average enrichments and exposures from Ref. 3.2 were used to develop input to ORIGEN2.1. This is equivalent to performing individual calculations for each sub-batch.

4.3 For fuel burned in more than one cycle, ORIGEN2.1 runs ignored refueling outages. This has no impact on short-lived isotopes which reach equilibrium concentrations shortly after cycle startup and has a conservative, albeit minimal, impact on long-lived isotopes which continually increase in concentration as a function of exposure; ignoring intermediate decay periods will increase the final concentrations.

4.4 For isotopic activities calculated at BOC, BOC is defined as 100 days into the cycle to ensure that all short-lived isotopes (half lives < 1 year) are at equilibrium levels.

5.0 DESIGN INPUT

5.1 Batch-average information for an equilibrium two-year cycle (Ref. 3.2) was obtained using the FCYCLE01 code starting from the PB3 Cycle 14 design.

5.2 The ORIGEN2.1 cross-section library, BWRUE.LIB, is used in this calculation as this is most representative of current PB3 two-year cycles. The library is based on an "extended cycle" reactor model where fuel achieves 40 GWd/mtU burnup in four cycles.

5.3 Equilibrium cycle lengths are 711 EFPD.

5.4 The power level used for determining exposure data for the equilibrium two-year cycle is 3514.9 MWt, which is approximately the Rated Thermal Power level for PB3 of 3514 MWt after the 1.62% Caldon power uprate.

6.0 OVERALL APPROACH AND METHODOLOGY

The isotopic core inventory is a function of the reactor power level and the exposure of the fuel. The PB Unit 3 isotopic core inventory is calculated using the ORIGEN2.1 code [Ref. 3.1]. A 3514.9 MWt 711 EFPD two-year equilibrium cycle design [from Ref. 3.2] is used as the basis for this calculation

6.1 Computer Code Information

The batch file used to execute ORIGEN2.1 (DTSQA Product ID# EX0004724) for this calculation provide the paths and filenames of the executable program and libraries that were called. The batch file used is **pb3_ast.bat.** The PC-based ORIGEN2.1 code used in this calculation was verified to be controlled by comparing the time/date/size stamp of the executable file to that documented in Ref. 3.3.

Volume in drive D is PER30290
Volume Serial Number is 11F7-0C1F
Directory of D:\Origen21\CODE

ORIGEN2 EXE 1,267,348 06-10-96 1:09p ORIGEN2.EXE

The time/date/size stamps of the library files used in this calculation were verified against those documented in Ref. 3.4.

Volume in drive D is PER30290
Volume Serial Number is 11F7-0C1F
Directory of D:\Origen21\LIBS

BWRUE LIB	173,676	08-01-91	2:10a
DECAY LIB	278,636	08-01-91	2:10a
GXUO2BRM LIB	167,526	08-01-91	2:10a

7.0 CALCULATION

Equilibrium Two-Year Cycle Isotopic Core Inventory

An equilibrium two-year cycle design for Peach Bottom Unit 3 based on the FCYCLE01 code using the actual Cycle 14 design as a reference cycle and a cycle length of 711 EFPD is used as the basis for the source term calculation. In addition, batch average burnups were increased to account for a 1.62% Caldon power uprate. The resulting batch-average burnups for once-burned, twice-burned and thrice-burned fuel batches are shown below.

Batch	# of FA	Avg. Enr. (w/o U235)	Avg. Burnup per Cycle (MWd/mtU)	Power (MW)	Loading (MTU)	U235 wt. (gms)	U238 wt. (gms)	Oxygen wt. (gms)
1	208	4.105	22800.49 18256.15 12116.92	1197.0 958.4 636.1	37.3256	1,532,215.88	35,793,384.12	5,018,379.87
2	280	4.107	22800.49 18256.15	1611.5 1290.3	50.2516	2,063,833.21	48,187,766.79	6,756,264.28
3	276	4.107	22800.49	1588.5	49.53372	2,034,349.88	47,499,370.12	6,659,746.22

The equilibrium cycle isotopic core inventory is calculated using ORIGEN2.1 and the BWR extended burnup cross-section library BWRUE . The input deck is **pb3_ast.inp** and the batch file is **pb3_ast.bat.**

The specific power for a batch in a given cycle is determined by multiplying the batch average burnup for that cycle by the batch loading and then dividing by the number of EFPD in the cycle. For example, the specific power for Batch 1 in its first cycle of operation is:

(22,800.49 * 37.3256) / 711 = 1197.0 MW.

The grams of U235 and U238 for each batch were determined by the following formulas:

$$\text{U235 (gms)} = \text{Batch loading} * (\text{Avg. Enr.}/100) * 10^6$$

$$\text{U238 (gms)} = \text{Batch loading} * (1 - \text{Avg. Enr.}/100) * 10^6$$

The corresponding weight of oxygen in UO_2 pellets for each batch is:

O (gms) = Total batch U weight (gm U) / 238 (gm U/gm atom U) * 2 (gm atom O/gm atom U) * 15.9994 gm O/gm atom O

The ORIGEN2.1 input deck is set up to deplete each fuel batch and write the 100 EFPD and EOC results to temporary storage vectors. Once all batches have been depleted, the results from the temporary vectors are combined to give the results for the entire core. The ORIGEN2.1 core inventory activity and composition results for the equilibrium two-year cycle at 100 EFPD (BOC) and EOC are shown below in

Tables 1 and 3, respectively. The maximum of the 100 EFPD and EOC values for each isotope are selected to generate the bounding isotopic core inventory activity and composition results as shown in Tables 2 and 4, respectively.

Table 1
ORIGEN2.1 Isotopic Activity Results for
Peach Bottom Unit 3

Isotope	100 EFPD (Ci)	EOC (Ci)
KR 83M	1.324E+07	1.158E+07
BR 84	2.373E+07	2.002E+07
BR 85	2.888E+07	2.404E+07
KR 85	8.806E+05	1.387E+06
KR 85M	2.922E+07	2.436E+07
RB 86	1.118E+05	2.291E+05
KR 87	5.739E+07	4.672E+07
KR 88	8.096E+07	6.570E+07
RB 88	8.197E+07	6.678E+07
SR 89	9.836E+07	8.846E+07
SR 90	6.982E+06	1.117E+07
Y 90	7.142E+06	1.150E+07
SR 91	1.336E+08	1.110E+08
Y 91	1.212E+08	1.143E+08
SR 92	1.412E+08	1.205E+08
Y 92	1.416E+08	1.210E+08
Y 93	1.591E+08	1.404E+08
ZR 95	1.521E+08	1.578E+08
NB 95	1.399E+08	1.586E+08
ZR 97	1.637E+08	1.578E+08
MO 99	1.771E+08	1.785E+08
TC 99M	1.551E+08	1.563E+08
RU103	1.234E+08	1.477E+08
RU105	7.642E+07	1.022E+08
RH105	7.310E+07	9.673E+07
RU106	3.856E+07	6.081E+07
SB127	8.572E+06	1.018E+07
TE127	8.402E+06	1.010E+07
TE127M	1.039E+06	1.355E+06
SB129	2.732E+07	3.036E+07
TE129	2.679E+07	2.988E+07
TE129M	3.875E+06	4.453E+06
I129	2.774E+00	4.816E+00
TE131M	1.269E+07	1.360E+07
I131	9.139E+07	9.444E+07

Isotope	100 EFPD (Ci)	EOC (Ci)
XE131M	1.015E+06	1.056E+06
TE132	1.322E+08	1.343E+08
I132	1.338E+08	1.364E+08
I133	1.953E+08	1.925E+08
XE133	1.904E+08	1.930E+08
XE133M	5.956E+06	6.007E+06
I134	2.167E+08	2.118E+08
CS134	1.335E+07	2.559E+07
I135	1.825E+08	1.806E+08
XE135	7.832E+07	7.086E+07
XE135M	3.636E+07	3.773E+07
CS136	3.568E+06	7.123E+06
CS137	9.460E+06	1.595E+07
BA137M	8.965E+06	1.510E+07
XE138	1.679E+08	1.589E+08
CS138	1.841E+08	1.760E+08
BA139	1.787E+08	1.719E+08
BA140	1.721E+08	1.661E+08
LA140	1.764E+08	1.722E+08
LA141	1.631E+08	1.563E+08
CE141	1.579E+08	1.575E+08
LA142	1.593E+08	1.510E+08
CE143	1.556E+08	1.449E+08
PR143	1.509E+08	1.416E+08
CE144	1.012E+08	1.264E+08
ND147	6.459E+07	6.321E+07
NP239	1.616E+09	1.897E+09
PU238	2.639E+05	6.312E+05
PU239	3.100E+04	4.218E+04
PU240	2.871E+04	4.526E+04
PU241	1.365E+07	2.173E+07
AM241	1.634E+04	3.349E+04
CM242	3.645E+06	8.393E+06
CM244	2.654E+05	9.147E+05

Table 2
Bounding Isotopic Core Inventory
Peach Bottom Unit 3

Isotope	Isotopic Activity (Ci)
KR 83M	1.324E+07
BR 84	2.373E+07
BR 85	2.888E+07
KR 85	1.387E+06
KR 85M	2.922E+07
RB 86	2.291E+05
KR 87	5.739E+07
KR 88	8.096E+07
RB 88	8.197E+07
SR 89	9.836E+07
SR 90	1.117E+07
Y 90	1.150E+07
SR 91	1.336E+08
Y 91	1.212E+08
SR 92	1.412E+08
Y 92	1.416E+08
Y 93	1.591E+08
ZR 95	1.578E+08
NB 95	1.586E+08
ZR 97	1.637E+08
MO 99	1.785E+08
TC 99M	1.563E+08
RU103	1.477E+08
RU105	1.022E+08
RH105	9.673E+07
RU106	6.081E+07
SB127	1.018E+07
TE127	1.010E+07
TE127M	1.355E+06
SB129	3.036E+07
TE129	2.988E+07
TE129M	4.453E+06
I129	4.816E+00
TE131M	1.360E+07
I131	9.444E+07

Isotope	Isotopic Activity (Ci)
XE131M	1.056E+06
TE132	1.343E+08
I132	1.364E+08
I133	1.953E+08
XE133	1.930E+08
XE133M	6.007E+06
I134	2.167E+08
CS134	2.559E+07
I135	1.825E+08
XE135	7.832E+07
XE135M	3.773E+07
CS136	7.123E+06
CS137	1.595E+07
BA137M	1.510E+07
XE138	1.679E+08
CS138	1.841E+08
BA139	1.787E+08
BA140	1.721E+08
LA140	1.764E+08
LA141	1.631E+08
CE141	1.579E+08
LA142	1.593E+08
CE143	1.556E+08
PR143	1.509E+08
CE144	1.264E+08
ND147	6.459E+07
NP239	1.897E+09
PU238	6.312E+05
PU239	4.218E+04
PU240	4.526E+04
PU241	2.173E+07
AM241	3.349E+04
CM242	8.393E+06
CM244	9.147E+05

Table 3
ORIGEN2.1 Isotopic Concentration Results for
Peach Bottom Unit 3

Isotope	100 EFPD (grams)	EOC (grams)
KR 83M	6.414E-01	5.611E-01
BR 84	3.370E-01	2.843E-01
BR 85	3.741E-02	3.114E-02
KR 85	2.244E+03	3.534E+03
KR 85M	3.550E+00	2.959E+00
RB 86	1.373E+00	2.814E+00
KR 87	2.025E+00	1.649E+00
KR 88	6.451E+00	5.235E+00
RB 88	6.826E-01	5.561E-01
SR 89	3.384E+03	3.044E+03
SR 90	5.116E+04	8.183E+04
Y 90	1.312E+01	2.113E+01
SR 91	3.683E+01	3.060E+01
Y 91	4.939E+03	4.658E+03
SR 92	1.123E+01	9.579E+00
Y 92	1.471E+01	1.256E+01
Y 93	4.768E+01	4.206E+01
ZR 95	7.079E+03	7.341E+03
NB 95	3.576E+03	4.054E+03
ZR 97	8.560E+01	8.251E+01
MO 99	3.691E+02	3.720E+02
TC 99M	2.947E+01	2.971E+01
RU103	3.822E+03	4.575E+03
RU105	1.136E+01	1.519E+01
RH105	8.657E+01	1.146E+02
RU106	1.152E+04	1.817E+04
SB127	3.208E+01	3.811E+01
TE127	3.182E+00	3.826E+00
TE127M	1.101E+02	1.436E+02
I127	4.533E+03	8.040E+03
SB129	4.856E+00	5.397E+00
TE129	1.279E+00	1.426E+00
TE129M	1.286E+02	1.478E+02
I129	1.571E+04	2.727E+04
TE131M	1.591E+01	1.704E+01
I131	7.369E+02	7.615E+02

Isotope	100 EFPD (grams)	EOC (grams)
XE131M	1.212E+01	1.260E+01
TE132	4.352E+02	4.423E+02
I132	1.296E+01	1.321E+01
I133	1.723E+02	1.699E+02
XE133	1.017E+03	1.031E+03
XE133M	1.328E+01	1.339E+01
CS133	1.025E+05	1.678E+05
I134	8.118E+00	7.936E+00
CS134	1.031E+04	1.977E+04
I135	5.195E+01	5.140E+01
XE135	3.065E+01	2.773E+01
XE135M	3.990E-01	4.140E-01
CS135	4.502E+04	7.841E+04
CS136	4.867E+01	9.715E+01
CS137	1.087E+05	1.832E+05
BA137M	1.666E-02	2.807E-02
XE138	1.745E+00	1.652E+00
CS138	4.348E+00	4.156E+00
BA139	1.092E+01	1.051E+01
BA140	2.359E+03	2.277E+03
LA140	3.168E+02	3.093E+02
LA141	2.883E+01	2.762E+01
CE141	5.541E+03	5.528E+03
LA142	1.115E+01	1.056E+01
CE143	2.342E+02	2.181E+02
PR143	2.241E+03	2.102E+03
CE144	3.172E+04	3.962E+04
ND147	8.039E+02	7.867E+02
NP239	6.963E+03	8.173E+03
PU238	1.541E+04	3.685E+04
PU239	4.985E+05	6.782E+05
PU240	1.259E+05	1.986E+05
PU241	1.324E+05	2.108E+05
AM241	4.759E+03	9.755E+03
CM242	1.102E+03	2.537E+03
CM244	3.279E+03	1.130E+04

Table 4
Bounding Isotopic Core Inventory
Peach Bottom Unit 3

Isotope	Isotopic Concentration (grams)
KR 83M	6.414E-01
BR 84	3.370E-01
BR 85	3.741E-02
KR 85	3.534E+03
KR 85M	3.550E+00
RB 86	2.814E+00
KR 87	2.025E+00
KR 88	6.451E+00
RB 88	6.826E-01
SR 89	3.384E+03
SR 90	8.183E+04
Y 90	2.113E+01
SR 91	3.683E+01
Y 91	4.939E+03
SR 92	1.123E+01
Y 92	1.471E+01
Y 93	4.768E+01
ZR 95	7.341E+03
NB 95	4.054E+03
ZR 97	8.560E+01
MO 99	3.720E+02
TC 99M	2.971E+01
RU103	4.575E+03
RU105	1.519E+01
RH105	1.146E+02
RU106	1.817E+04
SB127	3.811E+01
TE127	3.826E+00
TE127M	1.436E+02
I127	8.040E+03
SB129	5.397E+00
TE129	1.426E+00
TE129M	1.478E+02
I129	2.727E+04
TE131M	1.704E+01
I131	7.615E+02

Isotope	Isotopic Concentration (grams)
XE131M	1.260E+01
TE132	4.423E+02
I132	1.321E+01
I133	1.723E+02
XE133	1.031E+03
XE133M	1.339E+01
CS133	1.678E+05
I134	8.118E+00
CS134	1.977E+04
I135	5.195E+01
XE135	3.065E+01
XE135M	4.140E-01
CS135	7.841E+04
CS136	9.715E+01
CS137	1.832E+05
BA137M	2.807E-02
XE138	1.745E+00
CS138	4.348E+00
BA139	1.092E+01
BA140	2.359E+03
LA140	3.168E+02
LA141	2.883E+01
CE141	5.541E+03
LA142	1.115E+01
CE143	2.342E+02
PR143	2.241E+03
CE144	3.962E+04
ND147	8.039E+02
NP239	8.173E+03
PU238	3.685E+04
PU239	6.782E+05
PU240	1.986E+05
PU241	2.108E+05
AM241	9.755E+03
CM242	2.537E+03
CM244	1.130E+04

Input Deck pb3_ast.inp

```
-1
-1
-1
BAS     Grams of Heavy Metal per Fuel Batch
RDA     PLACE FUEL into vectors -1, -2 and -3
LIP     0  0  0
LIB     0  1  2  3   657   658   659  9  3  0  1 42
PHO     0 0 0 10
RDA     READ FUEL COMPOSITION FOR BATCH 3
INP     -1  1  -1 -1  1  1
RDA     READ FUEL COMPOSITION FOR BATCH 2
INP     -2  1  -1 -1  1  1
RDA     READ FUEL COMPOSITION FOR BATCH 1
INP     -3  1  -1 -1  1  1
RDA TIT     IRRADIATION OF TMI-1 CYCLE 1 FULL CORE
MOV     -3  1  0  1.0       BATCH 1 FRESH
HED     1    CHARGE
RDA     BATCH 1 BURNUP IN CYCLE 1
BUP
IRP      50.0   1197.0    1    9    4 2
IRP     100.0   1197.0    9    2    4 0
IRP     150.0   1197.0    2    9    4 0
IRP     200.0   1197.0    9    3    4 0
IRP     250.0   1197.0    3    9    4 0
IRP     300.0   1197.0    9    4    4 0
IRP     350.0   1197.0    4    9    4 0
IRP     400.0   1197.0    9    5    4 0
IRP     450.0   1197.0    5    9    4 0
IRP     500.0   1197.0    9    6    4 0
IRP     550.0   1197.0    6    9    4 0
IRP     600.0   1197.0    9    7    4 0
IRP     650.0   1197.0    7    9    4 0
IRP     711.0   1197.0    9    8    4 0
BUP
OPTL    4*8 5 8 5 17*8
OPTA    4*8 5 8 5 17*8
OPTF    4*8 5 8 5 17*8
OUT     -8  1  -1  0
MOV      8  1  0  1.0       BATCH 1 ONCE BURNED
HED     1    CHARGE
RDA     BATCH 1 BURNUP IN CYCLE 2
BUP
IRP     761.0    958.4    1    9    4 3
IRP     811.0    958.4    9    2    4 0
IRP     861.0    958.4    2    9    4 0
IRP     911.0    958.4    9    3    4 0
IRP     961.0    958.4    3    9    4 0
IRP    1011.0    958.4    9    4    4 0
IRP    1061.0    958.4    4    9    4 0
IRP    1111.0    958.4    9    5    4 0
IRP    1161.0    958.4    5    9    4 0
IRP    1211.0    958.4    9    6    4 0
IRP    1261.0    958.4    6    9    4 0
IRP    1311.0    958.4    9    7    4 0
IRP    1361.0    958.4    7    9    4 0
```

```
IRP   1422.0     958.4     9    8    4 0
BUP
OUT     -8  1  -1  0
MOV      8  1  0  1.0       BATCH 1 TWICE BURNED
HED     1     CHARGE
RDA     BATCH 1 BURNUP IN CYCLE 3
BUP
IRP   1472.0     636.1     1    9    4 3
IRP   1522.0     636.1     9    2    4 0
IRP   1572.0     636.1     2    9    4 0
IRP   1622.0     636.1     9    3    4 0
IRP   1672.0     636.1     3    9    4 0
IRP   1722.0     636.1     9    4    4 0
IRP   1772.0     636.1     4    9    4 0
IRP   1822.0     636.1     9    5    4 0
IRP   1872.0     636.1     5    9    4 0
IRP   1922.0     636.1     9    6    4 0
IRP   1972.0     636.1     6    9    4 0
IRP   2022.0     636.1     9    7    4 0
IRP   2072.0     636.1     7    9    4 0
IRP   2133.0     636.1     9    8    4 0
BUP
OUT     -8  1  -1  0
MOV      2 -9   0  1.0       BATCH 1 100 EFPD PLACED IN TEMP VECTOR -9
MOV      8 -10  0  1.0       BATCH 1 EOC3 PLACED IN TEMP VECTOR -10
RDA     BATCH 2 BURNUP IN CYCLE 2
MOV     -2  1  0  1.0       BATCH 2 FRESH
HED     1     CHARGE
RDA     BATCH 2 BURNUP IN CYCLE 2
BUP
IRP     50.0    1611.5     1    9    4 2
IRP    100.0    1611.5     9    2    4 0
IRP    150.0    1611.5     2    9    4 0
IRP    200.0    1611.5     9    3    4 0
IRP    250.0    1611.5     3    9    4 0
IRP    300.0    1611.5     9    4    4 0
IRP    350.0    1611.5     4    9    4 0
IRP    400.0    1611.5     9    5    4 0
IRP    450.0    1611.5     5    9    4 0
IRP    500.0    1611.5     9    6    4 0
IRP    550.0    1611.5     6    9    4 0
IRP    600.0    1611.5     9    7    4 0
IRP    650.0    1611.5     7    9    4 0
IRP    711.0    1611.5     9    8    4 0
BUP
OUT     -8  1  -1  0
MOV      8  1  0  1.0       BATCH 2 ONCE BURNED
HED     1     CHARGE
RDA     BATCH 2  BURNUP IN CYCLE 3
BUP
IRP    761.0    1290.3     1    9    4 3
IRP    811.0    1290.3     9    2    4 0
IRP    861.0    1290.3     2    9    4 0
IRP    911.0    1290.3     9    3    4 0
IRP    961.0    1290.3     3    9    4 0
IRP   1011.0    1290.3     9    4    4 0
IRP   1061.0    1290.3     4    9    4 0
```

```
IRP   1111.0    1290.3    9    5    4 0
IRP   1161.0    1290.3    5    9    4 0
IRP   1211.0    1290.3    9    6    4 0
IRP   1261.0    1290.3    6    9    4 0
IRP   1311.0    1290.3    9    7    4 0
IRP   1361.0    1290.3    7    9    4 0
IRP   1422.0    1290.3    9    8    4 0
BUP
OUT     -8  1   -1   0
ADD      2 -9    0  1.0       BATCH 2 100 EFPD ADDED TO TEMP VECTOR -9
ADD      8 -10   0  1.0       BATCH 2 EOC3 ADDED TO TEMP VECTOR -10
MOV     -1  1    0  1.0       BATCH 3 FRESH
HED      1    CHARGE
RDA      BATCH 3  BURNUP IN CYCLE 3
BUP
IRP     50.0    1588.5    1    9    4 2
IRP    100.0    1588.5    9    2    4 0
IRP    150.0    1588.5    2    9    4 0
IRP    200.0    1588.5    9    3    4 0
IRP    250.0    1588.5    3    9    4 0
IRP    300.0    1588.5    9    4    4 0
IRP    350.0    1588.5    4    9    4 0
IRP    400.0    1588.5    9    5    4 0
IRP    450.0    1588.5    5    9    4 0
IRP    500.0    1588.5    9    6    4 0
IRP    550.0    1588.5    6    9    4 0
IRP    600.0    1588.5    9    7    4 0
IRP    650.0    1588.5    7    9    4 0
IRP    711.0    1588.5    9    8    4 0
BUP
OUT     -8  1   -1   0
ADD      2 -9    0  1.0
ADD      8 -10   0  1.0
MOV     -9  1    0  1.0       CYCLE 3 @ 100 EFPD
MOV    -10  2    0  1.0       CYCLE 3 @ EOC
HED      1       100 EFPD
HED      2       EOC
OUT     -2  1   -1   0
END
2  922350  2034349.88   922380  47499370.12      0   0.0           UO2
4  080000  6659746.22        0        0.0                          UO2
0
2  922350  2063833.21   922380  48187766.79      0   0.0           UO2
4  080000  6756264.28        0        0.0                          UO2
0
2  922350  1532215.88   922380  35793384.12      0   0.0           UO2
4  080000  5018379.87        0        0.0                          UO2
0
END
```

Job Batch File pb3_ast.bat

```
echo off
echo ***********************************************************************
echo ***********************************************************************
echo **                                                                   **
echo **                           O R I G E N 2                           **
echo **             Oak Ridge Isotope GENeration and Depletion Code       **
echo **                         Version 2.1 (8-1-91)                      **
echo **                                                                   **
echo ***********************************************************************
echo **                                                                   **
echo **   Developed by:  Oak Ridge National Laboratory                    **
echo **                  Chemical Technology Division                     **
echo **                                                                   **
echo **   Technical Contact:  Scott B. Ludwig                             **
echo **                       (615) 574-7916   FTS 624-7916               **
echo **                                                                   **
echo **   Distributed by:  Radiation Shielding Information Center (RSIC)  **
echo **                    Oak Ridge National Laboratory                  **
echo **                    P.O. Box 2008                                  **
echo **                    Oak Ridge, TN  37831                           **
echo **                    (615) 574-6176   FTS 624-6176                  **
echo ***********************************************************************
echo ***********************************************************************
pause
echo ** Execution continuing ...                                          **
echo ***********************************************************************
echo ***********************************************************************
echo **                                                                   **
echo **     Version 2.1 (8-1-91) for mainframes and 80386 or 80486 PCs    **
echo **                                                                   **
copy pb3_ast.inp tape5.inp >nul
REM  (NOT USED IN THIS CASE) copy samp_2.u3 tape3.inp >nul
copy \origen21\libs\decay.lib+\origen21\libs\bwrue.lib tape9.inp >nul
copy \origen21\libs\gxuo2brm.lib tape10.inp >nul
\origen21\code\origen2
rem  combine and save files from run
copy tape12.out+tape6.out pb3_ast.u6 >nul
copy tape13.out+tape11.out pb3_ast.u11 >nul
ren tape7.out pb3_ast.pch
ren tape15.out pb3_ast.dbg
ren tape16.out pb3_ast.vxs
ren tape50.out pb3_ast.ech
rem  cleanup files
del tape*.inp
del tape*.out
echo ***********************************************************************
echo ****************** O R I G E N 2 - Version 2.1 ***********************
echo ********************* Execution Completed *************************
echo ***********************************************************************
echo on
```

-----Original Message-----
From: Mscisz, Thomas J. [mailto:thomas.mscisz@exeloncorp.com]
Sent: Wednesday, October 22, 2003 9:53 AM
To: 'paul.reichert@wgint.com'; Reichert, P.T.
Subject: FW: LGS Source Term Information

> -----Original Message-----
> From: Jaffa, Robert P.
> Sent: Monday, October 20, 2003 3:40 PM
> To: Mscisz, Thomas J.
> Cc: Tusar, James J.
> Subject: RE: LGS Source Term Information
>
> Tom,
>
> The attached email indicates that the equilibrium two-year cycle used to
> generate the source term for the Peach Bottom Unit 3 AST analysis will be
> bounding for the cycle designs currently being loaded in the Limerick
> Generating Station (LGS). Since LGS rated power is 1.6% lower than that
> assumed for the PB-3 AST source term, isotopic activities (short and
> long-lived) would be ~1.6% lower for the LGS source term (assuming similar
> cycle designs).
>
> LGS has the same core loading as PB-3 (in terms of mtU of uranium).
> Design cycle lengths for LGS are bounded by the 711 EFPD assumed for the
> PB-3 source term; shorter cycle lengths would result in reduced activities
> from long-lived isotopes. Since the batch sizes, enrichments, and burnup
> distributions are similar between PB-3 and LGS (as noted below), the PB-3
> AST source term would be bounding for current LGS cycle designs.
>
> Bob
> -----Original Message-----
> From: Tusar, James J.
> Sent: Monday, October 20, 2003 10:45 AM
> To: Jaffa, Robert P.
> Cc: Mscisz, Thomas J.
> Subject: LGS Source Term Information
>
> Action Required: Review Peach Bottom Source Term fuel cycle
> data/assumptions for applicability to the LGS Source Term Calculation
> Recommendation: Forward conclusion to Tom Mscisz
>
> Bob:
>
> The Peach Bottom Source Term fuel cycle assumptions are considered
> bounding relative to Limerick Generating Station. The Peach Bottom batch
> sizes, batch average burnups, and reload bundle average enrichments are
> similar to those at Limerick. For example, the bundle average enrichment
> used in the Peach Bottom analysis was approximately 4.11 wt. % U-235. The
> expected bundle average enrichment for LGS is approximately 4.16 wt. %
> U-235. Additionally, the rated thermal power level for the Peach Bottom
> analyses was 3514 MWt (uprated for Appendix K Thermal Power Optimization).

> This is bounding relative to Limerick's 3458 MWt rated thermal power by
> 1.62%. Therefore, the Peach Bottom Source Term fuel cycle data should be
> applicable to the Limerick Source Term calculations. Let me know if you
> have any other questions.
>
>
>
> James J. Tusar, PE
> Manager, BWR Design
> 200 Exelon Way, KSA 2-N
> Kennett Square, PA 19348
> james.tusar@exeloncorp.com
> 610.765.5818 (voice)
> 610.765.5651 (fax)
>
>

```
##########################################################################
   RADTRAD Version 3.03 (Spring 2001) run on  9/23/2005  at  9:48:13
##########################################################################

##########################################################################
                     File information
##########################################################################

 Plant file           = P:\Users\Nuc\Exelon EOC\Discipline
Files\Process\AST\Limerick AST\LGS FHA\RADTRAD\LGS FHA 24hr Delay Test - No Filter
Credit Rev1.psf
 Inventory file       = p:\users\nuc\exelon eoc\discipline
files\process\ast\limerick ast\lgs fha\radtrad\limerick ast source terms for
fha.nif
 Release file         = p:\users\nuc\exelon eoc\discipline
files\process\ast\limerick ast\lgs fha\radtrad\limerick ast fha.rft
 Dose Conversion file = c:\program files\radtrad3-03\defaults\fgr11&12.inp




     #####    ####   #####    #  #     #  #####   #     #  #####
     #    #  #       #        #  ##    #  #    #  #     #    #
     #    #  #       #        #  # #   #  #    #  #     #    #
     #####    ####   ####     #  #  #  #  #####   #     #    #
     #            #  #        #  #   # #  #       #     #    #
     #            #  #        #  #    ##  #       #     #    #
     #        ####   #        #  #     #  #        ####      #


 Radtrad 3.03 4/15/2001
 PBAPS FHA - TB/RB Ventilation Stack to CR Intake, EAB, & LPZ - 24 Hour Delay and
No Filtration Credit
 Nuclide Inventory File:
 p:\users\nuc\exelon eoc\discipline files\process\ast\limerick ast\lgs
fha\radtrad\limerick ast source terms for fha.nif
 Plant Power Level:
  2.6830E+01
 Compartments:
   3
 Compartment 1:
 Containment
   3
  1.0000E+02
   0
   0
   0
   0
   0
 Compartment 2:
 Environment
   2
  0.0000E+00
   0
   0
   0
   0
   0
```

```
Compartment 3:
Control Room
  1
 1.2600E+05
  0
  0
  0
  0
  0
Pathways:
  3
Pathway 1:
Leak to Environment
  1
  2
  2
Pathway 2:
Environment to  Control Room
  2
  3
  2
Pathway 3:
Control Room to  Environment Exhaust
  3
  2
  2
End of Plant Model File
Scenario Description Name:

Plant Model Filename:
ACCEPT\TEST1.PMF
Source Term:
  1
  1   1.0000E+00
c:\program files\radtrad3-03\defaults\fgr11&12.inp
p:\users\nuc\exelon eoc\discipline files\process\ast\limerick ast\lgs
fha\radtrad\limerick ast fha.rft
  2.4000E+01
  1
  0.0000E+00   7.0000E-01   3.0000E-01   1.0000E+00
Overlying Pool:
  0
  0.0000E+00
  0
  0
  0
  0
Compartments:
  3
Compartment 1:
  0
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 2:
```

```
  0
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 3:
  0
  1
  0
  0
  0
  0
  0
  0
  0
Pathways:
  3
Pathway 1:
  0
  0
  0
  0
  0
  1
  2
  2.4000E+01   1.0000E+01   0.0000E+00   0.0000E+00   0.0000E+00
  4.8000E+01   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 2:
  0
  0
  0
  0
  0
  1
  2
  2.4000E+01   1.2600E+05   0.0000E+00   0.0000E+00   0.0000E+00
  4.8000E+01   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 3:
  0
  0
  0
  0
  0
  1
```

```
  2
  2.4000E+01   1.2600E+05   1.0000E+02   1.0000E+02   1.0000E+02
  4.8000E+01   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Dose Locations:
  3
Location 1:
Exclusion Area Bndry
  2
  1
  2
  2.4000E+01   3.1800E-04
  4.8000E+01   0.0000E+00
  1
  2
  2.4000E+01   3.5000E-04
  4.8000E+01   0.0000E+00
  0
Location 2:
Low Population Zone
  2
  1
  2
  2.4000E+01   1.1500E-04
  4.8000E+01   0.0000E+00
  1
  2
  2.4000E+01   3.5000E-04
  4.8000E+01   0.0000E+00
  0
Location 3:
Control Room
  3
  0
  1
  2
  2.4000E+01   3.5000E-04
  4.8000E+01   0.0000E+00
  1
  2
  2.4000E+01   1.0000E+00
  4.8000E+01   0.0000E+00
Effective Volume Location:
  1
  2
  2.4000E+01   1.2600E-03
  4.8000E+01   0.0000E+00
Simulation Parameters:
  5
  2.4000E+01   1.0000E-03
  2.4010E+01   1.0000E-02
  2.4100E+01   1.0000E-01
  2.6000E+01   1.0000E+00
  4.8000E+01   0.0000E+00
Output Filename:
```

```
 P:\Users\Nuc\Exelon EOC\Discipline Files\Process\AST\Limerick AST\LGS
FHA\RADTRAD\LGS FHA 24hr Delay Test - No Filter Credit Rev1.o0
   1
   1
   1
   0
   0
 End of Scenario File
```

```
##########################################################################
   RADTRAD Version 3.03 (Spring 2001) run on  9/23/2005  at  9:48:13
##########################################################################

##########################################################################
                    Plant Description
##########################################################################

Number of Nuclides =  60

Inventory Power =   1.0000E+00 MWth
Plant Power Level =   2.6830E+01 MWth

Number of compartments    =   3

Compartment information

Compartment number  1  (Source term fraction =   1.0000E+00
)
Name: Containment
Compartment volume =   1.0000E+02 (Cubic feet)
Compartment type is Normal
Pathways into and out of compartment  1
   Exit Pathway Number    1: Leak to Environment

Compartment number  2
Name: Environment
Compartment type is Environment
Pathways into and out of compartment  2
   Inlet Pathway Number   1: Leak to Environment
   Inlet Pathway Number   3: Control Room to  Environment Exhaust
   Exit Pathway Number    2: Environment to  Control Room

Compartment number  3
Name: Control Room
Compartment volume =   1.2600E+05 (Cubic feet)
Compartment type is Control Room
Pathways into and out of compartment  3
   Inlet Pathway Number   2: Environment to  Control Room
   Exit Pathway Number    3: Control Room to  Environment Exhaust

Total number of pathways =   3
```

```
##########################################################################
   RADTRAD Version 3.03 (Spring 2001) run on  9/23/2005  at  9:48:13
##########################################################################

##########################################################################
                     Scenario Description
##########################################################################
```

Time between shutdown and first release = 2.4000E+01 (Hours)

Radioactive Decay is enabled
Calculation of Daughters is enabled

Release Fractions and Timings

	GAP	EARLY IN-VESSEL	LATE RELEASE	RELEASE MASS
	0.000001 hr	0.0000 hrs	0.0000 hrs	(gm)
NOBLES	5.0000E-02	0.0000E+00	0.0000E+00	3.108E+00
IODINE	2.5000E-04	0.0000E+00	0.0000E+00	2.795E-03
CESIUM	0.0000E+00	0.0000E+00	0.0000E+00	0.000E+00
TELLURIUM	0.0000E+00	0.0000E+00	0.0000E+00	0.000E+00
STRONTIUM	0.0000E+00	0.0000E+00	0.0000E+00	0.000E+00
BARIUM	0.0000E+00	0.0000E+00	0.0000E+00	0.000E+00
RUTHENIUM	0.0000E+00	0.0000E+00	0.0000E+00	0.000E+00
CERIUM	0.0000E+00	0.0000E+00	0.0000E+00	0.000E+00
LANTHANUM	0.0000E+00	0.0000E+00	0.0000E+00	0.000E+00

Inventory Power = 27. MWt

Nuclide Name	Group	Specific Inventory (Ci/MWt)	half life (s)	Whole Body DCF (Sv-m3/Bq-s)	Inhaled Thyroid (Sv/Bq)	Inhaled Effective (Sv/Bq)
Kr-85	1	7.892E+02	3.383E+08	1.190E-16	0.000E+00	0.000E+00
Kr-85m	1	8.313E+03	1.613E+04	7.480E-15	0.000E+00	0.000E+00
Kr-87	1	1.633E+04	4.578E+03	4.120E-14	0.000E+00	0.000E+00
Kr-88	1	2.303E+04	1.022E+04	1.020E-13	0.000E+00	0.000E+00
I-131	2	4.299E+04	6.947E+05	1.820E-14	2.920E-07	8.890E-09
I-132	2	3.881E+04	8.280E+03	1.120E-13	1.740E-09	1.030E-10
I-133	2	5.556E+04	7.488E+04	2.940E-14	4.860E-08	1.580E-09
I-134	2	6.165E+04	3.156E+03	1.300E-13	2.880E-10	3.550E-11
I-135	2	5.192E+04	2.380E+04	8.294E-14	8.460E-09	3.320E-10
Xe-133	1	5.491E+04	4.532E+05	1.560E-15	0.000E+00	0.000E+00
Xe-135	1	2.228E+04	3.272E+04	1.190E-14	0.000E+00	0.000E+00

Nuclide	Daughter	Fraction	Daughter	Fraction	Daughter	Fraction
Kr-85m	Kr-85	0.21	none	0.00	none	0.00
Kr-87	Rb-87	1.00	none	0.00	none	0.00
Kr-88	Rb-88	1.00	none	0.00	none	0.00
Sr-90	Y-90	1.00	none	0.00	none	0.00
Sr-91	Y-91m	0.58	Y-91	0.42	none	0.00
Sr-92	Y-92	1.00	none	0.00	none	0.00
Y-93	Zr-93	1.00	none	0.00	none	0.00
Zr-95	Nb-95m	0.01	Nb-95	0.99	none	0.00
Zr-97	Nb-97m	0.95	Nb-97	0.05	none	0.00
Mo-99	Tc-99m	0.88	Tc-99	0.12	none	0.00
Tc-99m	Tc-99	1.00	none	0.00	none	0.00
Ru-103	Rh-103m	1.00	none	0.00	none	0.00
Ru-105	Rh-105	1.00	none	0.00	none	0.00
Ru-106	Rh-106	1.00	none	0.00	none	0.00

Sb-127	Te-127m	0.18	Te-127	0.82	none	0.00
Sb-129	Te-129m	0.22	Te-129	0.77	none	0.00
Te-127m	Te-127	0.98	none	0.00	none	0.00
Te-129	I-129	1.00	none	0.00	none	0.00
Te-129m	Te-129	0.65	I-129	0.35	none	0.00
Te-131m	Te-131	0.22	I-131	0.78	none	0.00
Te-132	I-132	1.00	none	0.00	none	0.00
I-131	Xe-131m	0.01	none	0.00	none	0.00
I-133	Xe-133m	0.03	Xe-133	0.97	none	0.00
I-135	Xe-135m	0.15	Xe-135	0.85	none	0.00
Xe-135	Cs-135	1.00	none	0.00	none	0.00
Cs-137	Ba-137m	0.95	none	0.00	none	0.00
Ba-140	La-140	1.00	none	0.00	none	0.00
La-141	Ce-141	1.00	none	0.00	none	0.00
Ce-143	Pr-143	1.00	none	0.00	none	0.00
Ce-144	Pr-144m	0.02	Pr-144	0.98	none	0.00
Nd-147	Pm-147	1.00	none	0.00	none	0.00
Np-239	Pu-239	1.00	none	0.00	none	0.00
Pu-238	U-234	1.00	none	0.00	none	0.00
Pu-239	U-235	1.00	none	0.00	none	0.00
Pu-240	U-236	1.00	none	0.00	none	0.00
Pu-241	U-237	0.00	Am-241	1.00	none	0.00
Am-241	Np-237	1.00	none	0.00	none	0.00
Cm-242	Pu-238	1.00	none	0.00	none	0.00
Cm-244	Pu-240	1.00	none	0.00	none	0.00

Iodine fractions
Aerosol = 0.0000E+00
Elemental = 7.0000E-01
Organic = 3.0000E-01

COMPARTMENT DATA

Compartment number 1: Containment

Compartment number 2: Environment

Compartment number 3: Control Room

PATHWAY DATA

Pathway number 1: Leak to Environment

Pathway Filter: Removal Data

Time (hr)	Flow Rate (cfm)	Filter Efficiencies (%) Aerosol	Elemental	Organic
2.4000E+01	1.0000E+01	0.0000E+00	0.0000E+00	0.0000E+00
4.8000E+01	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00

Pathway number 2: Environment to Control Room

Pathway Filter: Removal Data

Time (hr)	Flow Rate (cfm)	Filter Efficiencies (%) Aerosol	Elemental	Organic
2.4000E+01	1.2600E+05	0.0000E+00	0.0000E+00	0.0000E+00
4.8000E+01	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00

Pathway number 3: Control Room to Environment Exhaust

```
     Pathway Filter: Removal Data

       Time (hr)      Flow Rate         Filter Efficiencies (%)
                        (cfm)        Aerosol      Elemental     Organic
       2.4000E+01    1.2600E+05   1.0000E+02   1.0000E+02   1.0000E+02
       4.8000E+01    0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00

LOCATION DATA
     Location Exclusion Area Bndry     is in compartment   2

     Location X/Q Data
      Time (hr)          X/Q (s * m^-3)
       2.4000E+01           3.1800E-04
       4.8000E+01           0.0000E+00

     Location Breathing Rate Data
      Time (hr)          Breathing Rate (m^3 * sec^-1)
       2.4000E+01                   3.5000E-04
       4.8000E+01                   0.0000E+00
     Location Low Population Zone     is in compartment   2

     Location X/Q Data
      Time (hr)          X/Q (s * m^-3)
       2.4000E+01           1.1500E-04
       4.8000E+01           0.0000E+00

     Location Breathing Rate Data
      Time (hr)          Breathing Rate (m^3 * sec^-1)
       2.4000E+01                   3.5000E-04
       4.8000E+01                   0.0000E+00
     Location Control Room     is in compartment   3

     Location X/Q Data
      Time (hr)          X/Q (s * m^-3)
       2.4000E+01           1.2600E-03
       4.8000E+01           0.0000E+00

     Location Breathing Rate Data
      Time (hr)          Breathing Rate (m^3 * sec^-1)
       2.4000E+01                   3.5000E-04
       4.8000E+01                   0.0000E+00

     Location Occupancy Factor Data
      Time (hr)          Occupancy Factor
       2.4000E+01             1.0000E+00
       4.8000E+01             0.0000E+00

USER SPECIFIED TIME STEP DATA - SUPPLEMENTAL TIME STEPS
     Time               Time step
      0.0000E+00          1.0000E-03
      1.0000E-02          1.0000E-02
      1.0000E-01          1.0000E-01
      2.0000E+00          1.0000E+00
      2.4000E+01          0.0000E+00
```

```
########################################################################
   RADTRAD Version 3.03 (Spring 2001) run on  9/23/2005  at  9:48:13
########################################################################



 ####   #    #  #####  #####  #    #  #####
#    #  #    #    #    #    # #    #    #
#    #  #    #    #    #    # #    #    #
#    #  #    #    #    #####  #    #    #
#    #  #    #    #    #      #    #    #
#    #  #    #    #    #      #    #    #
 ####    ####     #    #       ####     #


#####################################################################
                              Dose Output
#####################################################################

Exclusion Area Bndry Doses:

Time (h) =  24.0000    Whole Body   Thyroid       TEDE
Delta dose (rem)        1.2878E-06  1.0671E-04  4.5732E-06
Accumulated dose (rem)  1.2878E-06  1.0671E-04  4.5732E-06

Low Population Zone Doses:

Time (h) =  24.0000    Whole Body   Thyroid       TEDE
Delta dose (rem)        4.6570E-07  3.8589E-05  1.6538E-06
Accumulated dose (rem)  4.6570E-07  3.8589E-05  1.6538E-06

Control Room Doses:

Time (h) =  24.0000    Whole Body   Thyroid       TEDE
Delta dose (rem)        6.9103E-12  1.2684E-08  3.9743E-10
Accumulated dose (rem)  6.9103E-12  1.2684E-08  3.9743E-10

Exclusion Area Bndry Doses:

Time (h) =  48.0000    Whole Body   Thyroid       TEDE
Delta dose (rem)        4.2490E-01  3.5529E+01  1.5186E+00
Accumulated dose (rem)  4.2490E-01  3.5529E+01  1.5186E+00

Low Population Zone Doses:

Time (h) =  48.0000    Whole Body   Thyroid       TEDE
Delta dose (rem)        1.5366E-01  1.2848E+01  5.4918E-01
Accumulated dose (rem)  1.5366E-01  1.2849E+01  5.4918E-01

Control Room Doses:

Time (h) =  48.0000    Whole Body   Thyroid       TEDE
Delta dose (rem)        7.6986E-02  1.4274E+02  4.4708E+00
Accumulated dose (rem)  7.6986E-02  1.4274E+02  4.4708E+00
```

##
I-131 Summary
##

Time (hr)	Containment I-131 (Curies)	Environment I-131 (Curies)	Control Room I-131 (Curies)
24.000	2.6581E+02	7.9745E-04	5.9748E-05
24.400	2.4080E+01	2.4157E+02	2.0047E-01
24.700	3.9761E+00	2.6166E+02	3.3101E-02
25.000	6.5653E-01	2.6497E+02	5.4657E-03
25.300	1.0841E-01	2.6552E+02	9.0250E-04
25.600	1.7900E-02	2.6561E+02	1.4902E-04
25.900	2.9557E-03	2.6563E+02	2.4607E-05
26.200	4.8805E-04	2.6563E+02	4.0631E-06
26.500	8.0587E-05	2.6563E+02	6.7090E-07
26.800	1.3307E-05	2.6563E+02	1.1078E-07
27.100	2.1972E-06	2.6563E+02	1.8292E-08
27.400	3.6280E-07	2.6563E+02	3.0204E-09
27.700	5.9906E-08	2.6563E+02	4.9873E-10
28.000	9.8918E-09	2.6563E+02	8.2351E-11
28.300	1.6333E-09	2.6563E+02	1.3598E-11
28.600	2.6970E-10	2.6563E+02	2.2453E-12
28.900	4.4533E-11	2.6563E+02	3.7074E-13
29.200	7.3533E-12	2.6563E+02	6.1217E-14
29.500	1.2142E-12	2.6563E+02	1.0108E-14
29.800	2.0049E-13	2.6563E+02	1.6691E-15
30.100	3.3105E-14	2.6563E+02	2.7560E-16
30.400	5.4662E-15	2.6563E+02	4.5507E-17
30.700	9.0259E-16	2.6563E+02	7.5142E-18
31.000	1.4904E-16	2.6563E+02	1.2408E-18
31.300	2.4609E-17	2.6563E+02	2.0487E-19
31.600	4.0635E-18	2.6563E+02	3.3829E-20
31.900	6.7096E-19	2.6563E+02	5.5859E-21
32.200	1.1079E-19	2.6563E+02	9.2234E-22
32.500	1.8294E-20	2.6563E+02	1.5230E-22
32.800	3.0207E-21	2.6563E+02	2.5148E-23
33.100	4.9878E-22	2.6563E+02	4.1524E-24
33.400	8.2359E-23	2.6563E+02	6.8565E-25
33.700	1.3599E-23	2.6563E+02	1.1321E-25
34.000	2.2455E-24	2.6563E+02	1.8694E-26
34.300	3.7078E-25	2.6563E+02	3.0868E-27
48.000	7.0589E-61	2.6563E+02	5.8766E-63

##
Cumulative Dose Summary
##

	Exclusion Area Bndry		Low Population Zone		Control Room	
Time (hr)	Thyroid (rem)	TEDE (rem)	Thyroid (rem)	TEDE (rem)	Thyroid (rem)	TEDE (rem)
24.000	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
24.400	3.2315E+01	1.3822E+00	1.1686E+01	4.9985E-01	1.2817E+02	4.0148E+00
24.700	3.4999E+01	1.4962E+00	1.2657E+01	5.4109E-01	1.4033E+02	4.3956E+00
25.000	3.5442E+01	1.5149E+00	1.2817E+01	5.4785E-01	1.4234E+02	4.4584E+00
25.300	3.5515E+01	1.5180E+00	1.2843E+01	5.4896E-01	1.4267E+02	4.4688E+00
25.600	3.5527E+01	1.5185E+00	1.2848E+01	5.4914E-01	1.4273E+02	4.4705E+00
25.900	3.5529E+01	1.5186E+00	1.2848E+01	5.4917E-01	1.4274E+02	4.4707E+00
26.200	3.5529E+01	1.5186E+00	1.2848E+01	5.4918E-01	1.4274E+02	4.4708E+00
26.500	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
26.800	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00

27.100	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
27.400	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
27.700	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
28.000	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
28.300	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
28.600	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
28.900	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
29.200	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
29.500	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
29.800	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
30.100	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
30.400	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
30.700	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
31.000	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
31.300	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
31.600	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
31.900	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
32.200	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
32.500	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
32.800	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
33.100	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
33.400	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
33.700	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
34.000	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
34.300	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00
48.000	3.5529E+01	1.5186E+00	1.2849E+01	5.4918E-01	1.4274E+02	4.4708E+00

##

Worst Two-Hour Doses

##

Exclusion Area Bndry

Time (hr)	Whole Body (rem)	Thyroid (rem)	TEDE (rem)
24.0	4.2490E-01	3.5529E+01	1.5186E+00

```
########################################################################
   RADTRAD Version 3.03 (Spring 2001) run on  9/23/2005  at  9:50:17
########################################################################

########################################################################
                    File information
########################################################################

 Plant file           = P:\Users\Nuc\Exelon EOC\Discipline
Files\Process\AST\Limerick AST\LGS FHA\RADTRAD\LGS FHA 24hr Delay Test - With SGTS
Credit Rev1.psf
 Inventory file       = p:\users\nuc\exelon eoc\discipline
files\process\ast\limerick ast\lgs fha\radtrad\limerick ast source terms for
fha.nif
 Release file         = p:\users\nuc\exelon eoc\discipline
files\process\ast\limerick ast\lgs fha\radtrad\limerick ast fha.rft
 Dose Conversion file = c:\program files\radtrad3-03\defaults\fgr11&12.inp



     #####    ####   #####    #  #     #  #####   #    #  #####
     #    #  #       #        #  ##    #  #    #  #    #    #
     #    #  #       #        #  # #   #  #    #  #    #    #
     #####    ####   ####     #  #  #  #  #####   #    #    #
     #            #  #        #  #   # #  #       #    #    #
     #            #  #        #  #    ##  #       #    #    #
     #        ####   #        #  #     #  #        ####     #

 Radtrad 3.03 4/15/2001
 PBAPS FHA - TB/RB Ventilation Stack to CR Intake, EAB, & LPZ - 24 Hour Delay and
SGTS Filtration Credit
 Nuclide Inventory File:
 p:\users\nuc\exelon eoc\discipline files\process\ast\limerick ast\lgs
fha\radtrad\limerick ast source terms for fha.nif
 Plant Power Level:
  2.6830E+01
 Compartments:
    3
 Compartment 1:
 Containment
    3
  1.0000E+02
    0
    0
    0
    0
    0
 Compartment 2:
 Environment
    2
  0.0000E+00
    0
    0
    0
```

```
  0
  0
Compartment 3:
Control Room
  1
 1.2600E+05
  0
  0
  0
  0
  0
Pathways:
  3
Pathway 1:
Leak to Environment
  1
  2
  2
Pathway 2:
Environment to  Control Room
  2
  3
  2
Pathway 3:
Control Room to  Environment Exhaust
  3
  2
  2
End of Plant Model File
Scenario Description Name:

Plant Model Filename:
ACCEPT\TEST1.PMF
Source Term:
  1
  1   1.0000E+00
c:\program files\radtrad3-03\defaults\fgr11&12.inp
p:\users\nuc\exelon eoc\discipline files\process\ast\limerick ast\lgs
fha\radtrad\limerick ast fha.rft
  2.4000E+01
  1
  0.0000E+00   7.0000E-01   3.0000E-01   1.0000E+00
Overlying Pool:
  0
  0.0000E+00
  0
  0
  0
  0
Compartments:
  3
Compartment 1:
  0
  1
  0
  0
  0
  0
  0
  0
```

```
  0
Compartment 2:
  0
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 3:
  0
  1
  0
  0
  0
  0
  0
  0
  0
Pathways:
  3
Pathway 1:
  0
  0
  0
  0
  0
  1
  2
  2.4000E+01   1.0000E+01   9.9000E+01   9.9000E+01   9.9000E+01
  4.8000E+01   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 2:
  0
  0
  0
  0
  0
  1
  2
  2.4000E+01   1.2600E+05   0.0000E+00   0.0000E+00   0.0000E+00
  4.8000E+01   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 3:
  0
  0
  0
  0
```

```
  0
  1
  2
  2.4000E+01   1.2600E+05   1.0000E+02   1.0000E+02   1.0000E+02
  4.8000E+01   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Dose Locations:
  3
Location 1:
Exclusion Area Bndry
  2
  1
  2
  2.4000E+01   3.1800E-04
  4.8000E+01   0.0000E+00
  1
  2
  2.4000E+01   3.5000E-04
  4.8000E+01   0.0000E+00
  0
Location 2:
Low Population Zone
  2
  1
  2
  2.4000E+01   1.1500E-04
  4.8000E+01   0.0000E+00
  1
  2
  2.4000E+01   3.5000E-04
  4.8000E+01   0.0000E+00
  0
Location 3:
Control Room
  3
  0
  1
  2
  2.4000E+01   3.5000E-04
  4.8000E+01   0.0000E+00
  1
  2
  2.4000E+01   1.0000E+00
  4.8000E+01   0.0000E+00
Effective Volume Location:
  1
  2
  2.4000E+01   6.8800E-03
  4.8000E+01   0.0000E+00
Simulation Parameters:
  5
  2.4000E+01   1.0000E-03
  2.4010E+01   1.0000E-02
  2.4100E+01   1.0000E-01
  2.6000E+01   1.0000E+00
```

```
   4.8000E+01   0.0000E+00
 Output Filename:
 P:\Users\Nuc\Exelon EOC\Discipline Files\Process\AST\Limerick AST\LGS
FHA\RADTRAD\LGS FHA 24hr Delay Test - With SGTS Credit Rev1.o0
   1
   1
   1
   0
   0
 End of Scenario File
```

```
#######################################################################
   RADTRAD Version 3.03 (Spring 2001) run on  9/23/2005  at  9:50:17
#######################################################################

#######################################################################
                    Plant Description
#######################################################################

Number of Nuclides =  60

Inventory Power =   1.0000E+00 MWth
Plant Power Level =   2.6830E+01 MWth

Number of compartments    =   3

Compartment information

Compartment number  1  (Source term fraction =   1.0000E+00
)
Name: Containment
Compartment volume =   1.0000E+02 (Cubic feet)
Compartment type is Normal
Pathways into and out of compartment  1
   Exit Pathway Number    1: Leak to Environment

Compartment number  2
Name: Environment
Compartment type is Environment
Pathways into and out of compartment  2
   Inlet Pathway Number   1: Leak to Environment
   Inlet Pathway Number   3: Control Room to  Environment Exhaust
   Exit Pathway Number    2: Environment to  Control Room

Compartment number  3
Name: Control Room
Compartment volume =   1.2600E+05 (Cubic feet)
Compartment type is Control Room
Pathways into and out of compartment  3
   Inlet Pathway Number   2: Environment to  Control Room
   Exit Pathway Number    3: Control Room to  Environment Exhaust

Total number of pathways =   3
```

```
#########################################################################
   RADTRAD Version 3.03 (Spring 2001) run on  9/23/2005  at  9:50:17
#########################################################################

#########################################################################
                     Scenario Description
#########################################################################


Time between shutdown and first release =   2.4000E+01 (Hours)

Radioactive Decay is enabled
Calculation of Daughters is enabled

Release Fractions and Timings
               GAP        EARLY IN-VESSEL   LATE RELEASE     RELEASE MASS
             0.000001 hr    0.0000 hrs      0.0000 hrs          (gm)
NOBLES       5.0000E-02    0.0000E+00      0.0000E+00        3.108E+00
IODINE       2.5000E-04    0.0000E+00      0.0000E+00        2.795E-03
CESIUM       0.0000E+00    0.0000E+00      0.0000E+00        0.000E+00
TELLURIUM    0.0000E+00    0.0000E+00      0.0000E+00        0.000E+00
STRONTIUM    0.0000E+00    0.0000E+00      0.0000E+00        0.000E+00
BARIUM       0.0000E+00    0.0000E+00      0.0000E+00        0.000E+00
RUTHENIUM    0.0000E+00    0.0000E+00      0.0000E+00        0.000E+00
CERIUM       0.0000E+00    0.0000E+00      0.0000E+00        0.000E+00
LANTHANUM    0.0000E+00    0.0000E+00      0.0000E+00        0.000E+00

Inventory Power =        27. MWt

Nuclide  Group  Specific      half     Whole Body    Inhaled      Inhaled
 Name           Inventory     life        DCF        Thyroid     Effective
                (Ci/MWt)      (s)    (Sv-m3/Bq-s)   (Sv/Bq)      (Sv/Bq)
 Kr-85     1   7.892E+02  3.383E+08   1.190E-16   0.000E+00   0.000E+00
 Kr-85m    1   8.313E+03  1.613E+04   7.480E-15   0.000E+00   0.000E+00
 Kr-87     1   1.633E+04  4.578E+03   4.120E-14   0.000E+00   0.000E+00
 Kr-88     1   2.303E+04  1.022E+04   1.020E-13   0.000E+00   0.000E+00
 I-131     2   4.299E+04  6.947E+05   1.820E-14   2.920E-07   8.890E-09
 I-132     2   3.881E+04  8.280E+03   1.120E-13   1.740E-09   1.030E-10
 I-133     2   5.556E+04  7.488E+04   2.940E-14   4.860E-08   1.580E-09
 I-134     2   6.165E+04  3.156E+03   1.300E-13   2.880E-10   3.550E-11
 I-135     2   5.192E+04  2.380E+04   8.294E-14   8.460E-09   3.320E-10
 Xe-133    1   5.491E+04  4.532E+05   1.560E-15   0.000E+00   0.000E+00
 Xe-135    1   2.228E+04  3.272E+04   1.190E-14   0.000E+00   0.000E+00

Nuclide    Daughter  Fraction    Daughter  Fraction    Daughter  Fraction
 Kr-85m    Kr-85      0.21        none       0.00        none       0.00
 Kr-87     Rb-87      1.00        none       0.00        none       0.00
 Kr-88     Rb-88      1.00        none       0.00        none       0.00
 Sr-90     Y-90       1.00        none       0.00        none       0.00
 Sr-91     Y-91m      0.58        Y-91       0.42        none       0.00
 Sr-92     Y-92       1.00        none       0.00        none       0.00
 Y-93      Zr-93      1.00        none       0.00        none       0.00
 Zr-95     Nb-95m     0.01        Nb-95      0.99        none       0.00
 Zr-97     Nb-97m     0.95        Nb-97      0.05        none       0.00
 Mo-99     Tc-99m     0.88        Tc-99      0.12        none       0.00
 Tc-99m    Tc-99      1.00        none       0.00        none       0.00
 Ru-103    Rh-103m    1.00        none       0.00        none       0.00
 Ru-105    Rh-105     1.00        none       0.00        none       0.00
 Ru-106    Rh-106     1.00        none       0.00        none       0.00
```

Sb-127	Te-127m	0.18	Te-127	0.82	none	0.00
Sb-129	Te-129m	0.22	Te-129	0.77	none	0.00
Te-127m	Te-127	0.98	none	0.00	none	0.00
Te-129	I-129	1.00	none	0.00	none	0.00
Te-129m	Te-129	0.65	I-129	0.35	none	0.00
Te-131m	Te-131	0.22	I-131	0.78	none	0.00
Te-132	I-132	1.00	none	0.00	none	0.00
I-131	Xe-131m	0.01	none	0.00	none	0.00
I-133	Xe-133m	0.03	Xe-133	0.97	none	0.00
I-135	Xe-135m	0.15	Xe-135	0.85	none	0.00
Xe-135	Cs-135	1.00	none	0.00	none	0.00
Cs-137	Ba-137m	0.95	none	0.00	none	0.00
Ba-140	La-140	1.00	none	0.00	none	0.00
La-141	Ce-141	1.00	none	0.00	none	0.00
Ce-143	Pr-143	1.00	none	0.00	none	0.00
Ce-144	Pr-144m	0.02	Pr-144	0.98	none	0.00
Nd-147	Pm-147	1.00	none	0.00	none	0.00
Np-239	Pu-239	1.00	none	0.00	none	0.00
Pu-238	U-234	1.00	none	0.00	none	0.00
Pu-239	U-235	1.00	none	0.00	none	0.00
Pu-240	U-236	1.00	none	0.00	none	0.00
Pu-241	U-237	0.00	Am-241	1.00	none	0.00
Am-241	Np-237	1.00	none	0.00	none	0.00
Cm-242	Pu-238	1.00	none	0.00	none	0.00
Cm-244	Pu-240	1.00	none	0.00	none	0.00

Iodine fractions
- Aerosol = 0.0000E+00
- Elemental = 7.0000E-01
- Organic = 3.0000E-01

COMPARTMENT DATA

Compartment number 1: Containment

Compartment number 2: Environment

Compartment number 3: Control Room

PATHWAY DATA

Pathway number 1: Leak to Environment

Pathway Filter: Removal Data

Time (hr)	Flow Rate (cfm)	Filter Efficiencies (%) Aerosol	Elemental	Organic
2.4000E+01	1.0000E+01	9.9000E+01	9.9000E+01	9.9000E+01
4.8000E+01	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00

Pathway number 2: Environment to Control Room

Pathway Filter: Removal Data

Time (hr)	Flow Rate (cfm)	Filter Efficiencies (%) Aerosol	Elemental	Organic
2.4000E+01	1.2600E+05	0.0000E+00	0.0000E+00	0.0000E+00
4.8000E+01	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00

Pathway number 3: Control Room to Environment Exhaust

```
     Pathway Filter: Removal Data

       Time (hr)      Flow Rate          Filter Efficiencies (%)
                        (cfm)       Aerosol      Elemental     Organic
       2.4000E+01    1.2600E+05   1.0000E+02   1.0000E+02   1.0000E+02
       4.8000E+01    0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00

LOCATION DATA
     Location Exclusion Area Bndry     is in compartment   2

     Location X/Q Data
      Time (hr)         X/Q (s * m^-3)
       2.4000E+01          3.1800E-04
       4.8000E+01          0.0000E+00

     Location Breathing Rate Data
      Time (hr)         Breathing Rate (m^3 * sec^-1)
       2.4000E+01                 3.5000E-04
       4.8000E+01                 0.0000E+00
     Location Low Population Zone     is in compartment   2

     Location X/Q Data
      Time (hr)         X/Q (s * m^-3)
       2.4000E+01          1.1500E-04
       4.8000E+01          0.0000E+00

     Location Breathing Rate Data
      Time (hr)         Breathing Rate (m^3 * sec^-1)
       2.4000E+01                 3.5000E-04
       4.8000E+01                 0.0000E+00
     Location Control Room     is in compartment   3

     Location X/Q Data
      Time (hr)         X/Q (s * m^-3)
       2.4000E+01          6.8800E-03
       4.8000E+01          0.0000E+00

     Location Breathing Rate Data
      Time (hr)         Breathing Rate (m^3 * sec^-1)
       2.4000E+01                 3.5000E-04
       4.8000E+01                 0.0000E+00

     Location Occupancy Factor Data
      Time (hr)         Occupancy Factor
       2.4000E+01            1.0000E+00
       4.8000E+01            0.0000E+00

USER SPECIFIED TIME STEP DATA - SUPPLEMENTAL TIME STEPS
     Time               Time step
      0.0000E+00          1.0000E-03
      1.0000E-02          1.0000E-02
      1.0000E-01          1.0000E-01
      2.0000E+00          1.0000E+00
      2.4000E+01          0.0000E+00
```

```
#########################################################################
   RADTRAD Version 3.03 (Spring 2001) run on  9/23/2005  at  9:50:17
#########################################################################



     ####   #    #  #####  #####   #    #  #####
    #    #  #    #    #    #    #  #    #    #
    #    #  #    #    #    #    #  #    #    #
    #    #  #    #    #    #####   #    #    #
    #    #  #    #    #    #       #    #    #
    #    #  #    #    #    #       #    #    #
     ####    ####     #    #        ####     #


######################################################################
                              Dose Output
######################################################################

Exclusion Area Bndry Doses:

Time (h) =  24.0000    Whole Body   Thyroid       TEDE
Delta dose (rem)         1.1609E-06  1.0671E-06  1.1938E-06
Accumulated dose (rem)   1.1609E-06  1.0671E-06  1.1938E-06

Low Population Zone Doses:

Time (h) =  24.0000    Whole Body   Thyroid       TEDE
Delta dose (rem)         4.1982E-07  3.8589E-07  4.3170E-07
Accumulated dose (rem)   4.1982E-07  3.8589E-07  4.3170E-07

Control Room Doses:

Time (h) =  24.0000    Whole Body   Thyroid       TEDE
Delta dose (rem)         3.4016E-11  6.9257E-10  5.5339E-11
Accumulated dose (rem)   3.4016E-11  6.9257E-10  5.5339E-11

Exclusion Area Bndry Doses:

Time (h) =  48.0000    Whole Body   Thyroid       TEDE
Delta dose (rem)         3.8384E-01  3.5529E-01  3.9478E-01
Accumulated dose (rem)   3.8384E-01  3.5529E-01  3.9478E-01

Low Population Zone Doses:

Time (h) =  48.0000    Whole Body   Thyroid       TEDE
Delta dose (rem)         1.3881E-01  1.2848E-01  1.4276E-01
Accumulated dose (rem)   1.3881E-01  1.2849E-01  1.4277E-01

Control Room Doses:

Time (h) =  48.0000    Whole Body   Thyroid       TEDE
Delta dose (rem)         3.7983E-01  7.7939E+00  6.1975E-01
Accumulated dose (rem)   3.7983E-01  7.7939E+00  6.1975E-01

    161
```

Limerick FHA 24 hr Delay – With SGTS Filter Credit and North Stack CR X/Q

##
I-131 Summary
##

Time (hr)	Containment I-131 (Curies)	Environment I-131 (Curies)	Control Room I-131 (Curies)
24.000	2.6581E+02	7.9745E-06	3.2625E-06
24.400	2.4080E+01	2.4157E+00	1.0946E-02
24.700	3.9761E+00	2.6166E+00	1.8074E-03
25.000	6.5653E-01	2.6497E+00	2.9845E-04
25.300	1.0841E-01	2.6552E+00	4.9280E-05
25.600	1.7900E-02	2.6561E+00	8.1371E-06
25.900	2.9557E-03	2.6563E+00	1.3436E-06
26.200	4.8805E-04	2.6563E+00	2.2186E-07
26.500	8.0587E-05	2.6563E+00	3.6633E-08
26.800	1.3307E-05	2.6563E+00	6.0489E-09
27.100	2.1972E-06	2.6563E+00	9.9880E-10
27.400	3.6280E-07	2.6563E+00	1.6492E-10
27.700	5.9906E-08	2.6563E+00	2.7232E-11
28.000	9.8918E-09	2.6563E+00	4.4966E-12
28.300	1.6333E-09	2.6563E+00	7.4248E-13
28.600	2.6970E-10	2.6563E+00	1.2260E-13
28.900	4.4533E-11	2.6563E+00	2.0244E-14
29.200	7.3533E-12	2.6563E+00	3.3427E-15
29.500	1.2142E-12	2.6563E+00	5.5194E-16
29.800	2.0049E-13	2.6563E+00	9.1137E-17
30.100	3.3105E-14	2.6563E+00	1.5049E-17
30.400	5.4662E-15	2.6563E+00	2.4848E-18
30.700	9.0259E-16	2.6563E+00	4.1030E-19
31.000	1.4904E-16	2.6563E+00	6.7749E-20
31.300	2.4609E-17	2.6563E+00	1.1187E-20
31.600	4.0635E-18	2.6563E+00	1.8472E-21
31.900	6.7096E-19	2.6563E+00	3.0501E-22
32.200	1.1079E-19	2.6563E+00	5.0363E-23
32.500	1.8294E-20	2.6563E+00	8.3160E-24
32.800	3.0207E-21	2.6563E+00	1.3731E-24
33.100	4.9878E-22	2.6563E+00	2.2673E-25
33.400	8.2359E-23	2.6563E+00	3.7439E-26
33.700	1.3599E-23	2.6563E+00	6.1819E-27
34.000	2.2455E-24	2.6563E+00	1.0208E-27
34.300	3.7078E-25	2.6563E+00	1.6855E-28
48.000	7.0589E-61	2.6563E+00	3.2088E-64

##
Cumulative Dose Summary
##

Time	Exclusion Area Bndry		Low Population Zone		Control Room	
	Thyroid	TEDE	Thyroid	TEDE	Thyroid	TEDE
(hr)	(rem)	(rem)	(rem)	(rem)	(rem)	(rem)
24.000	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
24.400	3.2315E-01	3.5971E-01	1.1686E-01	1.3008E-01	6.9983E+00	5.5719E-01
24.700	3.4999E-01	3.8908E-01	1.2657E-01	1.4070E-01	7.6627E+00	6.0952E-01
25.000	3.5442E-01	3.9385E-01	1.2817E-01	1.4243E-01	7.7723E+00	6.1808E-01
25.300	3.5515E-01	3.9463E-01	1.2843E-01	1.4271E-01	7.7903E+00	6.1947E-01
25.600	3.5527E-01	3.9475E-01	1.2848E-01	1.4276E-01	7.7933E+00	6.1970E-01
25.900	3.5529E-01	3.9477E-01	1.2848E-01	1.4276E-01	7.7938E+00	6.1974E-01
26.200	3.5529E-01	3.9478E-01	1.2848E-01	1.4277E-01	7.7939E+00	6.1975E-01
26.500	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
26.800	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01

27.100	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
27.400	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
27.700	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
28.000	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
28.300	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
28.600	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
28.900	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
29.200	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
29.500	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
29.800	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
30.100	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
30.400	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
30.700	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
31.000	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
31.300	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
31.600	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
31.900	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
32.200	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
32.500	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
32.800	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
33.100	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
33.400	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
33.700	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
34.000	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
34.300	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01
48.000	3.5529E-01	3.9478E-01	1.2849E-01	1.4277E-01	7.7939E+00	6.1975E-01

##
Worst Two-Hour Doses
##

Exclusion Area Bndry

Time (hr)	Whole Body (rem)	Thyroid (rem)	TEDE (rem)
24.0	3.8384E-01	3.5529E-01	3.9477E-01

```
Nuclide Inventory Name:  Source Terms per this calculation
 Limerick Generating Station (LGS) FHA AST - in Ci/MW
 Power Level:
  0.1000E+01
 Nuclides:
  60
 Nuclide 001:
 Co-58
   7
  0.6117120000E+07
  0.5800E+02
  0.1529E+03
 none     0.0000E+00
 none     0.0000E+00
 none     0.0000E+00
 Nuclide 002:
 Co-60
   7
  0.1663401096E+09
  0.6000E+02
  0.1830E+03
 none     0.0000E+00
 none     0.0000E+00
 none     0.0000E+00
 Nuclide 003:
 Kr-85
   1
  0.3382974720E+09
  0.8500E+02
  0.7892E+03                    2.0*LOCA Value for FHA
 none     0.0000E+00
 none     0.0000E+00
 none     0.0000E+00
 Nuclide 004:
 Kr-85m
   1
  0.1612800000E+05
  0.8500E+02
  0.8313E+04
 Kr-85    0.2100E+00
 none     0.0000E+00
 none     0.0000E+00
 Nuclide 005:
 Kr-87
   1
  0.4578000000E+04
  0.8700E+02
  0.1633E+05
 Rb-87    0.1000E+01
 none     0.0000E+00
 none     0.0000E+00
 Nuclide 006:
 Kr-88
   1
  0.1022400000E+05
  0.8800E+02
  0.2303E+05
 Rb-88    0.1000E+01
 none     0.0000E+00
```

```
none      0.0000E+00
Nuclide 007:
Rb-86
  3
 0.1612224000E+07
 0.8600E+02
 0.6518E+02
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 008:
Sr-89
  5
 0.4363200000E+07
 0.8900E+02
 0.2798E+05
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 009:
Sr-90
  5
 0.9189573120E+09
 0.9000E+02
 0.3178E+04
Y-90      0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 010:
Sr-91
  5
 0.3420000000E+05
 0.9100E+02
 0.3801E+05
Y-91m     0.5800E+00
Y-91      0.4200E+00
none      0.0000E+00
Nuclide 011:
Sr-92
  5
 0.9756000000E+04
 0.9200E+02
 0.4017E+05
Y-92      0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 012:
Y-90
  9
 0.2304000000E+06
 0.9000E+02
 0.3272E+04
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 013:
Y-91
  9
 0.5055264000E+07
```

```
 0.9100E+02
 0.3448E+05
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 014:
Y-92
  9
 0.1274400000E+05
 0.9200E+02
 0.4029E+05
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 015:
Y-93
  9
 0.3636000000E+05
 0.9300E+02
 0.4526E+05
Zr-93     0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 016:
Zr-95
  9
 0.5527872000E+07
 0.9500E+02
 0.4489E+05
Nb-95m    0.7000E-02
Nb-95     0.9900E+00
none      0.0000E+00
Nuclide 017:
Zr-97
  9
 0.6084000000E+05
 0.9700E+02
 0.4657E+05
Nb-97m    0.9500E+00
Nb-97     0.5300E-01
none      0.0000E+00
Nuclide 018:
Nb-95
  9
 0.3036960000E+07
 0.9500E+02
 0.4512E+05
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 019:
Mo-99
  7
 0.2376000000E+06
 0.9900E+02
 0.5078+05
Tc-99m    0.8800E+00
Tc-99     0.1200E+00
none      0.0000E+00
```

```
Nuclide 020:
Tc-99m
  7
 0.2167200000E+05
 0.9900E+02
 0.4447E+05
Tc-99    0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 021:
Ru-103
  7
 0.3393792000E+07
 0.1030E+03
 0.4202E+05
Rh-103m  0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 022:
Ru-105
  7
 0.1598400000E+05
 0.1050E+03
 0.2908E+05
Rh-105   0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 023:
Ru-106
  7
 0.3181248000E+08
 0.1060E+03
 0.1730E+05
Rh-106   0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 024:
Rh-105
  7
 0.1272960000E+06
 0.1050E+03
 0.2752E+05
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 025:
Sb-127
  4
 0.3326400000E+06
 0.1270E+03
 0.2896E+04
Te-127m  0.1800E+00
Te-127   0.8200E+00
none     0.0000E+00
Nuclide 026:
Sb-129
  4
 0.1555200000E+05
 0.1290E+03
```

```
 0.8638E+04
Te-129m  0.2200E+00
Te-129   0.7700E+00
none     0.0000E+00
Nuclide 027:
Te-127
  4
 0.3366000000E+05
 0.1270E+03
 0.2873E+04
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 028:
Te-127m
  4
 0.9417600000E+07
 0.1270E+03
 0.3855E+03
Te-127   0.9800E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 029:
Te-129
  4
 0.4176000000E+04
 0.1290E+03
 0.8501E+04
I-129    0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 030:
Te-129m
  4
 0.2903040000E+07
 0.1290E+03
 0.1267E+04
Te-129   0.6500E+00
I-129    0.3500E+00
none     0.0000E+00
Nuclide 031:
Te-131m
  4
 0.1080000000E+06
 0.1310E+03
 0.3869E+04
Te-131   0.2200E+00
I-131    0.7800E+00
none     0.0000E+00
Nuclide 032:
Te-132
  4
 0.2815200000E+06
 0.1320E+03
 0.3821E+05
I-132    0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 033:
```

```
I-131
  2
 0.6946560000E+06
 0.1310E+03
 0.4299E+05                   1.6*LOCA Value for FHA
Xe-131m  0.1100E-01
none     0.0000E+00
none     0.0000E+00
Nuclide 034:
I-132
  2
 0.8280000000E+04
 0.1320E+03
 0.3881E+05
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 035:
I-133
  2
 0.7488000000E+05
 0.1330E+03
 0.5556E+05
Xe-133m  0.2900E-01
Xe-133   0.9700E+00
none     0.0000E+00
Nuclide 036:
I-134
  2
 0.3156000000E+04
 0.1340E+03
 0.6165E+05
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 037:
I-135
  2
 0.2379600000E+05
 0.1350E+03
 0.5192E+05
Xe-135m  0.1500E+00
Xe-135   0.8500E+00
none     0.0000E+00
Nuclide 038:
Xe-133
  1
 0.4531680000E+06
 0.1330E+03
 0.5491E+05
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 039:
Xe-135
  1
 0.3272400000E+05
 0.1350E+03
 0.2228E+05
```

```
Cs-135   0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 040:
Cs-134
  3
 0.6507177120E+08
 0.1340E+03
 0.7280E+04
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 041:
Cs-136
  3
 0.1131840000E+07
 0.1360E+03
 0.2027E+04
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 042:
Cs-137
  3
 0.9467280000E+09
 0.1370E+03
 0.4538E+04
Ba-137m  0.9500E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 043:
Ba-139
  6
 0.4962000000E+04
 0.1390E+03
 0.5084E+05
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 044:
Ba-140
  6
 0.1100736000E+07
 0.1400E+03
 0.4896E+05
La-140   0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 045:
La-140
  9
 0.1449792000E+06
 0.1400E+03
 0.5019E+05
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 046:
La-141
```

```
  9
 0.1414800000E+05
 0.1410E+03
 0.4640E+05
Ce-141    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 047:
La-142
  9
 0.5550000000E+04
 0.1420E+03
 0.4532E+05
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 048:
Ce-141
  8
 0.2808086400E+07
 0.1410E+03
 0.4492E+05
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 049:
Ce-143
  8
 0.1188000000E+06
 0.1430E+03
 0.4427E+05
Pr-143    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 050:
Ce-144
  8
 0.2456352000E+08
 0.1440E+03
 0.3596E+05
Pr-144m   0.1800E-01
Pr-144    0.9800E+00
none      0.0000E+00
Nuclide 051:
Pr-143
  9
 0.1171584000E+07
 0.1430E+03
 0.4293E+05
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 052:
Nd-147
  9
 0.9486720000E+06
 0.1470E+03
 0.1838E+05
Pm-147    0.1000E+01
```

```
none     0.0000E+00
none     0.0000E+00
Nuclide 053:
Np-239
  8
 0.2034720000E+06
 0.2390E+03
 0.5397E+06
Pu-239   0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 054:
Pu-238
  8
 0.2768863824E+10
 0.2380E+03
 0.1796E+03
U-234    0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 055:
Pu-239
  8
 0.7594336440E+12
 0.2390E+03
 0.1200E+02
U-235    0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 056:
Pu-240
  8
 0.2062920312E+12
 0.2400E+03
 0.1288E+02
U-236    0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 057:
Pu-241
  8
 0.4544294400E+09
 0.2410E+03
 0.6182E+04
U-237    0.2400E-04
Am-241   0.1000E+01
none     0.0000E+00
Nuclide 058:
Am-241
  9
 0.1363919472E+11
 0.2410E+03
 0.9528E+01
Np-237   0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 059:
Cm-242
  9
```

```
 0.1406592000E+08
 0.2420E+03
 0.2388E+04
Pu-238   0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 060:
Cm-244
  9
 0.5715081360E+09
 0.2440E+03
 0.2602E+03
Pu-240   0.1000E+01
none     0.0000E+00
none     0.0000E+00
End of Nuclear Inventory File
```

```
Release Fraction and Timing Name:
 Limerick Generating Station FHA, 8x8 bundle Pool I DF=200, Cs DF=infinity
 Duration (h):
  0.1000E-05  0.0000E+00  0.0000E+00  0.0000E+00
 Noble Gases:
  5.0000E-02  0.0000E+00  0.0000E+00  0.0000E+00
 Iodine:
  2.5000E-04  0.0000E+00  0.0000E+00  0.0000E+00
 Cesium:
  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
 Tellurium:
  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
 Strontium:
  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
 Barium:
  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
 Ruthenium:
  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
 Cerium:
  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
 Lanthanum:
  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
 Non-Radioactive Aerosols (kg):
  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
 End of Release File
```

LGS Fuel Handling Accident Assessment of Limiting Event

This Attachment benchmarks a fuel damage assessment approach as developed for GESTAR II by the General Electric Company for GE14 10x10 fuel dropping 34 feet over the open reactor vessel, and then applies it to the damage assessment of the same fuel and also an 8x8 fuel bundle for the corresponding drop over spent fuel stored in racks in the LGS Spent Fuel Pool (SFP). The results are then used in an Excel spreadsheet to show that the reactor vessel drop, with more than 23 feet of water covering the dropped and the struck fuel, bounds the results of the SFP drop, with less water coverage.

The analysis associated with the GE14 10x10 fuel is based on the NEDE-24011-P-A-US[1] analysis and the known result for a 34 ft drop, namely 172 broken rods. The general expression for the number of broken rods is:

$$\text{total broken rods} = \underbrace{\begin{matrix}\text{rods in}\\ \text{dropped}\\ \text{bundle}\end{matrix}}_{\text{dropped bundle}} + \underbrace{\frac{\underbrace{\text{drop height}\cdot(\text{bundle} + \text{mast weight})}_{\text{drop energy}}\cdot\underbrace{0.5}_{\text{fraction shared with impacted fuel}}\cdot\underbrace{\frac{\text{clad weight}}{\text{bundle weight - fuel weight}}}_{\text{fraction of energy shared between clad and fuel}}}{\underbrace{175\frac{ft-lbm}{rod}}_{\text{energy per rod failure}}}}_{\text{number broken due to initial impact}}$$

$$+\underbrace{\frac{\text{bundle length}\cdot(\text{mast weight} + 0.5\cdot\text{bundle weight})\cdot\underbrace{0.5}_{\text{fraction shared with impacted fuel}}\cdot\underbrace{\frac{\text{clad weight}}{\text{bundle weight - fuel weight}}}_{\text{fraction of energy shared between clad and fuel}}}{\underbrace{175\frac{ft-lbm}{rod}}_{\text{energy per rod failure}}}}_{\text{number broken due to secondary impact}}$$

where:

Drop Height	=34 ft[1]
Bundle Length	=160 in[1]
Mast Weight [Wet]	=619 lbs[1]
Bundle Weight [Wet]	=568 lbs[2]
Cladding Weight	=100.9 lbs[2]
Total assembly Weight [Dry]	=645lbs[2]
Total Pellet Weight	=455 lbs[3]
Energy per rod failure	=175 ft-lbm/rod[2]

Fraction of Energy Absorbed by Clad:

$$\frac{100.9\,lbm}{645\,lbm - 455\,lbm} = 0.531$$

Inserting the above values, one obtains the following:

$$\underbrace{92 rods}_{\text{dropped bundle}} + \underbrace{\frac{34 ft\cdot(619 lb + 568\,lbm)\cdot 0.5\cdot 0.531}{175\frac{ft-lbm}{rod}}}_{\text{initial impact}\approx 62\text{ rods}} + \underbrace{\frac{\frac{160 in}{12\frac{in}{ft}}\cdot(619 + 0.5\cdot 568)\cdot 0.5\cdot 0.531}{175\frac{ft-lbm}{rod}}}_{\text{secondary impact}\approx 18\text{ rods}} = 172 rods$$

---

[1] NEDE-24011-P-A-14-US, General Electric Standard Application for Reactor Fuel, Licensing Topical Report, June 2000
[2] Letter dated June 2, 2000 from J. Baumgartner, GNF Fuel Project Manager, to J. Carmody, Exelon
[3] GE Nuclear Energy Fuel Bundle Data Sheet 24A5424, Rev. 2

Thus, in the case of a 10x10 bundle the number of failed rods would be

92	from the impacting (dropped) assembly
62 (H/34)	from the impacted assemblies when H is the height of the drop
18	from the second impact.

Thus, the desired benchmark for the 34 foot drop height over the reactor core resulting in 172 failed rods is achieved.

The following analysis is for the same GE14 10x10 fuel dropped over the worst case SFP configuration. The drop height is conservatively considered as 2.33 feet for a drop from the Bottom of the Spent Fuel Assembly at Full Uplift to the Top of the Active Fuel in the Spent Fuel Rack, which from page E-7, rows 10 and 15 is 226.5 inches – 198.5 inches (the same drop to the page E-7 row 12 Top of Assembly Bail Handle in the Racks in the Spent Fuel Pool is only 9.2 inches). Again, the general expression for the number of broken rods is:

$$\text{total broken rods} = \underbrace{\begin{matrix}\text{rods in}\\ \text{dropped}\\ \text{bundle}\end{matrix}}_{\text{dropped bundle}} + \underbrace{\frac{\underbrace{\text{drop height}\cdot(\text{bundle}+\text{mast weight})}_{\text{drop energy}}\cdot\underbrace{0.5}_{\text{fraction shared with impacted fuel}}\cdot\underbrace{\frac{\text{clad weight}}{\text{bundle weight - fuel weight}}}_{\text{fraction of energy shared between clad and fuel}}}{\underbrace{175\frac{ft-lbm}{rod}}_{\text{energy per rod failure}}}}_{\text{number broken due to initial impact}}$$

$$+\underbrace{\frac{\text{bundle length}\cdot(\text{mast weight}+0.5\cdot\text{bundle weight})\cdot\underbrace{0.5}_{\text{fraction shared with impacted fuel}}\cdot\underbrace{\frac{\text{clad weight}}{\text{bundle weight - fuel weight}}}_{\text{fraction of energy shared between clad and fuel}}}{\underbrace{175\frac{ft-lbm}{rod}}_{\text{energy per rod failure}}}}_{\text{number broken due to secondary impact}}$$

where now:

Drop Height	=2.33 ft
Bundle Length	=160 in
Mast Weight [Wet]	=619 lbs
Bundle Weight [Wet]	=568 lbs
Cladding Weight	=100.9 lbs
Total assembly Weight [Dry]	=645lbs
Total Pellet Weight	=455 lbs
Energy per rod failure	=175 ft-lbm/rod

Fraction of Energy Absorbed by Clad:

$$\frac{100.9\,lbm}{645\,lbm-455\,lbm}=0.531$$

Inserting the above values, one obtains the following:

$$\underbrace{92rods}_{\text{dropped bundle}}+\underbrace{\frac{2.33ft\cdot(619lb+568\,lbm)\cdot 0.5\cdot 0.531}{175\frac{ft-lbm}{rod}}}_{\text{initial impact}\approx 4\text{ rods}}+\underbrace{\frac{\frac{160in}{12\frac{in}{ft}}\cdot(619+0.5\cdot 568)\cdot 0.5\cdot 0.531}{175\frac{ft-lbm}{rod}}}_{\text{secondary impact}\approx 18\text{ rods}}rods$$

Thus, in the case of a 10x10 bundle the number of failed rods from a worst case SFP drop would be

92	from the impacting (dropped) assembly
4	from the impacted assemblies
18	from the second impact

for a total of 114 10x10 rods damaged in the SFP.

A corresponding analysis is performed below for the GE 8x8 fuel (62 fuel rods and 2 water rods) addressed in LGS UFSAR Section 15.7.4 dropped over the worst case spent fuel pool configuration. The drop height is again conservatively considered as 2.33 feet for a drop from the Bottom of the Spent Fuel Assembly at Full Uplift to the Top of the Active Fuel in the Spent Fuel Rack. Using the general expression above for the number of broken rods with the energy per rod failure from LGS UFSAR Section 15.7.4.3.3.2 of 250 ft-lbm/rod,

$$\text{total broken rods} = \underbrace{\begin{array}{c}\text{rods in}\\ \text{dropped}\\ \text{bundle}\end{array}}_{\text{dropped bundle}} + \underbrace{\frac{\underbrace{\text{drop height}\cdot(\text{bundle}+\text{mast weight})}_{\text{drop energy}}\cdot \underbrace{0.5}_{\text{fraction shared with impacted fuel}}\cdot \underbrace{\frac{\text{clad weight}}{\text{bundle weight - fuel weight}}}_{\text{fraction of energy shared between clad and fuel}}}{\underbrace{250\frac{ft-lbm}{rod}}_{\text{energy per rod failure}}}}_{\text{number broken due to initial impact}}$$

$$+\underbrace{\frac{\text{bundle length}\cdot(\text{mast weight}+0.5\cdot\text{bundle weight})\cdot \underbrace{0.5}_{\text{fraction shared with impacted fuel}}\cdot \underbrace{\frac{\text{clad weight}}{\text{bundle weight - fuel weight}}}_{\text{fraction of energy shared between clad and fuel}}}{\underbrace{250\frac{ft-lbm}{rod}}_{\text{energy per rod failure}}}}_{\text{number broken due to secondary impact}}$$

where:

Drop Height =2.33 ft
Bundle Length =160 in
Mast Weight [Wet] =619 lbs
Total Assembly Weight [Dry] =678lbs for worst case (GE7/8 2 water rod) 8x8 fuel from PECO Nuclear Memorandum, Subject: GE7, GE8, and GE9 Bundle Masses, from V. K. Aggarwal to A, M. Olson, January 22, 1999.
Bundle Weight [Wet] =597 lbs using 0.88 wet/dry weight ratio of 10x10 fuel above

Fraction of Energy Absorbed by Clad = (mass of fuel cladding) / (mass of total assembly – mass of fuel) = 0.504 for worst case (GE9) 8x8 fuel from PECO Nuclear Memorandum, Subject: GE7, GE8, and GE9 Bundle Masses, from V. K. Aggarwal to A, M. Olson, January 22, 1999.

Inserting the above values, one obtains the following:

$$\underbrace{62rods}_{\text{dropped bundle}} + \underbrace{\frac{2.33ft\cdot(619lb+597\,lbm)\cdot 0.5\cdot 0.504}{250\frac{ft-lbm}{rod}}}_{\text{initial impact}\approx 2\text{ rods}} + \underbrace{\frac{\frac{160in}{12\frac{in}{ft}}\cdot(619+0.5\cdot 597)\cdot 0.5\cdot 0.504}{250\frac{ft-lbm}{rod}}}_{\text{secondary impact}\approx 12\text{ rods}} = 62+2+12rods$$

Thus, in the case of the worst case 8x8 bundle the number of failed rods for a SFP drop would be

62	from the impacting (dropped) assembly
+ 2	from the impacted assemblies
+ 12	from the second impact,

for a total Of 76 8x8 rods damaged.

Comparing the 114 damaged 10x10 rods to the 76 damaged 8x8 rods on a number of affected bundles basis, with 87.33 the effective number of rods in a 10x10 bundle and 62 rods in a 8x8 bundle, the 10x10 fuel bounds, as shown below:

10x10: 114/(87.33) = 1.305

8x8: 76/(62) = 1.226

	A	B	C	D	E	F	G	H	K	L	M
1	**LGS Fuel Handling Accident Assessment of Limiting Event**										
2	The balance of this Attachment:										
3	[a] Evaluates the reduced water coverage for dropped fuel lying horizontally on top										
4	of the racks and struck fuel in the racks for FHAs in the Spent Fuel Pool (SFP)										
5	compared to over the Reactor Well.										
6	[b] Evaluates impact of water coverages of 23 feet and less for determination of										
7	conservative SFP pool water Decontamination Factors (DF).										
8	[c] Justifies that a FHA over the Reactor Well is the limiting event.										
9											
10	**Baseline R.G. 1.183 based Analysis of DFs**										
11				RG 1.183	RG 1.183						
12	Water	RG 1.183	RG 1.183	Inorganic	Organic						
13	Coverage	Inorganic	Organic	Iodine	Iodine DF	Overall					
14	(feet)	Iodine DF	Iodine DF	Fraction	Fraction	DF					
15	23	500	1	0.9985	0.0015	286.0	**Inorganic Iodine DF Guidance Controlling**				
16	23	285.3	1	0.9985	0.0015	200.0	**Overall DF Guidance Controlling**				
17											
18											
19	**DFs determined per Burley Paper with Inorganic Iodine DF Guidance assumptions**										
20				RG 1.183	RG 1.183						
21	Water	RG 1.183	RG 1.183	Inorganic	Organic						
22	Coverage	Inorganic	Organic	Iodine	Iodine DF	Overall					
23	(feet)	Iodine DF	Iodine DF	Fraction	Fraction	DF					
24	23	500	1	0.9985	0.0015	286.0	capped at 200				
25	22.5	436.8	1	0.9985	0.0015	264.1	capped at 200				
26	22	381.6	1	0.9985	0.0015	242.9	capped at 200				
27	21.5	333.4	1	0.9985	0.0015	222.5	capped at 200				
28	21	291.3	1	0.9985	0.0015	202.9	capped at 200				
29	20.5	254.5	1	0.9985	0.0015	184.4					
30	20	222.3	1	0.9985	0.0015	166.9					
31	19.5	194.2	1	0.9985	0.0015	150.6					
32	19	169.7	1	0.9985	0.0015	135.4					
33	All water coverages are more than 21 feet. Therefore, the 200 DF is conservative for all cases.										
34											
35	**DFs determined per Burley Paper with Overall DF Guidance assumptions**										
36				RG 1.183	RG 1.183						
37	Water	RG 1.183	RG 1.183	Inorganic	Organic						
38	Coverage	Inorganic	Organic	Iodine	Iodine DF	Overall					
39	(feet)	Iodine DF	Iodine DF	Fraction	Fraction	DF					
40	23	285.3	1	0.9985	0.0015	200.0					
41	22.5	252.3	1	0.9985	0.0015	183.2					
42	22	223.1	1	0.9985	0.0015	167.4					
43	21.5	197.3	1	0.9985	0.0015	152.4					
44	21	174.5	1	0.9985	0.0015	138.5					
45	20.5	154.3	1	0.9985	0.0015	125.5					
46	20	136.5	1	0.9985	0.0015	113.4					
47	19.5	120.7	1	0.9985	0.0015	102.3					
48	19	106.7	1	0.9985	0.0015	92.1					
49	21.611	202.8	1	0.9985	0.0015	155.7	line 19 of page E-7; DF for dropped bundle				
50	22.612	259.4	1	0.9985	0.0015	186.9	line 21 of page E-7; DF for struck bundles				
51	Applying the appropriate DFs above for 92 10x10 rods damaged in the dropped bundle,										
52	and 22 10x10 rods damaged in the struck (in-rack) bundles (from page E-3):										
53				Overall weighted DF		**161.7**					
54	Resulting overall % of 200 DF for 23 feet coverage					80.8%					
55											
56	Fuel failure over SFP vs. over the reactor well = 114/172					66.3%					
57	Therefore, the drop over the reactor well is bounding when a 23 feet coverage DF of 200 is used.										

	A	B	C	D	E	F	G
1	**LGS Fuel**						
2	The balance of this Attachment:						
3	[a] Evaluates the reduced water coverage for dropped fuel lying horizontally on top of the racks and struck fuel in the racks for FHAs in the Sp						
4	Pool (SFP) compared to over the Reactor Well.						
5	[b] Evaluates impact of water coverages of 23 feet and less for determination of conservative SFP pool water Decontamination Factors (DF).						
6	[c] Justifies that a FHA over the Reactor Well is the limiting event.						
7							
8							
9							
10	**Baseline R.**						
11				RG 1.183	RG 1.183		
12	Water	RG 1.183	RG 1.183	Inorganic	Organic		
13	Coverage	Inorganic	Organic	Iodine	Iodine DF	Overall	
14	(feet)	Iodine DF	Iodine DF	Fraction	Fraction	DF	
15	23	500	1	0.9985	0.0015	**=1/(D15/B15+E15/C15)**	**Inorganic Iodine DF G**
16	23	285.3	1	0.9985	0.0015	**=1/(D16/B16+E16/C16)**	**Overall DF Guidance C**
17							
18							
19	**DFs determ**						
20				RG 1.183	RG 1.183		
21	Water	RG 1.183	RG 1.183	Inorganic	Organic		
22	Coverage	Inorganic	Organic	Iodine	Iodine DF	Overall	
23	(feet)	Iodine DF	Iodine DF	Fraction	Fraction	DF	
24	23	500	1	0.9985	0.0015	=1/(D24/B24+E24/C24)	capped at 200
25	=A24-0.5	=B$24^(A25/A$24)	1	0.9985	0.0015	=1/(D25/B25+E25/C25)	capped at 200
26	=A25-0.5	=B$24^(A26/A$24)	1	0.9985	0.0015	=1/(D26/B26+E26/C26)	capped at 200
27	=A26-0.5	=B$24^(A27/A$24)	1	0.9985	0.0015	=1/(D27/B27+E27/C27)	capped at 200
28	=A27-0.5	=B$24^(A28/A$24)	1	0.9985	0.0015	=1/(D28/B28+E28/C28)	capped at 200
29	=A28-0.5	=B$24^(A29/A$24)	1	0.9985	0.0015	=1/(D29/B29+E29/C29)	
30	=A29-0.5	=B$24^(A30/A$24)	1	0.9985	0.0015	=1/(D30/B30+E30/C30)	
31	=A30-0.5	=B$24^(A31/A$24)	1	0.9985	0.0015	=1/(D31/B31+E31/C31)	
32	=A31-0.5	=B$24^(A32/A$24)	1	0.9985	0.0015	=1/(D32/B32+E32/C32)	
33	All water cov						
34							
35	**DFs determ**						
36				RG 1.183	RG 1.183		
37	Water	RG 1.183	RG 1.183	Inorganic	Organic		
38	Coverage	Inorganic	Organic	Iodine	Iodine DF	Overall	
39	(feet)	Iodine DF	Iodine DF	Fraction	Fraction	DF	
40	23	285.3	1	0.9985	0.0015	=1/(D40/B40+E40/C40)	
41	=A40-0.5	=B$40^(A41/A$40)	1	0.9985	0.0015	=1/(D41/B41+E41/C41)	
42	=A41-0.5	=B$40^(A42/A$40)	1	0.9985	0.0015	=1/(D42/B42+E42/C42)	
43	=A42-0.5	=B$40^(A43/A$40)	1	0.9985	0.0015	=1/(D43/B43+E43/C43)	
44	=A43-0.5	=B$40^(A44/A$40)	1	0.9985	0.0015	=1/(D44/B44+E44/C44)	
45	=A44-0.5	=B$40^(A45/A$40)	1	0.9985	0.0015	=1/(D45/B45+E45/C45)	
46	=A45-0.5	=B$40^(A46/A$40)	1	0.9985	0.0015	=1/(D46/B46+E46/C46)	
47	=A46-0.5	=B$40^(A47/A$40)	1	0.9985	0.0015	=1/(D47/B47+E47/C47)	
48	=A47-0.5	=B$40^(A48/A$40)	1	0.9985	0.0015	=1/(D48/B48+E48/C48)	
49	21.611	=B$40^(A49/A$40)	1	0.9985	0.0015	=1/(D49/B49+E49/C49)	line 19 of page E-7; DF
50	22.6120833	=B$40^(A50/A$40)	1	0.9985	0.0015	=1/(D50/B50+E50/C50)	line 21 of page E-7; DF
51	Applying the appropriate DFs above for 92 10x10 rods damaged in the dropped bundle,						
52	and 22 10x10 rods damaged in the struck (in-rack) bundles (from page E-3):						
53					Overall weighted DF	**=(92*F49+(22)*F50)/114**	
54					200 DF for 23 feet coverage	=F53/200	
55							
56	Fuel failure c					0.663	
57	Therefore, th						

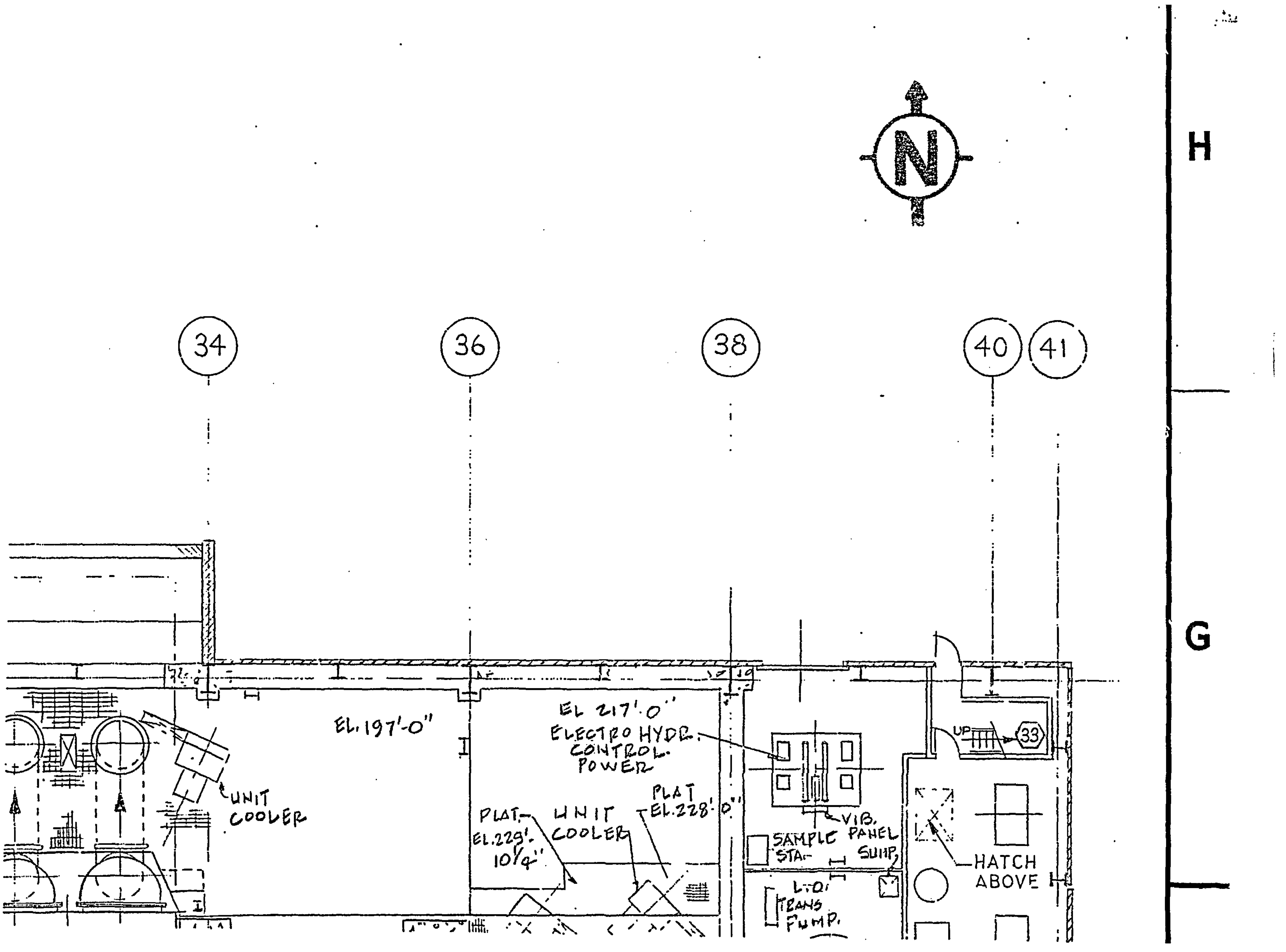
	A	B	C	D	E	F	G	H	K	L
1	**LGS Damaged Fuel Water Cover Assessment for Fuel Handling Accidents**									
2										
3	**Reference Points**									
4	**MSL**	**Water level**	**SPF Bottom**							
5	**(feet)**	**(inches)**	**(feet)**							
6	**352.000**	**506**	**39.250**	Refuel Floor						
7	**351.000**	**494**	**38.250**	Normal Spent Fuel Pool Water Level						
8	**350.000**	**482**	**37.250**	Tech Spec 3.9.8 water level for 22 ft above reactor pressure vessel flange						
9	**327.958**	**217.5**	**15.208**	Elevation at Reactor Vessel Flange						
10	**328.708**	**226.5**	**15.958**	Bottom of Spent Fuel Assembly at full uplift						
11	**328.389**	**222.7**	**15.639**	Top of Assembly Lying on Bail Handles						
12	**327.943**	**217.3**	**15.193**	Top of Assembly Bail Handle in Spent Fuel Pool						
13	**327.542**	**212.5**	**14.792**	Top of Fuel Rack Cell						
14	**327.388**	**210.7**	**14.638**	Top of Fuel Rod in Spent Fuel Rack						
15	**326.375**	**198.5**	**13.625**	Top of Active Fuel in Spent Fuel Rack						
16	**312.750**	**35.0**	**0.000**	Bottom of Spent Fuel Pool						
17	**296.417**	**-161.0**	**-16.333**	Top of Core						
18										
19	**21.611**	**Coverage over Assembly Lying on Bails With Water Level at T.S. Limit**								
20	**23.625**	**Coverage over Active Fuel With Water Level at T.S. Limit**								
21	**22.612**	**Coverage over Top of Fuel Rod With Water Level at T.S. Limit**								
22	**32.292**	**Drop Distance over Top of Core [Less than GESTAR 34 ft. drop assumption]**								
23	**1.167**	**Drop Distance over Spent Fuel Racks**								
24										
25	**References:**									
26		**Fuel Assembly Dimensions (inches) Based on GE DWG No. 829E431, Rev. 2**								
27		**as per LGS Drawing SDOC-340-H-VC-00021, Rev. 0**								
28	**176.16**	**Maximum Fuel Assembly Length**								
29	**5.348**	**Assembly thickness, lying on its side (from GE DWG No. 107E1593, Rev. 1) - average of 5.226**								
30		**Upper End Fitting width and 5.47 maximum overall bundle width**								
31										
32		**Technical Specification 3.9.8 LCO on Water Level - Reactor Vessel**								
33										
34		**UFSAR Section 9.1.2.2.2.1 for normal water level.**								
35										
36		**Drawing C-775 for bottom of pool elevation**								
37										
38		**GP-6,"Shutdown Operations - Refueling, Core Alteration and Core Off-**								
39		**loading", Attachment 4 for 9 inch distance from full up to top of vessel**								
40		**flange.**								

	A	B	C	D	E	F	H
1	LGS Damaged Fuel Wate						
2							
3		Reference Points					
4	MSL	Water level	SPF Bottom				
5	(feet)	(inches)	(feet)				
6	352	=B7+12	=C$10+(A6-A$10)	Refuel Floor			
7	351	494	=C$10+(A7-A$10)	Normal Spent Fuel Pool Water Level			
8	350	=B7-12	=C$10+(A8-A$10)	Tech Spec 3.9.8 water level for 22 feet a			
9	=A7-(B7-B9)/12	217.5	=C$10+(A9-A$10)	Elevation at Reactor Vessel Flange			
10	=A9+0.75	=B9+9	=C11+A10-A11	Bottom of Spent Fuel Assembly at full up			
11	=A$16+(C11-C$16)	=B$7-(A$7-A11)*12	=C12+$A29/12	Top of Assembly Lying on Bail Handles			
12	=A$16+(C12-C$16)	=B$7-(A$7-A12)*12	=182.32/12	Top of Assembly Bail Handle in Spent Fu			
13	=A$16+(C13-C$16)	=B$7-(A$7-A13)*12	=177.5/12	Top of Fuel Rack Cell			
14	=A$16+(C14-C$16)	=B$7-(A$7-A14)*12	=175.655/12	Top of Fuel Rod in Spent Fuel Rack			
15	=A$16+(C15-C$16)	=B$7-(A$7-A15)*12	=163.5/12	Top of Active Fuel in Spent Fuel Rack			
16	312.75	=B$7-(A$7-A16)*12	0	Bottom of Spent Fuel Pool			
17	=A16-(B16-B17)/12	-161	=A17-A16	Top of Core			
18							
19	=C8-C11	Coverage over Assembly Lyi					
20	=C8-C15	Coverage over Active Fuel W					
21	=C8-C14	Coverage over Top of Fuel R					
22	=A10-A17	Drop Distance over Top of C					
23	=A10-A13	Drop Distance over Spent Fu					
24							
25	References:						
26		Fuel Assembly Dimensions (					
27		as per LGS Drawing SDOC-3					
28	176.16	Maximum Fuel Assembly Le					
29	=5.226+(5.47-5.226)/2	Assembly thickness, lying o					
30		Upper End Fitting and 5.47 m					
31							
32		Technical Specification 3.9.8					
33							
34		UFSAR Section 9.1.2.2.2.1 fo					
35							
36		Drawing C-775 for bottom of					
37							
38		GP-6,"Shutdown Operations - Refueling, Core Alteration and Core Off-loading", Attachment 4 for 9 inch distance from full up to top of					
39		vessel flange.					
40							

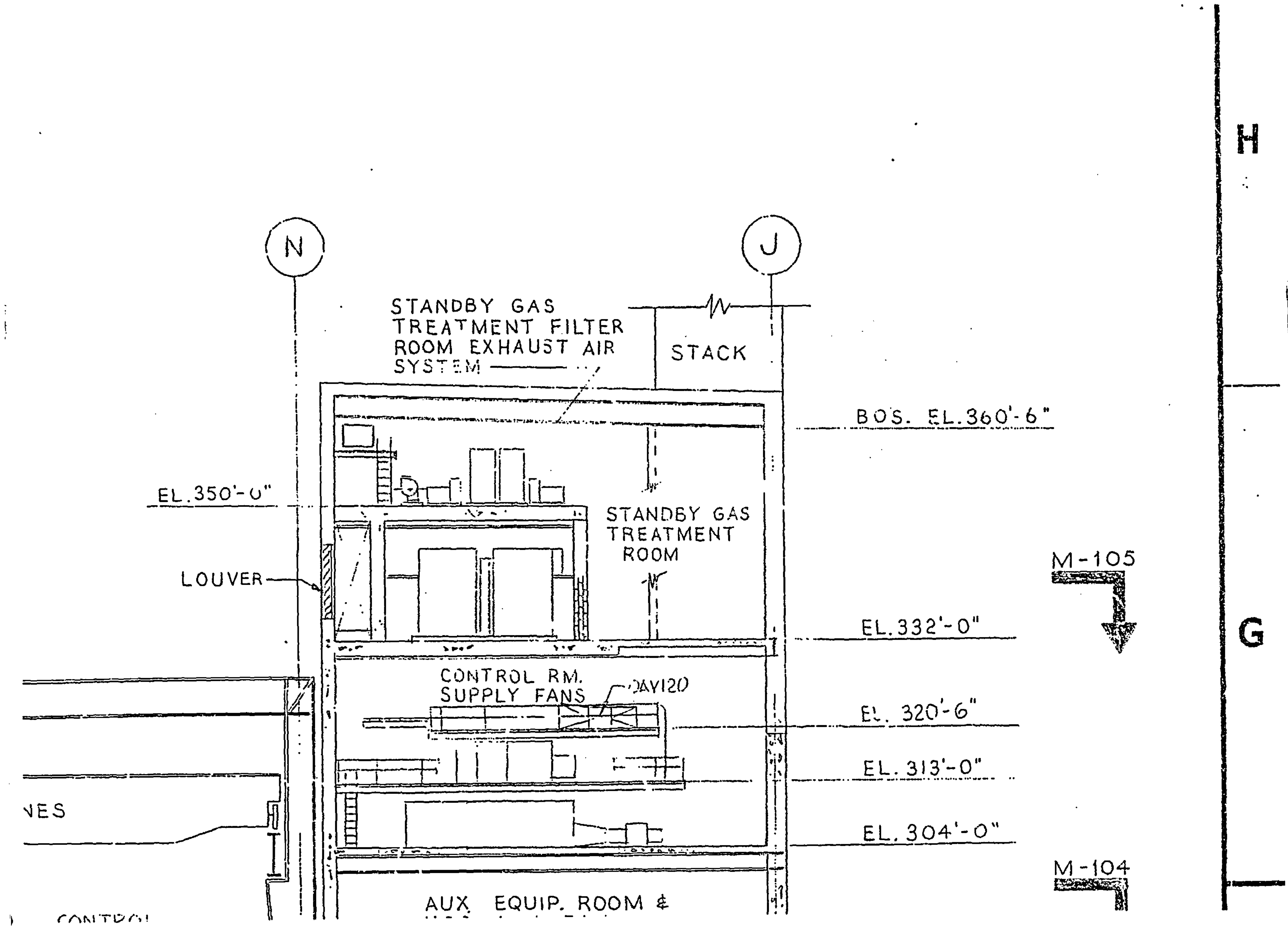
	A	B	C
1	**Attachment G: Evaluation of Bounds for Other Previously Considered FHA Scenarios**		
2			
3	Scenario		
4	LM-0656	LM-0657	Calculation Number
5	Crane	RCWP Jib	Scenario Descriptions
6	Collision	Crane Drop	
7	Fuel Array Basis GE14 (10x10), with 764 assemblies in core		
8	92	92	Total Rods / Assembly
9	87.33	87.33	Effective Rods / Assembly with 14 2/3 length rods
10	184	92	Fuel Rods Damaged in Dropped Assemblies (all rods)
11	155.7	200	Pool DF applicable to Dropped Assemblies (from Attachment E)
12	1.7	1.7	Peaking Factor
13	58	206	Fuel Rods Damaged from being struck by dropped assemblies or loads (limiting)
14	186.9	200	Pool DF applicable to Struck Assemblies (from Attachment E)
15	0.007604	0.007593	Damaged Core Fraction Multiplied by PF
16	**0.007608**		**New Dose Analysis Basis Damaged Core Fraction Multiplied by PF**
17			**Therefore, these scenarios are bounded by AST Dose Assessment.**
18			
19	Notes:		
20	1.	Fuel Damage Assessment are from the cited calculations unless otherwise indicated below. Both	
21		calculation use GESTAR II damage assessment methodology.	
22	2.	The worst case fuel damage for struck assemblies in LM-0656 is 42 rods. However, to determine	
23		the allowable margin for, e.g., future use of a Heavy Mast or fuel weight increases, a 58 rod	
24		damage assumption is used.	
25	3.	The worst case fuel damage for struck assemblies in LM-0657 is 154 rods with 68 damaged by	
26		the dropped assembly and 86 by the dropped Jib Crane and Load. As shown, a total of 206 fuel	
27		rods in Struck Assemblies can fail before the dose analysis basis is exceeded.	

	A	B
1	Attachment G: Evaluation of Bounds for Other Previously Con	
2		
3	Scenario	
4	LM-0656	LM-0657
5	Crane	RCWP Jib
6	Collision	Crane Drop
7	Fuel Array Basis GE14 (10x10), with 764 assemblie	
8	92	92
9	87.33	87.33
10	184	92
11	155.7	200
12	1.7	1.7
13	58	206
14	186.9	200
15	=(A10/A9/764)*A12*(200/A11)+(A13/A9/764)*A12*(200/A14)	=(B10/B9/764)*B12*(200/B11)+(B13/B9/764)*B12*(200/B14)
16	**0.007608**	
17		
18		
19	Notes:	
20	1.	Fuel Damage Assessment are from the cited calculations unless
21		methodology.
22	2.	The worst case fuel damage for struck assemblies in LM-0656 is
23		Heavy Mast or fuel weight increases, a 58 rod damage assumpti
24		
25	3.	The worst case fuel damage for struck assemblies in LM-0657 is
26		Crane and Load. As shown, a total of 206 fuel rods in Struck As
27		

	C
1	**sidered FHA Scenarios**
2	
3	
4	Calculation Number
5	Scenario Descriptions
6	
7	s in core
8	Total Rods / Assembly
9	Effective Rods / Assembly with 14 2/3 length rods
10	Fuel Rods Damaged in Dropped Assemblies (all rods)
11	Pool DF applicable to Dropped Assemblies (from Attachment E)
12	Peaking Factor
13	Fuel Rods Damaged from being struck by dropped assemblies or loads (limiting)
14	Pool DF applicable to Struck Assemblies (from Attachment E)
15	Damaged Core Fraction Multiplied by PF
16	**New Dose Analysis Basis Damaged Core Fraction Multiplied by PF**
17	**Therefore, these scenarios are bounded by AST Dose Assessment.**
18	
19	
20	otherwise indicated below. Both calculation use GESTAR II damage assessment
21	
22	42 rods. However, to determine the allowable margin for, e.g., future use of a
23	on is used.
24	
25	154 rods with 68 damaged by the dropped assembly and 86 by the dropped Jib
26	semblies can fail before the dose analysis basis is exceeded.
27	



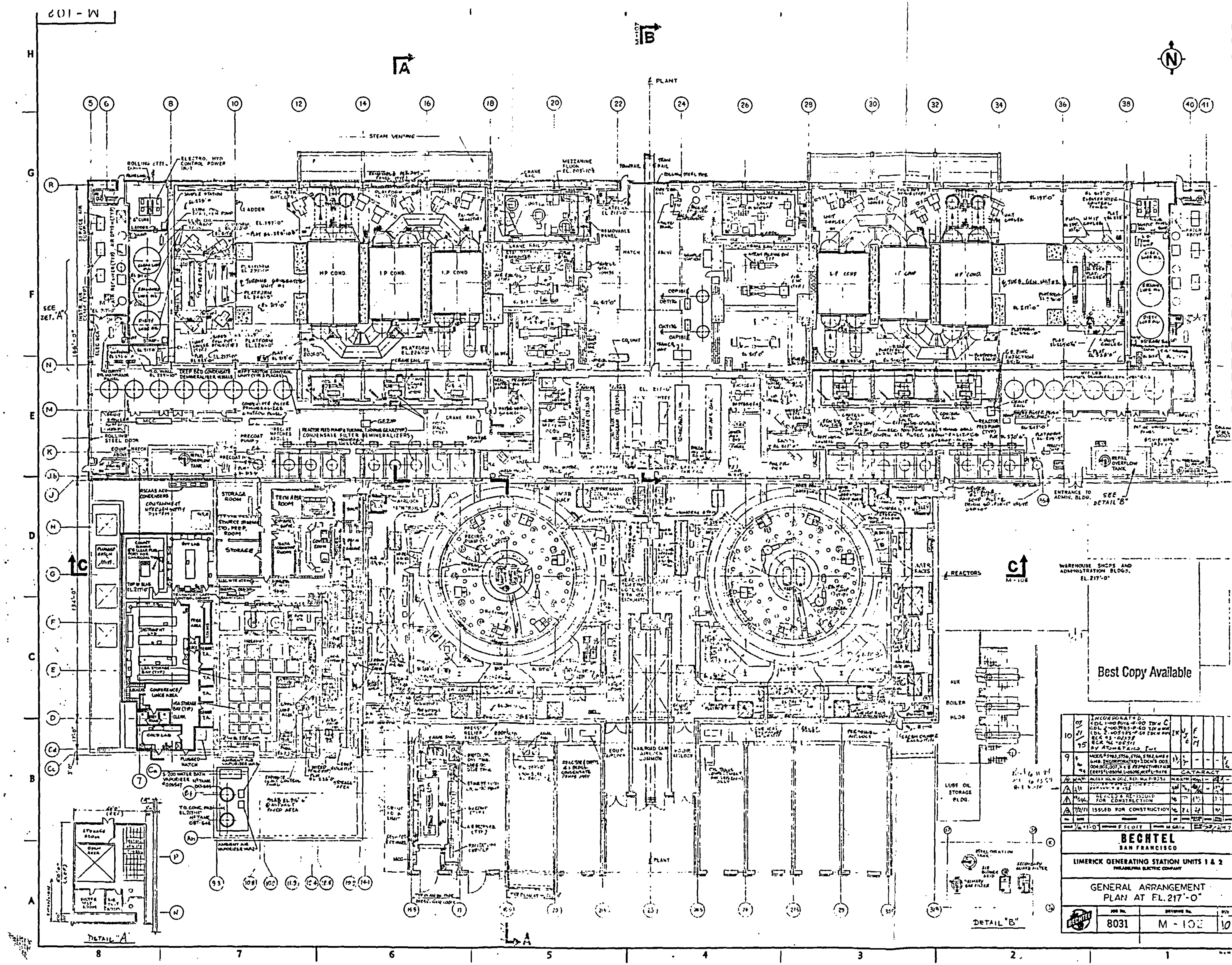
N
H
G
34
36
38
40
41
EL. 197'-0"
EL 217'.0"
ELECTRO HYDR.
CONTROL
POWER
UNIT
COOLER
PLAT.
EL. 229'.
10 1/4"
UNIT
COOLER
PLAT
EL. 228'.0"
VIB.
PANEL
SAMPLE
STA.
SUMP
LTD.
TRANS
PUMP
UP
33
HATCH
ABOVE



N
J
STANDBY GAS TREATMENT FILTER ROOM EXHAUST AIR SYSTEM
STACK
BOS. EL.360'-6"
EL.350'-0"
STANDBY GAS TREATMENT ROOM
M-105
LOUVER
EL.332'-0"
CONTROL RM. SUPPLY FANS
EL. 320'-6"
EL. 313'-0"
NES
EL. 304'-0"
M-104
AUX. EQUIP. ROOM &
H
G

ADDITIONAL ATTACHMENTS TO
10-10-05 Letter: Supplement to Request for LAR Application of AST

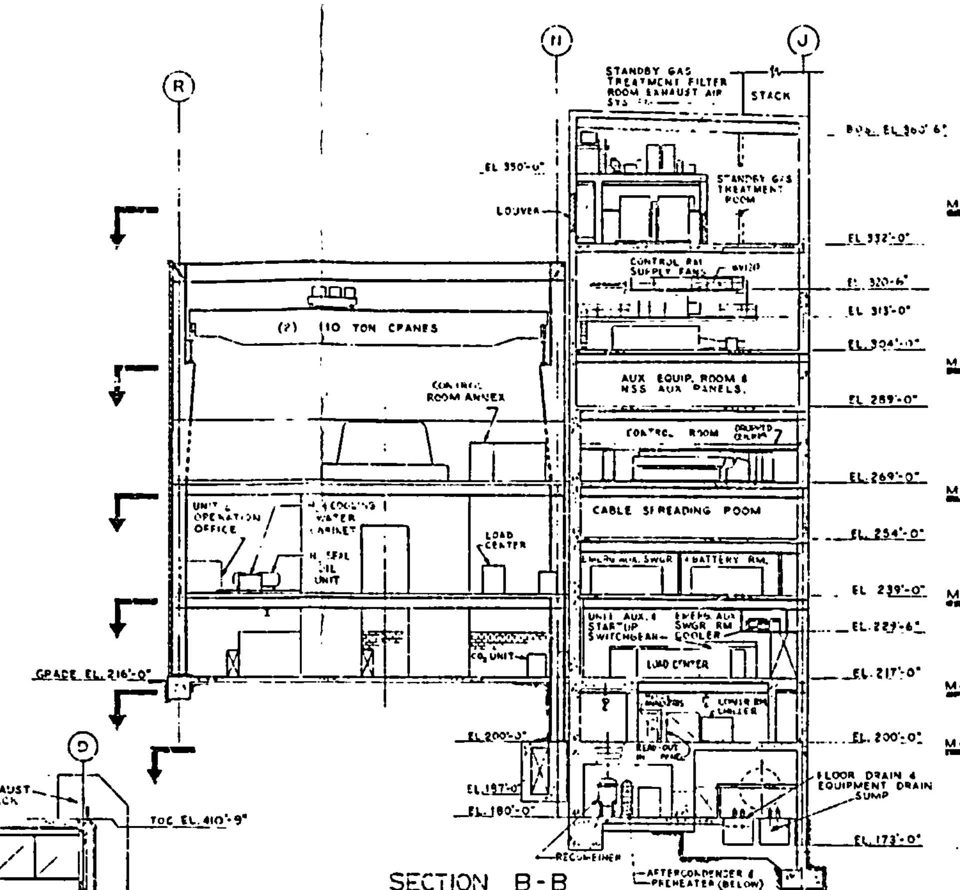
Attachment 010 AST – Drawing M-0102 (1 of 2).



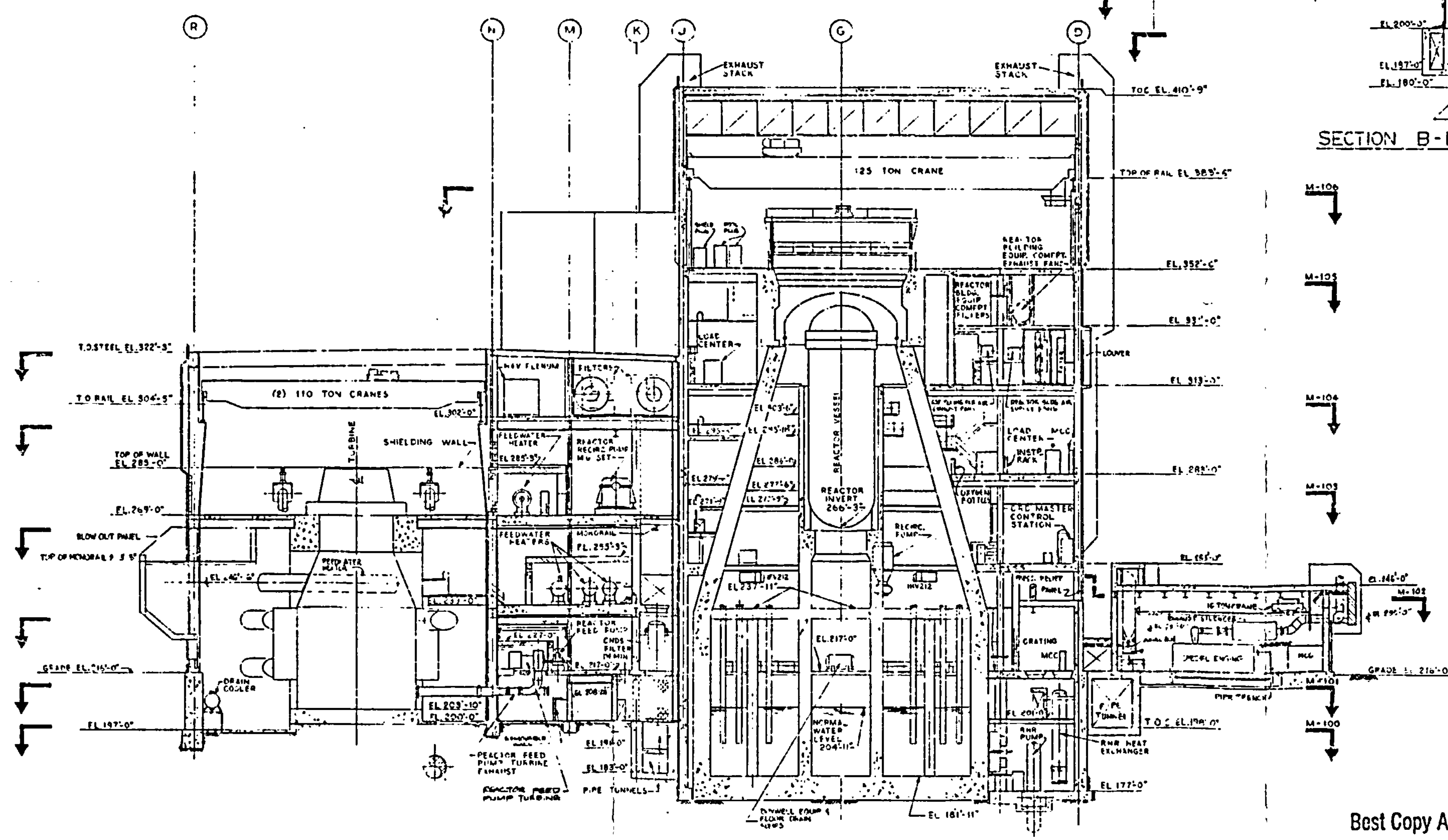
M-102
STEAM VENTING
PLANT
HP COND.
I.P. COND.
L.P. COND
REACTORS
WAREHOUSE SHOPS AND ADMINISTRATION BLDGS. EL. 217'-0"
ENTRANCE TO ADMIN. BLDG.
SEE DETAIL "B"
SEE DET. "A"
AUX BOILER BLDG
LUBE OIL STORAGE BLDG.
STORAGE
DETAIL "A"
DETAIL "B"
Best Copy Available
REVISED & RE-ISSUED FOR CONSTRUCTION
ISSUED FOR CONSTRUCTION
BECHTEL
SAN FRANCISCO
LIMERICK GENERATING STATION UNITS 1 & 2
PHILADELPHIA ELECTRIC COMPANY
GENERAL ARRANGEMENT
PLAN AT EL. 217'-0"
8031
M - 102
10

<u>ADDITIONAL ATTACHMENTS TO</u>
<u>10-10-05 Letter: Supplement to Request for LAR Application of AST</u>

Attachment 011 AST – Drawing M-0107 (2 of 2).



SECTION B-B



SECTION A-A

Best Copy Available

No.	DATE	REVISIONS
9		ADDED ENCLOSURE BLDG'S. PER PCN NO'S 20511 & 20512.
7		GENERAL REVISION FOR EQUIPMENT LOCATION DWGS. M-116, M-117, M-118, M-119, M-120, M-121, M-122, M-123, M-124
6		ADDED WALL, REF. FCR C 3213
5	/75	REVISED DIESEL GEN. BLDG. & ISSUED FOR CLIENT APPROVAL
4		LOCATED CONTROL RM CHILLER
3		REVISED & RE-ISSUED FOR CONSTRUCTION
2		
1		REVISED & RE-ISSUED FOR CONSTRUCTION
0	7/2/71	ISSUED FOR CONSTRUCTION

1/16" = 1'-0" F SCOTT M.GRIG 2/12/71

BECHTEL
SAN FRANCISCO

LIMERICK GENERATING STATION UNITS 1 & 2
PHILADELPHIA ELECTRIC COMPANY

GENERAL ARRANGEMENT
SECTION A-A & B-B

JOB No.	DRAWING No.	REV.
8031	M - 107	9