Attachment 6

Design Basis Fuel Handling Accident
Radiological Analysis with Revised Source Terms
XC-Q1111-98019, Revision 0

DESIGN ENGINEERING CALCULATION GRAND GULF NUCLEAR STATION UNIT ONE	CALC NO.: XC-Q1111-98019 REVISION: 0 PAGE i of iii

TITLE: Design Basis Fuel Handling Accident Radiological Analysis with Revised Source Terms

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REVIEW AND APPROVAL

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REVISION STATUS SHEET

ENGINEERING CALCULATION REVISION SUMMARY

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SHEET REVISION STATUS

SHEET NO.	REVISION	SHEET NO.	REVISION	SHEET NO.	REVISION
i	0	7	0	16	0
ii	0	8	0	17	0
iii	0	9	0	18	0
1	0	10	0	19	0
2	0	11	0	20	0
3	0	12	0	21	0
4	0	13	0	22	0
5	0	14	0	23	0
6	0	15	0	24	0
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APPENDIX/ATTACHMENT REVISION STATUS

APPENDIX NO.	REVISION	ATTACHMENT NO.	REVISION
A	0	1	0

CONTENTS

1.0 INTRODUCTION

This calculation applies the NUREG-1465 [1] revised source terms to the fuel handling accidents that are postulated to occur at Grand Gulf to ensure that the radiological consequences of the worst-case fuel handling accidents are within the appropriate acceptance criteria. This calculation considers the following relaxations to the current plant design and requirements.

- No secondary containment requirements while moving irradiated fuel after a certain decay time
- No secondary containment requirements while performing core alterations
- No automatic control room isolation
- No control room envelope requirements
- No control room fresh air system

2.0 BACKGROUND

The current design basis fuel handling accident is reported in Calculation XC-Q1J11-96005 [2] using the original offsite dose acceptance criteria of 25 rem thyroid and 6 rem whole body based on the Standard Review Plan Section 15.7.4 [3]. In addition, the requirements of 10CFR50, Appendix A, General Design Criterion 19 were applied in the control room. The release fractions and timing, iodine species distribution, and pool decontamination factors in Reg. Guide 1.25 [4] were also applied.

This analysis updates the fuel handling accident for the impact of the NUREG-1465 revised source terms. The changes from the current calculation are listed below.

- An offsite dose limit of 6 rem TEDE (based on 25% of 25 rem TEDE as proposed in Reference 37) is applied instead of the SRP 15.7.4 values of 25% of 10CFR100.
- A control room dose limit of 5 rem TEDE as proposed in Reference 37 is applied instead of the GDC 19 criteria of 5 rem whole body or its equivalent to any part of the body for the duration of the accident.
- Instead of the NUREG-1465 gap release fractions (assuming long-term fuel cooling) of 3%, this calculation applies the draft gap release fractions recently suggested by the Staff.
- Although NUREG-1465 reports that the iodine chemical species distribution will be expected to be primarily aerosol, the recent NRC guidance for this accident suggesting a release that is primarily elemental is applied.
- Draft pool decontamination factors that have recently been suggested by the Staff are applied instead of the Reg. Guide 1.25 pool DF values.
- Updated control room and EAB χ/Q values are applied.
- Secondary containment is neglected for drops of irradiated fuel after a certain decay time and during core alterations.
- No control room isolation or envelope control is assumed.
- The control room fresh air system is not credited.

3.0. GIVEN

A number of regulatory guidance documents are taken as given including:

- the Standard Review Plan,
- the General Design Criteria in 10CFR50, Appendix A,
- the requirements in Regulatory Guide 1.25, and
- the draft alternative source term regulatory guide, DG-1081 (in Reference 42).

The following sections describe how these requirements are addressed in this analysis considering the impacts of the revised source term.

3.1 Standard Review Plan (NUREG-0800)

3.1.1 Section 15.7.4

SRP Section 15.7.4 reports the acceptance criteria related to the consequences of a fuel handling accident as "well within" or 25% of the values of 10CFR 100.11 and further defined as 75 rem thyroid and 6 rem whole body.

Instead of the SRP acceptance criteria, this analysis applies an offsite dose acceptance criterion of 6 rem TEDE for the worst two-hours based on 25% of the 25 rem TEDE dose limit proposed in Reference 37.

3.2 10CFR50, Appendix A

Appendix A of 10CFR50 contains a number of design criteria. The criteria addressing the fuel handling accident are numbers 19, 61, and 63.

3.2.1 General Design Criterion 19

While SRP Section 15.7.4 reports the acceptance criteria related to the offsite radiological consequences of a fuel handling accident, General Design Criterion 19 reports the acceptance criteria for the control room as 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

This analysis applies a TEDE limit of 5 rem in the control room as proposed in Reference 37.

3.2.2 General Design Criterion 61

GDC 61 addresses the design of the systems associated with radioactivity control in the fuel storage and handling systems. Specifically, this criterion addresses "appropriate containment, confinement, and filtering systems".

This analysis will identify these systems as they are credited for radioactivity control. However, this calculation will not credit secondary containment or the Standby Gas Treatment System (SGTS) after a certain decay time, demonstrating that "appropriate containment, confinement, and filtering systems" are still available during a fuel handling accident without the SGTS.

3.2.3 General Design Criterion 63
GDC 63 requires means for detecting excessive radiation levels in fuel storage systems and to initiate the appropriate safety actions.
This analysis will identify these systems as they are credited for radioactivity control. However, this calculation will not credit secondary containment or SGTS after a certain decay time, demonstrating that "appropriate safety actions" do not require secondary containment or the initiation of SGTS during a fuel handling accident.

3.3 Regulatory Guide 1.25
Regulatory Guide 1.25 provides guidance for calculating the activity released to the buildings and the associated offsite doses. This guidance is divided into the following categories.

- Activity Release
- Atmospheric Diffusion Factors
- Dose Calculations

3.3.1 Activity Release
The assumptions in Regulatory Guide 1.25 related to the curies of released activity are addressed below.

Assumption 1.a
The accident occurs at a time after shutdown identified in the Technical Specifications as the earliest time fuel handling operations may begin.
The minimum decay time is 24 hours per Technical Requirements Manual (TRM) 6.9.1 [5] and is used in the analysis. Additional analyses are performed for longer shutdown times.

Assumption 1.b
The maximum fuel rod pressurization is 1200 psig.
This requirement is interpreted to be applicable at the time of the accident (not during operations). For ANF 9x9-5 fuel, the maximum rod internal pressure during operation is predicted to be approximately 1300 psi [6] at the maximum burnup. At the reduced fuel temperatures during shutdown, the rod internal pressure is significantly less than that experienced during operation. Therefore, the maximum fuel rod pressurization for SPC 9x9-5 fuel will remain below 1200 psig.

For GE11 fuel, GE has calculated that the maximum fuel rod pressurization remains below 1200 psig for reactor coolant temperatures up to 200 °F [7] which is the maximum reactor coolant temperature permitted in Modes 4 and 5 per GGNS Technical Specification Table 1.1-1.

Assumption 1.c
The minimum water depth between the top of the damaged fuel rods and the pool surface is 23 feet.
Technical Specification 3.9.6 prohibits the movement of irradiated fuel in the vessel without at least 22'-8" of water above the reactor flange thereby providing significantly more than 23 feet over any damaged rods in the event of a drop over the core. Technical Specification 3.9.7 prohibits the movement of other objects (new fuel assemblies or control blades) in the vessel

without at least 23 feet or water coverage. TRM TR3.7.6 and Technical Specification 3.7.6 require at least 23 feet of water over the top of irradiated bundles in the fuel pool racks at all times.[1]

<u>*Assumption 1.d*</u>
The fraction of the total fission products which are released to the fuel rod's pellet/clad gap shall be taken as:

- *10% of the noble gases (excluding Kr-85)*
- *30% of the Kr-85 inventory*
- *10% of the iodine inventory.*

Although Section 3.6 of NUREG-1465 reports that a 3% gap fraction is applicable to the fuel handling accident, this calculation will apply the release fractions reported by the Staff in the draft technical guidance (included in Reference 42). Specifically, the following release fractions are applied based on rod average exposures up to 62 GWd/MTU.

Table 3-1 Draft NRC Gap Fractions for High Burnup Fuel

Isotope/Group	NRC Draft Gap Fractions
I-131	12%
Kr-85	15%
Other Noble Gases and Iodine	10%
Alkali metals	10%

<u>*Assumption 1.e*</u>
The fission product inventories are calculated assuming 100% full power operation through end of core life with a minimum radial power peaking factor of 1.5 for BWRs.
The fuel handling accident source terms are generated in Reference 8. This analysis uses the ORIGEN2.1 methodology to generate the bundle source term inventory conservatively assuming full power operation through the end of bundle life. The radial peaking factor assumed in this analysis is 1.70. Core power and power distribution uncertainties are also considered in the calculation of these source terms.

<u>*Assumption 1.f*</u>
The iodine gap inventory is composed of 99.75% inorganic species and 0.25% organic species.
Although NUREG-1465 reports that 95% of the iodine is release in the particulate form of CsI, this calculation assumes that 99.75% of the gap inventory is in the elemental form while the remainder is in the organic form. The NRC's draft guidance also supports this species distribution. The cesium and rubidium are assumed to be in the particulate form and the noble gases are assumed in the elemental form consistent with Section 3.5 of NUREG-1465.

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1 Even a bundle laying on top of the fuel bale handles of seated bundles in the racks would maintain 23 feet of water coverage based on the nominal water level at 207' 10". The fuel bale handle rises approximately 5" above the top of the racks. Considering the 176 1/8" height of the racks, a 6" wide bundle, and the fuel pool floor at El. 167'6", there would be 24.74 feet of water above the top of a horizontal bundle laying on the bale handles of bundles seated in the racks.

<u>Assumption 1.g</u>
The pool decontamination factors for the inorganic and organic species are 133 and 1 respectively, giving an overall effective decontamination factor of 100.
The draft regulatory guidance in Reference 42 permits an elemental iodine DF of 500 for 23 feet of water. Considering the iodine species distribution, an overall DF of 222.5 is applied instead of the 200 value reported in the draft regulatory guidance. Although an increased pool DF is expected for water depths of 46 feet or greater, such as the drop over the core, this calculation will conservatively apply the value for 23 feet of water.

$$\text{Overall Iodine DF (23 feet)} = \frac{1}{\dfrac{0.9975}{500} + \dfrac{0.0025}{1}} = 222.5$$

For the particulate radionuclides, an infinite DF is applied consistent with the draft regulatory guidance. Appendix B confirms that essentially all the particulates are scrubbed by a 23-foot sub-cooled water pool.

<u>Assumption 1.h</u>
The retention of noble gases in the pool is negligible (i.e., decontamination factor of 1).
This analysis credits this assumption.

<u>Assumption 1.i</u>
The activity that escapes from the pool to the building is released to the environment over a 2-hour period.
This 2-hour assumption is consistent with the source terms being vented from the release area rather than diffusing through cracks or seals. For example, a drop in containment with 50% mixing in the containment volume (for a net volume of 7E5 ft^3), would be completely exhausted by the containment ventilation system flow of 6000 cfm in approximately 117 minutes. This analysis credits this 2-hour assumption releasing all airborne activity over a two-hour period. As described in Appendix A, the source terms are assumed to be released at a constant rate over this 2-hour period for the purposes of determining the control room dose. The EAB and control room 2-hour χ/Q values are also applied. No decay of the fission products is conservatively assumed during residence in the containment or auxiliary building or during transport to the environment. The worst-case EAB dose would consequently occur over the first two hours.

<u>Assumption 1.j</u>
For scenarios in which the activity is exhausted through charcoal adsorbers, plant-specific filter efficiencies can be credited.
This analysis credits no filtration through the Standby Gas Treatment or Control Room Fresh Air filters or charcoal beds after a certain decay period. The airborne release from the fuel pool is assumed released to the environment without any mixing in the containment or secondary containment atmosphere.

For accidents that occur before this decay period, secondary containment and SGTS is credited while the control room fresh air supply system is neglected. Technical Specification 5.5.7 [5] reports the acceptance criteria for the Ventilation Filter Testing Program. For the SGTS charcoal adsorbers, Technical Specification 5.5.7c requires testing in accordance with Regulatory Guide 1.52, Revision 2 [43] with an acceptance criteria of a methyl iodide penetration of less than 0.175% at a relative humidity of 70%. For 8-inch charcoal beds such as in SGTS [44], the R.G. 1.52 assigned filtration efficiency is 99% for both elemental and organic iodine. A SGTS flowrate of 4000 cfm is assumed based on TSPS 376 and 378 in Supplement 6 to NUREG-0831 [45]. The SGTS is not credited for the removal of any noble gases.

The non-safety charcoal beds in the containment ventilation system are not credited in this calculation. In the control room, the control room fresh air supply (CRFAS) system is not credited in this calculation.

Assumption 1.k
The effluent from the filter system passes directly to the emergency exhaust system without mixing in the surrounding building atmosphere and is then released as an elevated plume for those facilities with stacks. Credit for mixing is evaluated by the NRC on an individual case basis.
As reported in Section 15.3.3 of NUREG-0831 [45], the NRC has permitted credit for 50% mixing for fuel handling accidents in the GGNS primary containment. Consistent with the NRC assumptions in NUREG-0831 Table 15-4, a containment free volume of 1.4E6 ft^3 is credited in this analysis. Since GGNS does not have a stack, a ground release is assumed in this analysis.

Assumption Conditions
Footnote 1 of R.G. 1.25 indicates that the above assumptions are only applicable if the following three conditions are met. These conditions are primarily related to the fraction of the fission products which are released to the pellet/clad gap. If any condition is not met, the impact on the above assumptions would need to be assessed. As shown below, all of these conditions are met for GGNS.

a. *The peak linear heat generation rate is not to exceed 20.5 kW/ft.*
Siemens' mechanical design report for 9x9-5 fuel reports the 9x9-5 the design maximum LHGR for 9x9-5 assemblies is less than 14 kW/ft [22]. For the GE11 fuel type, Figure 2.2.1 of Reference 24 indicates that the fuel rod maximum power remains less than 15 kW/ft. Cycle-specific LHGR multipliers ensure that the LHGR remains at or below these maximum values.

b. *The maximum fuel centerline temperature is to be less than 4500 ºF.*
The maximum temperature for SPC 9x9-5 and GE11 fuel satisfy this requirement [10,11].

c. *The average burnup of the peak bundle is less than 25,000 MWd/MTU.*
As addressed in Assumption 1.d, the gap fractions applied in this calculation are applicable to fuel rod exposures up to 62 GWd/MTU per the recent NRC guidance [42]. The GGNS fuel rods have been confirmed to not exceed a burnup of 60 GWd/MTU [39].

3.3.2 Atmospheric Diffusion Factors

Consistent with the guidance in R.G. 1.25, Footnote 6, a ground release is postulated since GGNS does not have a stack that is (1) more than two and one-half times the height of any structure close enough to affect the dispersion of the plume, and (2) located far enough from any structure which could affect the dispersion of the plume.

The offsite atmospheric diffusion factors used in this analysis are calculated in XC-Q1C84-92009 [13] using the PAVAN code. The equations used to generate these parameters are consistent with those presented in R.G. 1.25 for ground releases.

The control room χ/Q used in this analysis is calculated in XC-Q1111-98011 [14] using the ARCON96 code. Since procedures will be in place to quickly close any open doors or equipment hatches in the secondary containment boundary, the released source terms are assumed to be released from secondary containment via either the containment ventilation system, Standby Gas Treatment system (with no credit for filters), or Fuel Handling Area Ventilation system. Since the containment ventilation louvers on the roof of the Aux. Building are the closest of these potential release points to the control room intakes (per A-1107 [38]) and result in the highest χ/Q values, the source terms are assumed to be released via the containment ventilation exhaust louvers over a 2-hour period. No credit is taken for the non-safety charcoal beds in this release path.

The χ/Q values applied in this analysis are summarized below.

Table 3-1 GGNS 2-Hour Accident χ/Q Values

Location	χ/Q (m^3/s)
EAB (696 m)	9.56E-4
LPZ (3219 m)	1.94E-4
Control Room	2.75E-3

Since the EAB dose must meet the same acceptance criteria as the LPZ dose and the EAB has a higher χ/Q value, the offsite dose impact will be evaluated only at the EAB.

3.3.3 Dose Calculations

This calculation applies the R.G. 1.25 approach to calculating the thyroid dose with updated dose conversion factors. Instead of the R.G. 1.25 gamma and beta approach to calculating the whole body dose, this analysis applies a dose conversion factor approach similar to that applied for the thyroid dose calculation. This approach is identical to that applied by the NRC's RADTRAD code described in NUREG/CR-6604 [15].

3.4 Source Terms

The transient bundle source terms have been calculated for the fuel handling accident in Reference 8. Isotopes with bundle activities less than 1 Curie are neglected. Since the particulate radionuclides (*i.e.*, cesium and rubidium) are completely retained by the water pool (as discussed in Section 3.3.1), these nuclides are not modeled in this calculation. The activities applied in this calculation are listed below as calculated in Reference 8. The inventories at times after 3 days are calculated by applying the decay constants reported below to the activity after 3 days of decay.[2]

Table 3-2 Bundle Activities (Curies) of NUREG –1465 Bundle Gap Isotopes

	Decay Time				
Isotope	Decay Constant (day^{-1})	24 Hours	48 Hours	72 Hours	7 Days
BR 82	4.713E-01	1.698E+03	1.060E+03	6.618E+02	1.005E+02
BR 83	6.960E+00	3.876E+01	3.677E-02	3.488E-05	2.825E-17
I130	1.346E+00	5.233E+03	1.362E+03	3.546E+02	1.628E+00
I131	8.621E-02	2.227E+05	2.057E+05	1.894E+05	1.342E+05
I132	7.296E+00	2.789E+05	2.254E+05	1.822E+05	3.851E-08
I133	7.998E-01	2.204E+05	9.905E+04	4.452E+04	1.816E+03
I135	1.901E+01	3.599E+04	2.906E+03	2.347E+02	2.204E-31
KR 83M	8.944E+00	1.499E+02	1.551E-01	1.486E-04	4.315E-20
KR 85	1.770E-04	3.102E+03	3.101E+03	3.101E+03	3.099E+03
KR 85M	3.713E+00	2.185E+03	5.334E+01	1.302E+00	4.613E-07
KR 88	5.858E+00	7.203E+02	2.054E+00	5.856E-03	3.908E-13
XE129M	7.806E-02	8.772E+00	8.044E+00	7.376E+00	5.398E+00
XE131M	5.815E-02	2.703E+03	2.684E+03	2.657E+03	2.106E+03
XE133	1.320E-01	4.360E+05	4.014E+05	3.610E+05	2.129E+05
XE133M	3.165E-01	1.334E+04	1.083E+04	8.394E+03	2.367E+03
XE135	1.828E+00	1.197E+05	2.683E+04	4.919E+03	3.282E+00
XE135M	2.718E+00	5.765E+03	4.655E+02	3.760E+01	7.131E-04

3.5 Dose Conversion Factors

The effective dose conversion factors for the TEDE calculations are identical to those applied in RADTRAD [15] and are based on FGR 11 [16] and 12 [17]. In some cases, these values include the DCFs of the isotope's decay products as noted in NUREG/CR-6604 Table 1.4.3.3-2. These dose conversion factors are reported below.

2 Since the inventories after 3 days of decay are calculated based solely on the decay constant, they do not include the impact of decay chains. This application is appropriate since the 3-day ORIGEN activities capture any relevant short-lived isotopes such as Te-131 and Te-131m.

Table 3-3 Dose Conversion Factors

Isotope	Whole Body (Rem-m³/Ci-s)	Inhalation (Rem/Ci)
BR 82	4.810000E-01	1.528100E+03
BR 83	1.413400E-03	8.917000E+01
I130	3.848000E-01	2.641800E+03
I131	6.734000E-02	3.289300E+04
I132	4.144000E-01	3.811000E+02
I133	1.087800E-01	5.846000E+03
I135	3.068780E-01[3]	1.228400E+03
KR 83M	5.550000E-06	0.000000E+00
KR 85	4.403000E-04	0.000000E+00
KR 85M	2.767600E-02	0.000000E+00
KR 88	3.774000E-01	0.000000E+00
XE129M	3.922000E-03	0.000000E+00
XE131M	1.439300E-03	0.000000E+00
XE133	5.772000E-03	0.000000E+00
XE133M	5.069000E-03	0.000000E+00
XE135	4.403000E-02	0.000000E+00
XE135M	7.548000E-02	0.000000E+00

3 Includes Xe-135m

4.0 ASSUMPTIONS

4.1 Generic GE Fuel Handling Assumptions

This analysis incorporates the assumptions in the generic GE fuel handling accident in GESTAR-II [18]. This accident scenario assumes the dropped assembly (a channeled fuel bundle) impacts four seated bundles at a small angle to vertical. Then, the assembly tips over and impacts horizontally on top of other bundles. These assumptions (and their appropriateness to the Siemens bundles currently in the GGNS reactor) are addressed below.

i. The grapple cable is assumed to break allowing the grapple head and three sections of the telescoping NF-500 mast to remain attached to the falling assembly. This assumption is consistent with NS&RA recommendations [19]. TRM 6.9.3 requires that fuel bundles be moved in containment only with the refueling platform. TRM 6.9.5 requires that the fuel handling platform be used to handle irradiated bundles in the auxiliary building.

ii. Dissipation of the mechanical energy of the dropped objects by the water (*i.e.*, drag forces) is neglected.

iii. All rods in the dropped assembly are assumed to fail in bending. No credit is taken for the lateral support provided by the channel.

iv. One half of the energy is assumed to be absorbed by the dropped assembly and one-half by the impacted assemblies. This assumption was addressed by GE for GGNS in Reference 20 and is not applied to "light loads".

v. The wet weight of a fuel assembly (bundle plus channel) is assumed to be 600 pounds. This weight bounds the Siemens 9x9-5 assembly (268 kgs or 590 pounds per Reference 21) and the GE11 assembly (562 pounds per GESTAR-II [18]).

vi. The height of the fuel assembly is assumed to be 180 inches. This height bounds the 176-inch height of the Siemens bundles [22] and the 160-inches applied for the GE bundles reported in GESTAR-II.

vii. The wet weight of the GGNS NF-500 mast is assumed to be 619 pounds [23]. This weight is consistent the value reported in GESTAR-II and bounds drops with the lighter NF-400 mast in the auxiliary building.

viii. No energy is assumed absorbed by the fuel pellets. Only the non-fuel portions of the bundle are assumed to absorb the impact energy. A bounding value of the cladding fraction of the non-fuel bundle mass is assumed to be 52% for channeled bundles. These values conservatively overestimate the cladding fraction for Siemens bundles per Reference 21 and bounds the GE NRC-approved value of 51% in GESTAR-II. All struck bundles (both in the core and the pools) are assumed to be channeled since GGNS does not re-use channels.

4.2 Treatment of Part-Length Rods

The GE11 fuel bundle is composed of 74 fuel rods [24], eight of which are part-length rods which are attached at the bottom of the bundle and are approximately 60% of the length of the full-length rods. In the event a seated GE11 bundle is struck, these part-length rods would not see any significant impact loads. On this basis, only the 66 full-length rods in the struck GE11 bundles will be considered for failure in this analysis.

The bundle source terms are assumed to be evenly distributed among the fuel rods. Considering the 66 full-length rods and the 8 part-length rods at 60% of the length of the full-length rods, there are 70.8 effective full-length rods in each GE11 bundle. Since the source terms have been calculated on a bundle basis [8], the source terms in each rod are assumed to be 1/70th of the bundle source terms.

4.3 Previously Discharge Bundles

Since the GE 8x8 and Siemens 8x8 fuel designs have not been in the core for at least one cycle, these fuel types have a significantly reduced source term inventory. For example, the I-131 inventory would be negligible considering the 8-day half-life of this important isotope. On this basis, these fuel designs will not be considered in this analysis as they are bounded by the designs in the reactor.

4.4 Treatment of Bromine

As a halogen, bromine isotopes are modeled identical to iodine in terms of pool and charcoal decontamination factors.

5.0 CALCULATIONS

5.1 Failure Thresholds

In a fuel handling accident, rod failures will occur due to both bending and compression loads. The rods in the dropped bundle experience a bending load due to the assumed slight angle from vertical at which they strike the seated bundles while the impacted rods experience a compression load. Since the bending threshold is small (~1 ft-lb) for fuel rods, all rods in the dropped bundle are assumed to fail in bending.

As discussed by Siemens in Reference 25, a transient-induced strain as low as 1% can result in fuel rod failure in compression. This 1% limit is also discussed for GE fuel in GESTAR-II. For the GE fuel bundles, GESTAR-II reports a bounding compression failure threshold of 200 ft-lbs for GE fuel designs based on a uniform 1% plastic deformation. This value represents a 20% reduction from the original 250 ft-lb threshold reported in Reference 26. Considering the stronger cold-worked cladding of the Siemens bundles relative to the fully-annealed cladding of the GE bundles, the 200 ft-lb compression failure threshold would also be a conservatively low value for the SPC 9x9-5 bundles.

Considering that the GE11 bundles have more source terms per rod than the SPC 9x9-5 design (due to the fewer effective full-length rods, 70 versus 76) and that the rod failure threshold is assumed identical to the SPC fuel, the GE fuel design would be the limiting fuel design in a fuel handling accident. As such, this analysis will conservatively assume that a GE11 bundle is dropped onto seated GE11 bundles.

5.2 Failed Rods Calculation Methodology

This analysis applies GE's GESTAR-II methods for the calculation of the number of rod failures resulting from a fuel handling accident. As reported in Section 4, the dropped assembly impacts four seated bundles at a small angle to vertical. Then, the dropped assembly tips over and impacts horizontally on top of other bundles. The number of rods that fail in the struck bundles can be calculated from the equations derived in GESTAR-II, Section S.2.2.3.5 as follows.

$$\text{Number of Impacted Rod Failures} = \frac{0.5 * M_c\left[(W_a + W_m) * H_d + W_m H_a + \frac{1}{2} W_a H_a\right]}{FT} \tag{5-1}$$

where:

- M_c = Cladding fraction of non-fuel bundle mass (0.52 from Section 4)
- W_a = Wet weight of the dropped assembly (600 lbs from Section 4)
- W_m = Wet weight of the dropped sections of the mast (619 lbs from Section 4)
- H_d = Drop height [ft] (specific to scenario)
- H_a = Height of the dropped assembly (180 inches from Section 4)
- FT = Rod failure threshold (200 ft-lbs from Section 5.1)

With the values reported above, Equation 5-1 can be simplified as:

$$\text{Number of Impacted Rod Failures} = 1.585 * H_d + 17.921 \tag{5-2}$$

5.3 Drop Scenarios

A number of different fuel-related accidents can be postulated to occur at GGNS considering the relaxations proposed in Section 1. This section develops the worst-case accidents and calculates the number of damaged fuel rods for each case. These cases include:

1. drop of an irradiated assembly over the core,
2. drop of an un-irradiated assembly over the core,
3. drop of an irradiated assembly over the racks, and
4. drop of an un-irradiated assembly over the racks.

These cases are discussed in detail below.

5.3.1 Drop Scenario 1: Drop of an irradiated fuel assembly over the core

In this scenario, an irradiated fuel assembly is being moved over the core by the refueling platform and the mast is at its fully retracted position. The grapple cable breaks dropping the irradiated assembly, the grapple head, and all three sections of the mast. This assembly strikes the upper tie plates of the irradiated assemblies in the core failing all rods of the dropped assembly in bending and a number of rods in the struck assemblies.

Figure 5-1 illustrates the heights involved in this scenario. UFSAR Section 9.1.4.2.7.1 indicates that the refueling platform main grapple, at its fully retracted position, maintains at least 7' 9" (93") of water shielding over the active fuel. This shielding is ensured by Procedure 06-OP-1C71-V-0002 [27][4]. Reference 28 indicates that the normal water level is 23 feet (207' 10"-184' 10" [41]) above the flange[5]. Reference 29 indicates that the elevations of the top of the active fuel in the vessel (based on 150-inch active fuel lengths) and vessel flange are 366.31" and 745" AVZ respectively. Reference 30 indicates that the top of the fuel assembly upper tie plate (where the dropped item would impact) is approximately 12 inches above the active fuel for the Siemens 9x9 designs. A similar distance is expected for GE11 designs with 150-inch active fuel length.

Assuming a short pellet stack of 146 inches and neglecting the bundle nosepiece, the bottom of the bundle would be 239 inches (7'9"+146") below the surface of the pool. The pool surface is calculated to be 654.69 inches (745"-366.31"+23') above the TAF or 642.69 inches (654.69"-12") above the upper tie plates of the seated fuel bundles. The drop height is therefore calculated as 403.69 inches (642.69"-239") or less than the 34 feet assumed by GE in Section S.2.2.3.5 of GESTAR-II. Using a bounding value of 34 feet in Equation 5-2, 72 rods fail in the impacted bundles.

$$\text{Number of Impacted Rod Failures} = 1.585 * 34 + 17.921 = 72$$

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4 Note that Reference 27 requires that an additional 17.33 inches be added to the distance measured from the surface to the bottom of the bail handle. This requirement ensures a total of 7 feet 9 inches of water shielding between the active fuel and the pool surface.

5 Note that this water height is conservatively greater than that in Technical Specification 3.9.6 which requires that level be at least 22' 8" above the vessel flange.

Considering the 70 rod failures in the dropped assembly, the total number of rod failures is summed to be 142, representing 2.029 failed bundles. This result is consistent with the 140 (123*172/151) 9x9 rods predicted to fail with the NF500 mast reported in Section 3.8 of Reference 36.

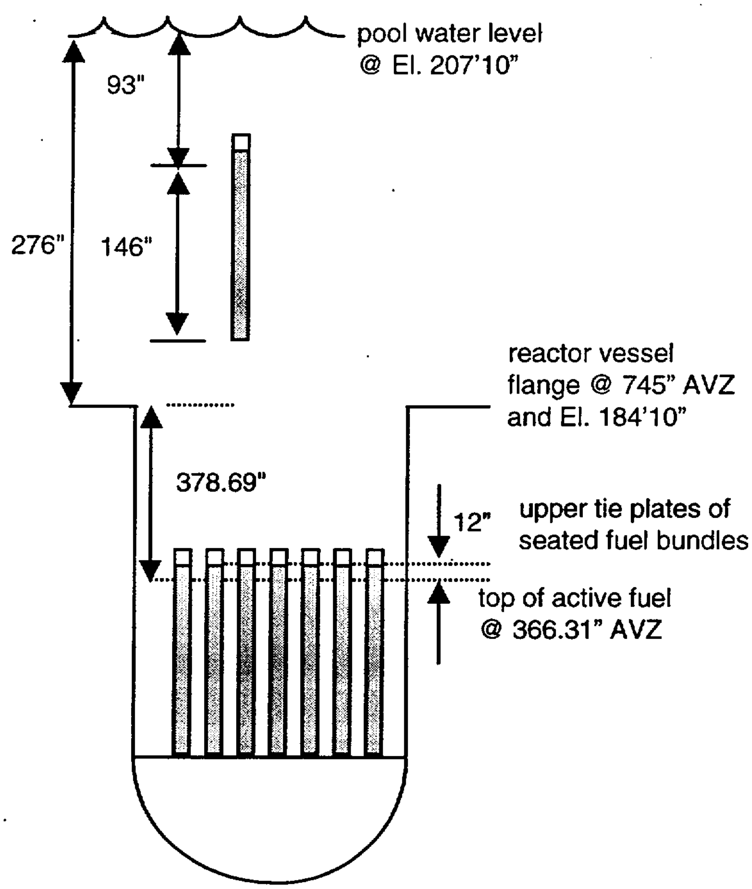

Figure 5-1 Core Drop Heights

5.3.2 Drop Scenario 2: Drop of an un-irradiated fuel assembly over the core

As calculated above, the drop of an un-irradiated bundle would result in 72 struck rod failures. Since an un-irradiated fuel assembly contains no source terms, the number of rod failures for dose purposes is only those struck rods or 72 rods.

5.3.3 Drop Scenario 3: Drop of an irradiated fuel assembly over the racks

In this scenario, an irradiated fuel assembly is being moved over the upper containment racks by the refueling platform and the mast is at its fully retracted position. The grapple cable breaks dropping the irradiated assembly, the grapple head, and all three sections of the mast. This assembly strikes irradiated assemblies seated in the racks failing all rods of the dropped assembly in bending and a number of rods in the struck assemblies.

Figure 5-2 illustrates the heights involved in this scenario. The elevation of the fuel pool bottom is 167' 6" (Ref. 34, conservatively neglecting the 1/4" liner plate) while the racks are 176.125" tall [35]. Assuming a short pellet stack of 146 inches and neglecting the bundle nosepiece, the maximum drop distance to the top of the racks can be calculated as 207'10"-167'6"-93"-146"-176.125" = 68.875 inches or 5.8 feet. This distance is rounded up to 6 feet. Using Eq. 5-2, 28 rods fail in the impacted bundles.

$$\text{Number of Impacted Rod Failures} = 1.585 * 6 + 17.921 = 28$$

Considering the 70 rod failures in the dropped assembly, the total number of rod failures is summed to be 98, representing 1.4 failed bundles.

This accident bounds the failures that would be calculated by a fuel handling accident over the spent fuel pool due to the heavier NF-500 mast on the refueling platform. Reference 31 indicates that the total wet weight of the movable sections of the fuel handling platform mast is 237 pounds (7- and 10-inch sections plus the grapple head) compared to the 619 pounds used in this analysis. The drop height for this scenario is identical to that in containment since the racks are identical, the pool bottom is at the same elevation [32], and the same amount of water shielding is available[6].

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6 UFSAR Section 9.1.4.2.7.3 indicates that the fuel handling platform main grapple, at its fully retracted position, maintains at least 7' 9" of water shielding over the active fuel. For the spent fuel pool, this shielding is ensured by the surveillances in Procedure 06-OP-1F11-V-001 [33] which requires at least 6' 3.67" between the bottom of the bail handle (on the dummy bundle) and the surface of the fuel pool. An additional 17.33 inches exists between the bottom of the handle and the active fuel as included in Reference 27 for the refueling platform.

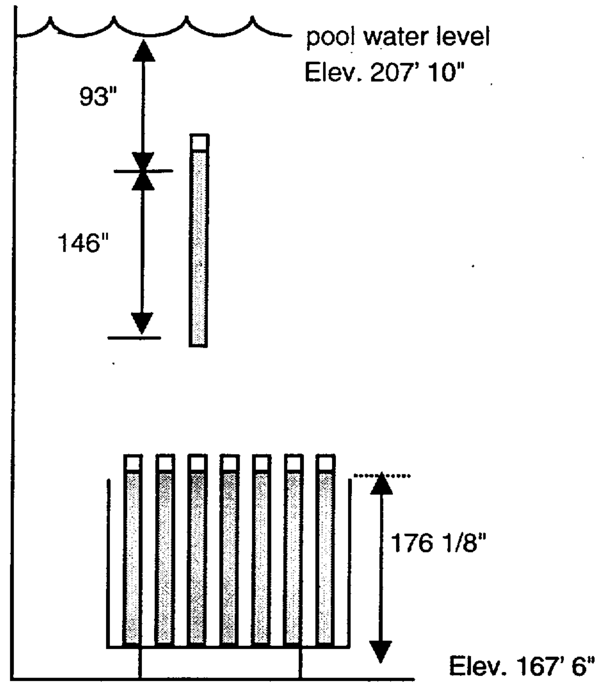

Figure 5-2 Spent Fuel Pool Drop Heights

5.3.4 Drop Scenario 4: Drop of an un-irradiated fuel assembly over the racks

As calculated above, the drop of an un-irradiated bundle would result in 28 struck rod failures. Since an un-irradiated fuel assembly contains no source terms, the number of rod failures for dose purposes is only those struck rods or 28 rods.

5.3.5 Drop Scenario Summary

The following table summarizes the number of rods calculated to fail in each of the analyzed scenarios.

Table 5-1 Irradiated Fuel Failures by Scenario

Number	Scenario	Number Failed Irradiated Rods
1	Drop of an irradiated fuel assembly over the core	142
2	Drop of an un-irradiated fuel assembly over the core	72
3	Drop of an irradiated fuel assembly over the racks	98
4	Drop of an un-irradiated fuel assembly over the racks	28

5.4 Decontamination Factor Scenarios

Several scenarios of fuel handling accidents can be postulated to occur within the GGNS containment or auxiliary building. This analysis addresses dropping an irradiated fuel bundle within the containment or auxiliary building with secondary containment as required by the Technical Specifications. However, since the Technical Specifications do not require secondary containment when un-irradiated fuel assemblies are moved over the spent fuel pools, this analysis also considers the drop of un-irradiated bundles without secondary containment. An additional scenario in which irradiated fuel is dropped without secondary containment after a certain decay period is also included in support of ongoing licensing efforts. The analyzed scenarios are summarized in Table 5-2.

5.4.1 DF Scenario 1: Drop in the Auxiliary Building

In this event, an irradiated fuel assembly is dropped onto irradiated assemblies in the spent fuel pool. Since irradiated fuel is being moved, secondary containment is active as required by Technical Specifications. As discussed in the UFSAR Section 9.4.2, the fuel handling area ventilation system in the auxiliary building is designed to close prior to the release of any significant amount of unfiltered air into the environment. This performance is ensured through requirements on the response times of the Fuel Handling Area Ventilation Exhaust and Pool Sweep Exhaust radiation monitors in TRM Table TR3.3.6.2-1 and isolation times time of the Fuel Handling Area ventilation exhaust dampers in TRM Table TR3.6.4.2-1. On this basis, no unfiltered activity is assumed to be released to the environment through this system.

As discussed in Section 4.3.1, the SGTS filter trains remove 99% of the iodine passing through them. A bypass leakage path is assumed to release 1 cfm of unfiltered leakage directly to the environment. The effect of this bypass leakage is to reduce the effective decontamination factor from 100 to 97.6 as calculated below.

Effective Unfiltered Release: $(4000\,\text{cfm})\cdot(0.01)+(1\,\text{cfm})\cdot(1.0)=41\,\text{cfm}$

Decontamination Factor: $\dfrac{4000\,\text{cfm}+1\,\text{cfm}}{41\,\text{cfm}}=97.6$

5.4.2 DF Scenario 2: Drop in Containment With Release Completely Pulled Through the Open Eq. Hatch

In this scenario, an irradiated fuel assembly is dropped in the containment. Since irradiated fuel is being moved, secondary containment is active as required by Technical Specifications. For this case, it is assumed that all the iodine is transferred into the spent fuel pool area via the open equipment hatch due to the lower pressure maintained in that area by the fuel handling area ventilation system. As in the previous case, the Fuel Handling Area ventilation system will isolate prior to the release of any activity into the environment and the iodine decontamination factor is thus 97.6. This case, therefore, will result in the same offsite doses as the previous case.

5.4.3 DF Scenario 3: Drop in Containment With Release Partially Pulled Through the Open Eq. Hatch

This event is similar to the above case except a limited amount of activity is released through the containment ventilation system prior to its isolation. Even though the containment ventilation system has charcoal filters, no credit is taken for iodine removal since they are not safety-related. After the containment ventilation system is isolated, the remaining iodine will be pulled through the open equipment hatch into the fuel handling area from which it will be exhausted through the SGTS filters.

Since TRM Table TR3.3.6.1-1 does not report a required response time for the instrumentation response time of the Containment and Drywell Ventilation Exhaust radiation monitor, a bounding value of 10 seconds will be applied for this calculation. TRM Table TR3.6.1.3-1 requires that the maximum isolation time of the Containment and Drywell Area Ventilation supply and exhaust dampers to be 4 seconds. However, for the purposes of this analysis, this isolation time is increased to 110 seconds.

Therefore, the maximum total response time for complete isolation of the ventilation exhaust is 120 seconds. If the travel time from the detector to the isolation valve is conservatively neglected, a release directly from the containment atmosphere to the environment would exist for 120 seconds. Reference 12 indicates that this flowrate is 6000 cfm. Section 15.3.3 of NUREG-0831 [45] also credits a 6000 cfm flowrate for the containment ventilation system. With a containment volume of 1.4E6 ft^3 and 50% mixing in the containment, the release fraction can be determined as follows:

Containment volume containing fission products	$(1.4 \cdot 10^6\ ft^3)(50\%) = 7.0 \cdot 10^5\ ft^3$
Volume released in 120 seconds	$(120\ sec)\left(\frac{1\,min}{60\ sec}\right)\left(6000\,\frac{ft^3}{min}\right) = 12{,}000\ ft^3$
Fraction of fission products released in 120 seconds	$\frac{12{,}000\ ft^3}{7.0 \cdot 10^5\ ft^3} = 0.0171 = 1.71\%$

After the containment ventilation system is secured, the remaining iodine will be released via the SGTS system filters with a decontamination factor of 100. The bypass leakage path decreases the effectiveness of secondary containment and effectively releases 1.025% (41 cfm / 4001 cfm) of the SGTS flow as unfiltered leakage. The effective overall iodine decontamination factor for this event is calculated below to be 36.8.

Fraction of iodine released to environment	$\underbrace{(0.0171)(100\%)}_{\text{Unfiltered}} + \underbrace{(1-0.0171)(1.025\%)}_{\text{Filtered}} = 0.0272$
Overall Effective Decontamination Factor	$\frac{1.0}{0.0272} = 36.8$

5.4.4 DF Scenario 5: Drop of Fuel Without Secondary Containment

In support of an ongoing licensing effort, additional cases are considered in which SGTS is not available during movement of both irradiated and un-irradiated fuel. In this case, all of the iodine is released to the environment with no filtering. The effective decontamination factor for this case is therefore 1.0.

Table 5-2 Summary of Airborne Iodine Decon Factors

Case	Location	Dropped Object	Release Path	Sec. Cont. Status	Effective DF
1	Auxiliary Building	irradiated fuel (Figure 6-3)	totally through SGTS	active	97.6
2	Containment	irradiated fuel (Figure 6-4)	into Aux. Bldg. through equipment hatch, totally through SGTS	active	97.6
3	Containment	irradiated fuel (Figure 6-4)	through cont. vent. system until isolated; remainder through equipment hatch and through SGTS	active	36.8
4	Auxiliary Building or Containment	irradiated fuel	direct to environment - no filtering	inactive	1.0

5.4.5 Summary of Decontamination Factors

The following tables summarizes the pool, building, and total iodine decontamination factors developed for this analysis.

Table 5-3 Summary of Total Iodine Decontamination Factors

	Accident Scenario	Pool DF	Building DF	Total DF
1.	Drop in Auxiliary Building or in Containment with Release **Completely** Pulled Through Open Equipment Hatch	222.5	97.6	21716
2.	Drop in Containment with Release **Partially** Pulled Through Open Equipment Hatch	222.5	36.8	8188
3.	Drop in Auxiliary Building or Containment without Secondary Containment	222.5	1	222.5

5.5 Spectrum of Accidents

A variety of different scenarios can be developed from the drop scenarios developed in Section 5.3 and the decontamination factors developed in Section 5.4. The limiting scenarios evaluated in this analysis are presented in Table 5-4.

Considering the larger number of fuel failures for drops over the core relative to drops of the racks and the assumed pool decontamination factor, a drop over the core will result in more airborne activity than a drop over the racks. Considering the bounding case in which this activity is partially pulled through the equipment hatch (*i.e.,* a limited amount is released unfiltered to the environment during isolation of the containment ventilation system), the worst case fuel handling accident can be developed based on the minimum allowable decay time of 24 hours. This case is labeled Case A in Table 5-4.

GGNS Technical Specification 3.6.4.3 does not require secondary containment during the movement of un-irradiated bundles when core alterations are not being performed. Consequently, the drop of an un-irradiated bundle over the racks can be postulated without secondary containment under the current Technical Specifications. The impacted bundles are assumed to have the minimum decay of 24 hours. This case is labeled Case B in Table 5-4.

GGNS proposes to allow handling of irradiated fuel after a certain decay time without secondary containment. This decay time is calculated in the following sections to be 7 days. This case is labeled Case C in Table 5-4. GGNS proposes to allow core alterations without secondary containment. Since core alterations with an irradiated bundle would fall under Case C above (and require secondary containment until after 7 days of decay), this case would involve the drop of an un-irradiated bundle. The impacted bundles in the core are assumed to have the minimum decay of 24 hours. This case is labeled Case D in Table 5-4.

Table 5-4 Limiting Cases

CASE	DESCRIPTION	Decay Time (days)	Total Iodine DF	Failed Irradiated Rods
A	Drop of Irradiated Assembly Over Core with Release Partially Pulled through Equipment Hatch (Limiting Case Under Current Technical Specifications)	1	8188	142
B	Drop of Un-irradiated Assembly Over Racks without Secondary Containment (Limiting Case Under Current Technical Specifications)	1	222.5	28
C	Drop of Irradiated Assembly Over Core without Secondary Containment (Proposed Relaxation to Current Technical Specifications to allow movement of irradiated fuel without secondary containment after 7 days of decay)	7	222.5	142
D	Drop of Un-irradiated Assembly Over Core without Secondary Containment (Proposed Relaxation to Technical Specifications to allow core alterations without secondary containment)	1	222.5	72

5.6 Dose Calculations

This section describes the methodology used to determine the offsite and control room doses.

5.6.1 EAB Dose Formulations

The following formulas are applied to calculate the EAB doses from each isotope.

Whole Body Dose:

$$D_j^{wb} = \sum_i \frac{DCF_i^{wb} * Q_i * f_i}{DF_{i,j}} * \frac{\chi}{Q} * FB_i$$

where:

D_j^{wb} = Whole Body Dose for Scenario j [Rem]
DCF_i^{wb} = Whole Body Dose Conversion Factor for Isotope i (Rem-m^3/Ci-s from Table 3-3)
Q_i = Bundle Inventory for Isotope i (Ci from Table 3-2)
f_i = Release Fraction for Isotope i (from Section 3.3.1)
$DF_{i,j}$ = Decontamination Factor for Isotope i in Scenario j (from Table 5-4)
χ/Q = EAB Atmospheric Dispersion Factor (9.56E-4 s/m^3 per Table 3-1)
FB_j = Number of Failed Bundles for Scenario j (from Table 5-4)

Inhalation Dose:

$$D_j^{inh} = \sum_i \frac{DCF_i^{inh} * Q_i * f_i}{DF_{i,j}} * BR * \frac{\chi}{Q} * FB_j$$

where:

D_j^{inh} = Inhalation Dose for Scenario j [Rem]
DCF_i^{inh} = Inhalation Dose Conversion Factor for Isotope i (Rem/Ci from Table 3-3)
BR = Breathing Rate [3.47E-4 m^3/s per Ref. 9]

The TEDE dose is the sum of the whole body and inhalation doses.

$$D_j^{TEDE} = \frac{\chi}{Q} * FB_j * \sum_i \frac{Q_i * f_i}{DF_{i,j}} * \left(DCF_i^{wb} + DCF_i^{inh} * BR\right)$$

5.6.2 Control Room Dose Formulations

The methodology for the control room dose calculation is developed in Appendix A and concludes that, for the case of no filtered recirculation or post-accident purge, the following formula can be applied to calculate the control room TEDE dose.

$$D_j^{TEDE} = \frac{\chi}{Q} * \sum_i \left(\frac{DCF_i^{wb}}{MCGF} + DCF_i^{inhal} * BR\right) * Q_{i,j} = \frac{\chi}{Q} * FB_j \sum_i \left(\frac{DCF_i^{wb}}{MCGF} + DCF_i^{inhal} * BR\right) * \frac{Q_i * f_i}{DF_{i,j}}$$

where:

$Q_{i,j}$ = Activity of Isotope i in Containment Atmosphere for Scenario j, MCGF = Murphy-Campe Geometry Factor = $1173/V_{CR}^{0.338}$ = 17.5, and

χ/Q = Atmospheric Dispersion Factor from Release Point to Control Room Intake (2.75E-3 s/m^3 from Table 3-1).

The control room free volume is 2.53E5 ft^3 based on Reference 49. Although this volume considers the entire airtight boundary including the control cabinet area above the control room, the primary control room on El. 166' represents over 85% of this volume. Consequently, a Murphy-Campe Geometry Factor generated from this total volume is conservative and leads to an underestimate of this geometry factor.

The GGNS control room is separated from the Auxiliary Building atmosphere by 5 feet of concrete (3 feet for the control building wall [46] and 3 feet for the Auxiliary Building wall [47]). The side walls and roof of the control room are 2 feet thick concrete [46,48]. Considering this shielding, the contribution to the control room dose from the Aux Bldg atmosphere and release plume is neglected.

6.0 RESULTS

The radiological consequences of a fuel handling accident over the core and racks are reported in Attachment 1 and summarized below.

CASE	DESCRIPTION	EAB DOSE (Rem TEDE)	CONTROL ROOM DOSE (Rem TEDE)
A	Drop of Irradiated Assembly Over Core with Release Partially Pulled through Equipment Hatch	1.762	0.518
B	Drop of Un-irradiated Assembly Over Racks without Secondary Containment	0.970	1.817
C	Drop of Irradiated Assembly Over Core without Secondary Containment after 7 days of decay	1.856	4.658
D	Drop of Un-irradiated Assembly Over Core without Secondary Containment	2.495	4.672

These results demonstrate that the design basis fuel handling accidents meet the proposed NRC acceptance criteria of 6 rem TEDE offsite and 5 rem TEDE in the control room with the application of the NUREG-1465 revised source terms. This calculation considered:

- Movement of irradiated fuel without secondary containment after 7 days of decay,
- Core alterations without secondary containment,
- No Control Room Fresh Air system, and
- No control room envelope (except for shielding).

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APPENDIX A: CONTROL ROOM DOSE MODEL

The control room is modeled as a homogeneously mixed volume with an inflow and an equivalent outflow. A filtered recirculation flow as well as a post-accident purge flow can also be considered if applicable. The transient activity concentration inside the control room can then be calculated relative to the activity concentration outside the control room envelope based on a postulated 2-hour release. If the release rate from secondary containment is constant, the activity concentration outside the control room is also constant and is defined by the product of the release rate from the secondary containment (Ci/s) and the control room χ/Q value (s/m^3). This model is illustrated below.

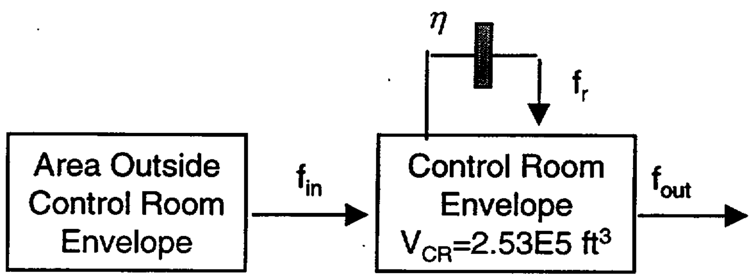

Figure A-1 Control Room Model

An activity balance for the control room concentration can then be developed for each isotope.

$$\frac{d}{dt}C_{CR}(t) = \frac{C_{out} \cdot f_{in}}{V_{CR}} - \frac{C_{CR}(t) * f_{out}}{V_{CR}} - \frac{C_{CR}(t) * f_r * \eta}{V_{CR}} - \lambda \cdot C_{CR}(t) \quad \text{(A-1)}$$

where:

$C_{CR}(t)$ = transient activity concentration in the control room
C_{out} = constant activity concentration outside the control room envelope
f_{in} = leakage rate into the control room
V_{CR} = volume of the control room
f_{out} = leakage rate out of the control room
f_r = recirculation flow rate
η = efficiency of the control room recirculation filters (isotope and species dependent)
λ = decay constant of isotope

Conservatively neglecting decay in the control room[7] and assuming the area outside the control room envelope is so large that the concentration is unchanged with the inleakage, the control room activity concentration can be written as:

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7 A majority of the short-term isotopes have already decayed to insignificant activities before the accident due to the 24-hour decay period required before moving irradiated fuel. In addition, since a majority of the control room TEDE dose is expected from I-131, which has an 8.04-day half-life, the decay of this important isotope is expected to be negligible over the duration of this calculation.

$$C_{CR}(t) = C_{out}\left(\frac{f_{in}}{f_{out} + f_r * \eta}\right)\left[1 - e^{-\left(\frac{f_{out}+f_r * \eta}{V_{CR}}\right) * t}\right] \qquad \text{(A-2)}$$

After the release ends, the concentration outside the control room would become zero and the control room concentration would decay off with the following expression where t_r is the time at which the release ends. At this time, the operators may initiate the Control Room Fresh Air system fans to draw fresh air in from the outside (at a flow of f_p in addition to the outleakage), further diluting the source terms in the control room. The control room source term concentration would then decrease exponentially by the following formula.

$$C_{CR}(t) = C_{CR}(t = t_r) \cdot e^{-\left(\frac{f_{out}+f_p+f_r * \eta}{V_{CR}}\right) * (t - t_r)} \qquad \text{(A-3)}$$

These equations would produce a transient control room activity that peaks at time t_r and decays off as the residual activity leaks out of the control room. This transient concentration is illustrated below for three inleakage flows, a 2.53E5 ft^3 control room volume, no recirculating filtered flow or purge, and a release period of 2 hours. As expected, for the simple case of no cleanup and no purge, the higher inleakage rate leads to the highest peak source term concentration; however, after the accident, the source term concentration drops off faster due to the higher inleakage.

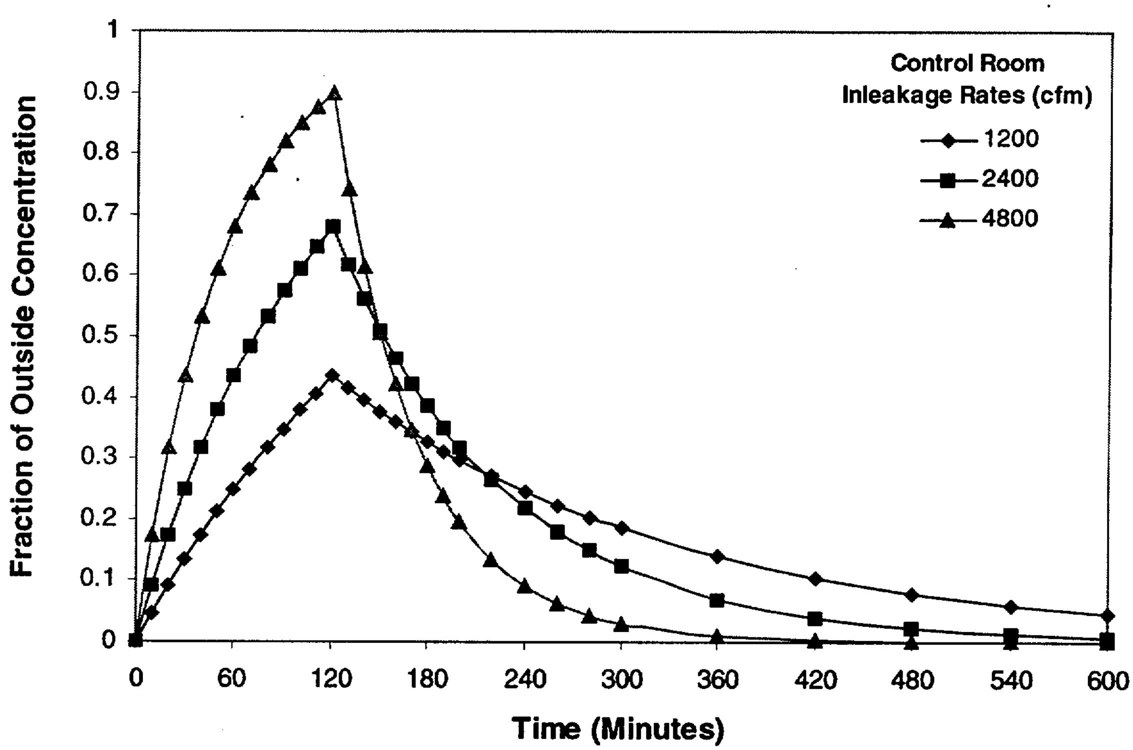

Figure A-2 Control Room Activity Transient

As expected, a filtered recirculation flow will reduce the peak activity concentration while a purge will expedite the concentration decrease after the accident as illustrated below for the 1200-cfm inleakage case.

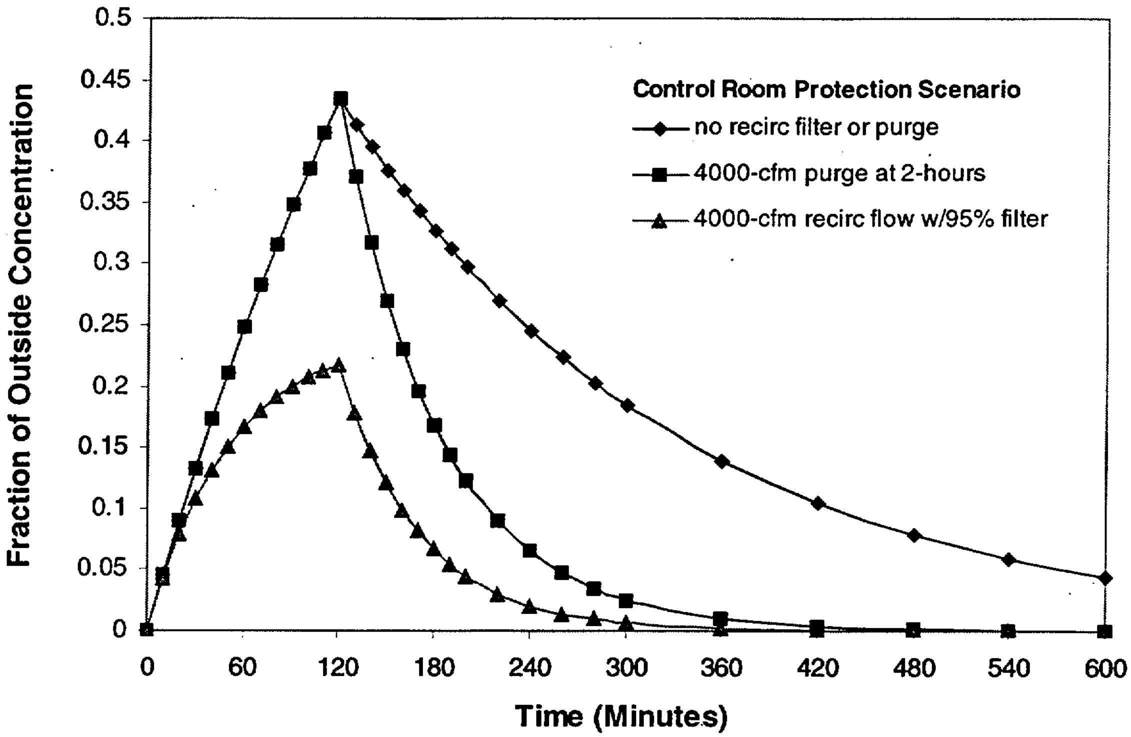

Figure A-3 Control Room Activity Transient for 1200-cfm Inleakage Case

The integrated dose to the control room personnel is based on the time integral of this transient concentration.

$$\int_0^\infty C_{CR}(t)dt = C_{out}\left(\frac{f_{in}}{f_{out}+f_r*\eta}\right)\left[t_r + \frac{V_{CR}}{f_{out}+f_r*\eta}\left(e^{-\left(\frac{f_{out}+f_r*\eta}{V_{CR}}\right)t_r}-1\right)+\left(\frac{V_{CR}}{f_{out}+f_p+f_r*\eta}\right)\left(1-e^{-\left(\frac{f_{out}+f_r*\eta}{V_{CR}}\right)t_r}\right)\right] \quad \text{(A-4)}$$

It is interesting to note that, for the simple case of no cleanup (f_r=0 and η=0) and no purge (f_p=0), Equation A-4 can be simplified such that the integral is not dependent on the inleakage rate since the inleakage (f_{in}) is equal to the outleakage (f_{out}).

$$\int_0^\infty C_{CR}(t)dt = C_{out}t_r$$

As can be seen in the scenarios illustrated in Figure A-2, the areas under each curve are the same and identical to the case of no control room envelope (*i.e.,* infinite leakage). The integrated exposure outside the control room envelope would be equal to the concentration outside the envelope (C_{out}) multiplied by the exposure period (t_r). The integrated doses for this simple case of no cleanup or purge can then be calculated for each isotope with the applicable dose conversion factor.

$$\mathrm{Dose}_{WB} = \frac{DCF_{WB}}{MCGF} \cdot \int_0^{\infty} C_{CR}(t)dt = \frac{DCF_{WB}}{MCGF} \cdot C_{out} \cdot t_r$$

$$\mathrm{Dose}_{Inhal} = DCF_{inhal} \cdot BR \cdot \int_0^{\infty} C_{CR}(t)dt = DCF_{inhal} \cdot BR \cdot C_{out} \cdot t_r$$

where:

DCF_{WB} is the air immersion dose conversion coefficient (Rem-m^3/Ci-s),
DCF_{inhal} is the inhalation effective dose conversion coefficient (Rem/Ci),
BR is the control room breathing rate = 3.47E-4 m^3/s,
MCGF is the Murphy-Campe Geometry Factor = $1173/V_{CR}^{0.338}$ = 17.5.

For a constant release, the activity concentration outside the control room (Ci/m^3) is constant and equal to the product of the release rate from the enclosure building (Ci/s) and the control room χ/Q value (s/m^3). Since the TEDE dose is the sum of the whole body and inhalation doses, the TEDE dose can be calculated for this simple case of no cleanup or purge as:

$$\mathrm{Dose}_{TEDE} = \left(\frac{DCF_{WB}}{MCGF} + DCF_{inhal} \cdot BR\right) \cdot C_{out} \cdot t_r = \frac{\chi}{Q} \cdot t_r \cdot \sum_i \left(\frac{DCF^i_{WB}}{MCGF} + DCF^i_{inhal} \cdot BR\right) \cdot \dot{Q}^i \qquad \text{(A-5)}$$

where $\dot{Q}^i$ is the release rate of isotope i (Ci/s) from the release point and χ/Q is the dispersion coefficient (s/m^3) from the release point to the control room intakes. Considering a constant release rate, $\dot{Q}^i$ is the activity of isotope i in the containment atmosphere (Q^i) divided by the release time, t_r and Equation A-5 reduces to:

$$\mathrm{Dose}_{TEDE} = \frac{\chi}{Q} \cdot \sum_i \left(\frac{DCF^i_{WB}}{MCGF} + DCF^i_{inhal} \cdot BR\right) \cdot Q^i \qquad \text{(A-6)}$$

CASE A RESULTS

EAB X/Q (s/m^3)=	9.56E-04									
Cont Room X/Q (s/m^3)=	2.75E-03									
Breathing Rate (m^3/s)=	3.47E-04									
Failed Rods=	142									
Radial Peaking Factor=	1.700									
Days of Decay=	1									
						FGR11&12 DCFs				
		Bundle	Release	Total	Total	Rem-m^3/Ci-s	Rem/Ci	EAB	CRFAS	Cont Room
	Isotope	Activity (Ci)	Fract	DF	Release (Ci)	W. Body	Inhal	TEDE	Factor	TEDE
	BR 82	1.698E+03	10%	8,188.0	4.207E-02	4.810000E-01	1.528100E+03	4.07E-05	1.00	6.45E-05
	BR 83	3.876E+01	10%	8,188.0	9.603E-04	1.413400E-03	8.917000E+01	2.97E-08	1.00	8.19E-08
	I130	5.233E+03	10%	8,188.0	1.296E-01	3.848000E-01	2.641800E+03	1.61E-04	1.00	3.35E-04
	I131	2.227E+05	12%	8,188.0	6.621E+00	6.734000E-02	3.289300E+04	7.27E-02	1.00	2.08E-01
	I132	2.789E+05	10%	8,188.0	6.910E+00	4.144000E-01	3.811000E+02	3.61E-03	1.00	2.96E-03
	I133	2.204E+05	10%	8,188.0	5.460E+00	1.087800E-01	5.846000E+03	1.12E-02	1.00	3.06E-02
	I135	3.599E+04	10%	8,188.0	8.916E-01	3.068780E-01	1.228400E+03	6.25E-04	1.00	1.09E-03
	KR 83M	1.499E+02	10%	1	3.041E+01	5.550000E-06	0.000000E+00	1.61E-07	1.00	2.65E-08
	KR 85	3.102E+03	15%	1	9.439E+02	4.403000E-04	0.000000E+00	3.97E-04	1.00	6.53E-05
	KR 85M	2.185E+03	10%	1	4.432E+02	2.767600E-02	0.000000E+00	1.17E-02	1.00	1.93E-03
	KR 88	7.203E+02	10%	1	1.461E+02	3.774000E-01	0.000000E+00	5.27E-02	1.00	8.67E-03
	XE129M	8.772E+00	10%	1	1.779E+00	3.922000E-03	0.000000E+00	6.67E-06	1.00	1.10E-06
	XE131M	2.703E+03	10%	1	5.483E+02	1.439300E-03	0.000000E+00	7.54E-04	1.00	1.24E-04
	XE133	4.360E+05	10%	1	8.845E+04	5.772000E-03	0.000000E+00	4.88E-01	1.00	8.02E-02
	XE133M	1.334E+04	10%	1	2.706E+03	5.069000E-03	0.000000E+00	1.31E-02	1.00	2.16E-03
	XE135	1.197E+05	10%	1	2.428E+04	4.403000E-02	0.000000E+00	1.02E+00	1.00	1.68E-01
	XE135M	5.765E+03	10%	1	1.169E+03	7.548000E-02	0.000000E+00	8.44E-02	1.00	1.39E-02
								1.762		0.518
						Regulatory Limit =		6		5

CASE B RESULTS

EAB X/Q (s/m^3)=	9.56E-04									
Cont Room X/Q (s/m^3)=	2.75E-03									
Breathing Rate (m^3/s)=	3.47E-04									
Failed Rods=	28									
Radial Peaking Factor=	1.700									
Days of Decay=	1									
						FGR11&12 DCFs				
		Bundle	Release	Total	Total	Rem-m^3/Ci-s	Rem/Ci	EAB	CRFAS	Cont Room
	Isotope	Activity (Ci)	Fract	DF	Release (Ci)	W. Body	Inhal	TEDE	Factor	TEDE
	BR 82	1.698E+03	10%	222.5	3.053E-01	4.810000E-01	1.528100E+03	2.95E-04	1.00	4.68E-04
	BR 83	3.876E+01	10%	222.5	6.968E-03	1.413400E-03	8.917000E+01	2.16E-07	1.00	5.94E-07
	I130	5.233E+03	10%	222.5	9.408E-01	3.848000E-01	2.641800E+03	1.17E-03	1.00	2.43E-03
	I131	2.227E+05	12%	222.5	4.804E+01	6.734000E-02	3.289300E+04	5.27E-01	1.00	1.51E+00
	I132	2.789E+05	10%	222.5	5.014E+01	4.144000E-01	3.811000E+02	2.62E-02	1.00	2.15E-02
	I133	2.204E+05	10%	222.5	3.962E+01	1.087800E-01	5.846000E+03	8.10E-02	1.00	2.22E-01
	I135	3.599E+04	10%	222.5	6.470E+00	3.068780E-01	1.228400E+03	4.53E-03	1.00	7.90E-03
	KR 83M	1.499E+02	10%	1	5.996E+00	5.550000E-06	0.000000E+00	3.18E-08	1.00	5.23E-09
	KR 85	3.102E+03	15%	1	1.861E+02	4.403000E-04	0.000000E+00	7.83E-05	1.00	1.29E-05
	KR 85M	2.185E+03	10%	1	8.740E+01	2.767600E-02	0.000000E+00	2.31E-03	1.00	3.80E-04
	KR 88	7.203E+02	10%	1	2.881E+01	3.774000E-01	0.000000E+00	1.04E-02	1.00	1.71E-03
	XE129M	8.772E+00	10%	1	3.509E-01	3.922000E-03	0.000000E+00	1.32E-06	1.00	2.16E-07
	XE131M	2.703E+03	10%	1	1.081E+02	1.439300E-03	0.000000E+00	1.49E-04	1.00	2.45E-05
	XE133	4.360E+05	10%	1	1.744E+04	5.772000E-03	0.000000E+00	9.62E-02	1.00	1.58E-02
	XE133M	1.334E+04	10%	1	5.336E+02	5.069000E-03	0.000000E+00	2.59E-03	1.00	4.25E-04
	XE135	1.197E+05	10%	1	4.788E+03	4.403000E-02	0.000000E+00	2.02E-01	1.00	3.31E-02
	XE135M	5.765E+03	10%	1	2.306E+02	7.548000E-02	0.000000E+00	1.66E-02	1.00	2.74E-03
								0.970		**1.817**
							Regulatory Limit =	**6**		**5**

CASE C RESULTS

EAB X/Q (s/m^3)=	9.56E-04									
Cont Room X/Q (s/m^3)=	2.75E-03									
Breathing Rate (m^3/s)=	3.47E-04									
Failed Rods=	142									
Radial Peaking Factor=	1.700									
Days of Decay=	7									
						FGR11&12 DCFs				
		Bundle	Release	Total	Total	Rem-m^3/Ci-s	Rem/Ci	EAB	CRFAS	Cont Room
	Isotope	Activity (Ci)	Fract	DF	Release (Ci)	W. Body	Inhal	TEDE	Factor	TEDE
	BR 82	1.005E+02	10%	222.5	9.161E-02	4.810000E-01	1.528100E+03	8.86E-05	1.00	1.41E-04
	BR 83	2.825E-17	10%	222.5	2.575E-20	1.413400E-03	8.917000E+01	7.97E-25	1.00	2.20E-24
	I130	1.628E+00	10%	222.5	1.484E-03	3.848000E-01	2.641800E+03	1.85E-06	1.00	3.83E-06
	I131	1.342E+05	12%	222.5	1.468E+02	6.734000E-02	3.289300E+04	1.61E+00	1.00	4.61E+00
	I132	3.851E-08	10%	222.5	3.511E-11	4.144000E-01	3.811000E+02	1.83E-14	1.00	1.51E-14
	I133	1.816E+03	10%	222.5	1.656E+00	1.087800E-01	5.846000E+03	3.38E-03	1.00	9.27E-03
	I135	2.204E-31	10%	222.5	2.009E-34	3.068780E-01	1.228400E+03	1.41E-37	1.00	2.45E-37
	KR 83M	4.315E-20	10%	1	8.753E-21	5.550000E-06	0.000000E+00	4.64E-29	1.00	7.63E-30
	KR 85	3.099E+03	15%	1	9.429E+02	4.403000E-04	0.000000E+00	3.97E-04	1.00	6.52E-05
	KR 85M	4.613E-07	10%	1	9.357E-08	2.767600E-02	0.000000E+00	2.48E-12	1.00	4.07E-13
	KR 88	3.908E-13	10%	1	7.927E-14	3.774000E-01	0.000000E+00	2.86E-17	1.00	4.70E-18
	XE129M	5.398E+00	10%	1	1.095E+00	3.922000E-03	0.000000E+00	4.11E-06	1.00	6.75E-07
	XE131M	2.106E+03	10%	1	4.271E+02	1.439300E-03	0.000000E+00	5.88E-04	1.00	9.66E-05
	XE133	2.129E+05	10%	1	4.319E+04	5.772000E-03	0.000000E+00	2.38E-01	1.00	3.92E-02
	XE133M	2.367E+03	10%	1	4.801E+02	5.069000E-03	0.000000E+00	2.33E-03	1.00	3.82E-04
	XE135	3.282E+00	10%	1	6.658E-01	4.403000E-02	0.000000E+00	2.80E-05	1.00	4.61E-06
	XE135M	7.131E-04	10%	1	1.447E-04	7.548000E-02	0.000000E+00	1.04E-08	1.00	1.72E-09
								1.856		**4.658**
							Regulatory Limit =	6		5

CASE D RESULTS

EAB X/Q (s/m^3)=	9.56E-04									
Cont Room X/Q (s/m^3)=	2.75E-03									
Breathing Rate (m^3/s)=	3.47E-04									
Failed Rods=	72									
Radial Peaking Factor=	1.700									
Days of Decay=	1									
						FGR11&12 DCFs				
		Bundle	Release	Total	Total	Rem-m^3/Ci-s	Rem/Ci	EAB	CRFAS	Cont Room
	Isotope	Activity (Ci)	Fract	DF	Release (Ci)	W. Body	Inhal	TEDE	Factor	TEDE
	BR 82	1.698E+03	10%	222.5	7.850E-01	4.810000E-01	1.528100E+03	7.59E-04	1.00	1.20E-03
	BR 83	3.876E+01	10%	222.5	1.792E-02	1.413400E-03	8.917000E+01	5.54E-07	1.00	1.53E-06
	I130	5.233E+03	10%	222.5	2.419E+00	3.848000E-01	2.641800E+03	3.01E-03	1.00	6.24E-03
	I131	2.227E+05	12%	222.5	1.235E+02	6.734000E-02	3.289300E+04	1.36E+00	1.00	3.88E+00
	I132	2.789E+05	10%	222.5	1.289E+02	4.144000E-01	3.811000E+02	6.74E-02	1.00	5.53E-02
	I133	2.204E+05	10%	222.5	1.019E+02	1.087800E-01	5.846000E+03	2.08E-01	1.00	5.70E-01
	I135	3.599E+04	10%	222.5	1.664E+01	3.068780E-01	1.228400E+03	1.17E-02	1.00	2.03E-02
	KR 83M	1.499E+02	10%	1	1.542E+01	5.550000E-06	0.000000E+00	8.18E-08	1.00	1.34E-08
	KR 85	3.102E+03	15%	1	4.786E+02	4.403000E-04	0.000000E+00	2.01E-04	1.00	3.31E-05
	KR 85M	2.185E+03	10%	1	2.247E+02	2.767600E-02	0.000000E+00	5.95E-03	1.00	9.77E-04
	KR 88	7.203E+02	10%	1	7.409E+01	3.774000E-01	0.000000E+00	2.67E-02	1.00	4.39E-03
	XE129M	8.772E+00	10%	1	9.023E-01	3.922000E-03	0.000000E+00	3.38E-06	1.00	5.56E-07
	XE131M	2.703E+03	10%	1	2.780E+02	1.439300E-03	0.000000E+00	3.83E-04	1.00	6.29E-05
	XE133	4.360E+05	10%	1	4.485E+04	5.772000E-03	0.000000E+00	2.47E-01	1.00	4.07E-02
	XE133M	1.334E+04	10%	1	1.372E+03	5.069000E-03	0.000000E+00	6.65E-03	1.00	1.09E-03
	XE135	1.197E+05	10%	1	1.231E+04	4.403000E-02	0.000000E+00	5.18E-01	1.00	8.52E-02
	XE135M	5.765E+03	10%	1	5.930E+02	7.548000E-02	0.000000E+00	4.28E-02	1.00	7.03E-03
								2.495		**4.672**
							Regulatory Limit =	**6**		**5**