CALCULATION TITLE PAGE

Total Number of Pages: 26

MP2 Spent Fuel Cask Drop- Dose Analysis

TITLE

CASKDOSE-02686-R2	0	NA
CALCULATION No.	**Revision No.**	**System Name**

NA	NA	NA	CSK
VENDOR CALCULATION No.	**Structure**	**System Number**	**Component**

NUCLEAR INDICATOR:	Safety Evaluation or Screen Attached	Calc. Supports DCR/MMOD?	Calc. Supports Other Process?
☒CAT1 ☐RWQA ☐SBOQA ☐FPQA ☐ATWSQA ☐NON-QA	☐YES ☒NO	☐YES ☒NO	☒YES ☐NO

INCORPORATES:

CCN NO:	AGAINST REV.
NA	NA

NA	BARCR 99-MP2-81 PTSCR 2-13-99
DCR/MMOD No.	**Reference**

Executive Summary

The dose consequences of a spent fuel cask drop are within the acceptance criteria of GDC 19 and 10CFR100 provided that the EBFS system is aligned to AEAS mode during cask movement over the spent fuel pool. This mode of operation does allow the Aux. Bldg elevator shaft smoke hole to remain open during the cask activities.

This calculation bounds the dose consequences of a fuel handling accident involving the drop of a consolidated fuel canister .

This calculation will provide the basis for a PTSCR. The safety screen will be performed as part of the PTSCR package.

This calculation supersedes XX-XXX-113RA, 81-166-86RA and 82-232-88RA.

Calculation results are conditional based upon manual initiation of spent fuel pool ventilation in AES mode prior to cask movement over the pool.

Approvals (Print & Sign Name)		
Preparer: S. M. Torf		Date: 9/23/99
Interdiscipline Reviewer: NA	Discipline:	Date:
Interdiscipline Reviewer :NA	Discipline:	Date:
Independent Reviewer: J. L. Wheeler		Date: 9/23/99
Supervisor: W. J. Eakin		Date: 9/23/99

Installation Verification

☐ Calculation represents the installed configuration and approved licensing condition (Calculation of Record)

☐ N/A does not affect plant configuration (e.g., study, hypothetical analysis, etc.)

Preparer/Designer Engineer: (Print and Sign) Date:

~~MP2 ENG Initial:~~

ORB Date: 9/23/99

PassPort DATABASE INPUTs

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Calculation Number: CASKDOSE-02686-R2 Revision: 0

Vendor Calculation Number/Other: NA Revision: NA

CCN # NA QA ☒ Yes ☐ No Calc Voided: ☐ Yes ☒ No

Superseded By: NA Supersedes Calc: XX-XXX-113RA, 81-166-86RA, 82-232-88RA

Discipline (Up to 10) Z

Unit	Project Reference (EWA, DCR or MMOD)	Component Id	Computer Code	Rev. No./ Level No.
MP2	NA	RM-9799A	TACT III	83.0
		RM-9799B	CRADLE	2
		RM-8139		
		RM-8142		
		RM-8156		
		RM-8157		

PMMS CODES*					
Structure	System	Component	Reference Calculation	Rev No.	CCN
NA	NA	CSK	NUC-181	1	NA
			07077.13-WM(B)-03	0	NA
			96RAD-1378R2	2	NA
			XX-XXX-37RA	2	NA
			97ENG-01757M2	0	1
			WM(B)-05	0	NA
			81-115-81RA	0	NA
			07077.13-WM(B)-02	0	NA
			UR(B)-453	0	1

*The codes required must be alpha codes designed for structure, system and component.
*Use a separate line to post information to be entered (one document per line).

Reference Drawing	Sheet	Rev. No.
25203-27022	1	5
25203-27015	1	21

Comments:
NA

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1. PURPOSE

The purpose of this calculation is to determine the maximum dose at the EAB, LPZ and MP2 Control Room, attributable to a spent fuel cask dropped in the MP2 spent fuel pool.

This calculation addresses issues identified in CRs M2-98-2112, and M2-99-0414.

2. SUMMARY OF RESULTS

The MP2 Control Room, EAB and LPZ doses are summarized below. It should be noted that these doses assume a 2% EBFS bypass (the smoke hole cover is open) and that the EBFS is operating in AES mode per Reference 5. It should be noted that a Tech Spec change request is under development to require that EBFS be in AES mode of operation during cask movement over the spent fuel pool. This change would allow cask movement over the spent fuel pool with the smoke hole cover open. The results of this calculation do not require automatic initiation of control room isolation by spent fuel pool area radiation monitors or control room inlet radiation monitors.

	EAB	LPZ	Control Room
Thyroid, rem	1.62E-03	3.62E-04	4.75E-01
Whole Body, rem	1.26E-01	3.27E-02	3.25E-01
Beta Skin, rem	N/A	N/A	2.51E+01

All regulatory acceptance criteria have been met. The doses listed above are "well within" the 10 CFR 100 exposure guidelines (300 rem -thyroid and 25 rem - whole body). "Well within" is defined as less than 25% or less of the 10CFR100 exposure guideline values per SRP 15.7.5. In addition, GDC 19 criteria are met. The GDC 19 criteria for the control room is listed below and states:

> *"Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident."*

This calculation bounds the dose consequences from the drop of 1 consolidated fuel canister (CFC). The source term used in this calculation includes 1560 fuel assemblies, 1376 fuel assemblies of which are from 688 CFCs (aged 5 years) and the remainder are 184 assemblies (aged 1 year).

3. REFERENCES/DESIGN INPUTS

1. ERC 25203-ER-98-0207, Rev. 0, "Fuel Assemblies Stored Within a 21 Foot Radius of the MP2 Spent Fuel Pool Cask Lay down Area", dated 8/21/98

2. MP2 Technical Specifications

3. GCE-85-435, "Cask Drop - MP2 Spent fuel Pool Re-rack", R. N. Smart to D. W. Marzilli, dated July 10, 1985

4. NUREG-0383, "Directory of Certificates of Compliance for Radioactive Materials Packages"

5. SP 2614F, Rev. 0, "Fuel Handling Area Integrity"

6. DBS-2314G, "Design Basis Summary - Enclosure Building Filtration System"

7. SRP 15.7.4, "Radiological Consequences of a Fuel Handling Accident"

8. SRP 15.7.5, "Spent Fuel Cask Drop Accidents"

9. Regulatory Guide 1.25, "Assumptions used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors"

10. ICRP 38, Vols 11 - 13, 1983, Radionuclide Transformations

11. MP2 Facility Operating License, Docket 50-336

12. Regulatory Guide 1.49, "Power Levels of Nuclear Plants"

13. NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light water Reactors"

14. Calc. # NUC-181, Rev. 1, "MP2 Design Basis Loss of Coolant Accident - Radiation Source Terms", dated 6/19/98

15. Radiological Health Handbook, 1/1970

16. Drawing # 25203-27022, Rev. 5, "General Arrangement - Containment & Aux. Bldg, Section B-B"

17. NRC Letter A13531, "Issuance of License Amendment 208, MP2 Enclosure Building Integrity", dated 9/30/97

18. Calc. # 07077.13-WM(B)-03, Rev. 0, "Normalized concentrations (X/Q) at the EAB and LPZ for Gaseous Releases from the Unit 1 Stack and the Unit 2 Containment, Stack, PORV's/ ADV's and MSLB", 7/25/1998

19. ICRP 30, Supplement to Part 1, Volume 3, "Annals of the ICRP, Limits for Intakes of Radionuclides by Workers"

20. TACT III, Atmospheric Transport Code System, Oak Ridge National Laboratory, CCC-447, Version 83.0 (Operated on the Wethersfield IBM 3090 Mainframe)

21. Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I "

22. Regulatory Guide 1.4, Rev. 2, "Assumptions used for Evaluating the Potential Radiological consequences of a Loss of coolant Accident for Pressurized Water Reactors"

23. Intentionally Blank

24. Memo RLW:98:253, "Results of Fuel Rod Gas Pressure Analysis for Fuel Handling Accident for Millstone Unit 2", R. L. Westcott (Project Manager - Siemens) to C. H. Wu (NUSCO), dated 7/27/98 (copy is attached in Attachment B)`

25. Calc. # 96RAD-1378R2, Rev. 2, "MP2- Fuel Handling Accident in the Spent Fuel Pool with 2% EBFS Bypass", dated 8/19/98

26. CRADLE, Control Room Accident Dose Level Evaluations, QA Category 1 Calculation #XX-XXX-37 RA, Rev. 2, Level 1, D. Miller, 11/22/82, (Operated on the Wethersfield IBM 3090 Mainframe)

27. Dwg# 25203-27015, Rev. 21, General Arrangement Aux. Bldg Plan El. 36'6" and 38'6"

28. Intentionally Blank

29. Intentionally Blank

30. Calc. # 81-115-81RA, Rev. 0, "Control Room Radiation Monitor Setpoint Level", dated 1/25/85

31. Calc. # UR(B)-453, Rev. 0, "MP2 Control Room Operator Doses Following a MP3 LOCA Assuming Duct Leakage and Damper Bypass", CCN # 1, dated 4/16/99

32. Intentionally Blank

33. Regulatory Guide 1.25, Rev 0, Assmptions used for Evaluating the Potential Rad Consequences of a FHA for BWRs and PWRs, dated 3/23/72

34. Calc. # 97ENG-01757M2, Rev. 0 , CCN# 1, "Temperature Analysis of the Auxiliary Building during Normal Plant Operations" , dated 10/29/98

35. Calc. # WM(B)-05, Rev. 0, X/Qs at Unit 2 Control room Intake From the top of the Unit 1 Stack Under fumigation Conditions", dated 7/8/98

36. Calc. # 07077.13-WM(B)-02, Rev. 0, "Normalized X/Qs at the Unit 2 & 3 Control Room and TSC for Releases From Unit 2", dated 7/29/98

37. ERC 25203-ER-98-0050, Rev. 02, "Control Room Filtration Design Basis Parameters...", dated 6/10/98

4. DATA AND ASSUMPTIONS

1. Based on Reference 3.1, there are 4 different, worst case spent fuel scenarios, based upon 21 feet from the outer edge of the cask pit. (This is more conservative than the Tech. Spec 3.9.16.1 requirement for use of the center of the spent fuel cask set-down area.). It should be noted that if any of the contents of an entire fuel storage module (or rack) is damaged, then it has been assumed that all within the module will be damaged.

   a) Case 1: 1097 fuel assemblies, 6 of which were consolidated assemblies. This case assumes no further consolidation.

   b) Case 2: 1117 fuel assemblies, 6 of which were consolidated assemblies. This case assumes Case 1 quantities plus 40 blocked (empty) locations will be opened to allow fuel storage.

   c) Case 3: 1782 fuel assemblies, 1376 of which were consolidated assemblies. This case assumes maximum consolidation.

   d) Case 4: 1802 fuel assemblies, 1376 of which were consolidated. This case assumes Case 3 quantities plus 40 blocked (empty) locations will be opened to allow fuel storage.

2. Based on Reference 3.2, Section 3.9.20, "Spent Fuel Pool - Consolidation", consolidated fuel assemblies must be at least 5 years decayed.

3. Based on Reference 3.2, Section 3.9.16.1, "Shielded Cask", whenever a shielded cask is on the refueling floor, all fuel within a distance "L" from the center of the spent fuel cask set-down area shall have decayed for at least 1 year.

4. Based on Reference 3.3, the distance "L" is defined as 21 feet. "L" is the major cask dimension.

a) Reference 3.4 was reviewed to validate the 21 foot dimension. The largest cask identified was the TN-9 (C of C# 9016), with the longest dimension of 18.9 feet and a diagonal measurement of 19.7 feet.

5. Based upon Reference 3.5, the Enclosure Building Filtration System (EBFS) is placed in Auxiliary Exhaust System (AES) mode when a cask (or heavy load) is moved over the spent fuel pool. The EBFS contains charçoal and HEPA filters (Reference 3.6). The adequacy of the ventilation envelope is assured by Tech. Spec. 3.9.14 which requires closure of specific doors. (It should be noted here that a Tech. Spec. change is in progress to require AES operation during cask movement over the spent fuel pool.)

6. The radial peaking factor is assumed to be 1 due to the large number of fuel bundles involved in this analysis.

7. The I-129 thyroid dose conversion factor in Reference 3.19 is:
   a) I-129 thyroid - CDE - inhalation - DCF = 1.6E-06 Sv/ Bq = 5.92E+06 Rem/ Ci inhaled.

8. Based upon References 3.11 and 3.12, the reactor power level is 2754 MW_t. Ref. 3.11 provides a licensed power level of 2700 MW_t and Ref. 3.12 requires a 2% margin on the power level in the radiological analysis. 2700 x 1.02 is 2754.

9. Based upon Reference 3.9,
   a) core activity available for release is 10% of core Noble gases (except Kr-85), 30% of core Kr-85 and 10% of core iodines. It should be noted that fuel stored in the spent fuel pool is a mixture of burnups, some of which exceed the 25,000 MWd/Ton average burnup. Based upon Ref. 3.13, the applicability of the release fractions can be applied to extended burnup fuel except for I-131 which should be 12%. With the 1 year and 5 year requirements for spent fuel decay, I-131 activity will be insignificant, therefore of no consideration in this calculation.
   b) the iodine chemical form is 99.75% inorganic and 0.25% organic and based upon the pool DF for iodine of 133 for inorganic and 1 for organic, an effective DF is provided of 100. this results in a release fraction of 75% inorganic and 25% for organic iodines.
   c) the filter removal efficiency for iodine is 90% for inorganic and 70% for organic
   d) the radioactive release occurs over a two hour time frame

10. Breathing rate for persons off-site is based on Reference 3.22
   a) 0-8 hrs: 3.47E-04 m^3/sec
   b) 8 - 24 hrs: 1.75E-04 m^3/sec
   c) 24 -720 hrs: 2.32E-04 m^3/sec

11. Breathing rate for control room is 3.47E-04 m^3/sec, based on Reference 3.26. NOTE: The CRADLE code stores breathing rates internal to the code but this is used for the I-129 determination and is consistent with the code.

12. Intentionally Blank

13. An elevated, main stack release can be assumed because:
    a) Reference 3.9 requires a stack height that is at least two and one half times the height of any structure close enough to affect plume dispersion
    b) the height of the Enclosure Building is 147 ft above grade. (Reference 3.16)
    c) the stack height is 375 ft (Reference 3.17)

14. EAB and LPZ atmospheric diffusion factors, χ / qs, for a stack release are provided in Reference 3.18 and are (sec/m^3):

			Unit 1 STACK (fumigation)	Unit 2 STACK
a)	EAB	(0-2 hr):	1.00E-04	3.66E-04
b)	LPZ	(0-4 hr):	2.69E-05	4.8E-05

15. MP2 Control Room atmospheric diffusion factors, χ / qs, for a <u>Unit 1 stack</u> release are provided by Reference 3.35, and they are (sec/m^3):

	Unit 2 Control Room
0 - 4 hours	2.51E-04
4 - 8	1.96E-05
8 - 24	5.46E-06
24 - 96	3.43E-07
96 - 720	6.44E-09

16. Based on Reference 3.24 and 3.13 (page 3-11), the assumptions used in Regulatory Guide 1.25 (Reference 3.9) with regard to peak linear power density, center-line fuel temperature, rod pressurization and burnup are appropriate with the exception of I-131 release rates. Since the fuel is decayed at least 1 year, I-131 will have gone through approximately 46 half-lives and therefore is not a consideration.

17. Atmospheric diffusion factor, χ / qs, for a <u>Unit 2 Stack</u> release to the MP2 Control Room is provided by Reference 3.36 and for 0 to 2 hours post-accident is 2.92E-03 (sec/m^3). The smoke hole is normally open to the Aux. Bldg which exhausts through the main exhaust system out the Unit 2 stack.

18. Intake flow rate prior to control room isolation: 800 CFM (Ref. 3.25). It is assumed that the outflow matches the inflow to avoid pressurization of the control room.

19. Control Room volume: 35,650 ft^3. (Ref 3.25)

20. Based on Reference 3.10, the average gamma energies per disintegration for Kr-85 and Xe-133 are 0.002 and 0.046 Mev/ dis, respectively.

21. Based on Reference 3.30, detector response factor is 1 mr/hr = 0.009 uCi/cc DEQ Xe-133

22. Based on Reference 3.2, (Tech. Spec. 3.4.3.3), the control room rad. monitors will isolate the control room at 2 mr/hr. NOTE: procedurally, lower alarm setpoints are used but procedures do not have the stability of a Tech. Spec.

23. The fraction of release bypassing the EBFS via the smoke hole is 2%. (Reference 25)

24. Based upon Reference 3.14 for core inventory and # of fuel bundles and Reference 3.15 for isotopic half-lives:

a) total # of fuel bundles in a core is 217

b) core iodine and noble gas inventory at time of shutdown is

ISOTOPE	INVENTORY, Ci	$T_{1/2}$, hours
I129	4.137E+00	1.40E+11
I130	5.101E+06	1.24E+01
I131	7.719E+07	1.94E+02
I132	1.105E+08	2.28E+00
I133	1.504E+08	2.08E+01
I134	1.666E+08	8.73E-01
I135	1.407E+08	6.70E+00
I136	6.673E+07	2.36E-02
KR83M	1.080E+07	1.86E+00
KR85	1.194E+06	9.41E+04
KR85M	2.451E+07	4.40E+00
KR87	4.860E+07	1.27E+00
KR88	6.865E+07	2.79E+00
KR89	8.593E+07	5.30E-02
KR90	8.522E+07	8.97E-03
XE131M	8.615E+05	2.87E+02
XE133	1.569E+08	1.26E+02
XE133M	4.808E+06	5.42E+01
XE135	5.658E+07	9.16E+00
XE135M	3.104E+07	2.62E-01
XE137	1.316E+08	6.37E-02
XE138	1.316E+08	2.37E-01
XE139	1.050E+08	1.11E-02

25. Based on Reference 3.37,

1. control room recirculation starts 10 minutes after control room isolation
2. the unfiltered inleakage rate is 130 cfm

5. METHOD OF CALCULATION

This analysis of the MP2 spent fuel pool cask drop event uses the assumptions provided in Section 4. Analyses will be performed assuming the elevator smoke hole cover is open, resulting in 2% bypass of EBFS.

The TACT III (version 83.0) computer code (Reference 3.20) was used in this calculation. It is used to determine the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) doses in support of 10 CFR 100 requirements.

The CRADLE (Version 2) computer code (Reference 3.26) is used for control room dose calculations used in this calculation.

Both TACT III and CRADLE are run on the IBM 3090 mainframe computer located in Wethersfield, CT.

This analysis assumes that a cask with major dimension of 21 feet is dropped in the spent fuel pool. Technical specification requirements are that all fuel within a distance "L" from the center of the cask set-down area be at least 1 year decayed.

The number of fuel assemblies that can be affected is a combination of consolidated and unconsolidated fuel assemblies. Unconsolidated fuel assemblies are assumed to be 1 year decayed and consolidated assemblies are assumed to be 5 years decayed. Two evaluations will be performed using 1) the maximum number of unconsolidated assemblies with existing 6 consolidated bundles and 2) the maximum number of consolidated bundles with the remainder of unconsolidated bundles.

The release rate will be determined to evaluate whether the control room ventilation inlet radiation monitors (Control Room Ventilation Intake RM9799 A&B) will isolate on a radiation signal as a result of this release. In addition, the spent fuel pool area radiation monitors (RM 8139/ 8142/ 8156/ 8157) will be evaluated for impact.

Thyroid dose conversion factors currently used in the TACTIII and CRADLE codes are from Reg. Guide 1.109 (Reference 3.21). The R.G. 1.109 factors are more than 20 years old. ICRP 30 (Reference 3.19) provides more up-to-date thyroid dose conversion factors and will be used to adjust the TACT III output to ICRP 30 factors using a simple ratio.

It should be noted that CRADLE does not address I-129. The dose from I-129 will be determined assuming release from the Unit 2 stack with no credit taken for any EBFS or control room charcoal filtration. The dose is determined by taking the amount of I-129 (release rate) released from the pool multiplied by the thyroid dose conversion factor, the appropriate Chi/ Q, and the duration of the release. This results in a conservative thyroid dose determination.

6. BODY OF CALCULATION

6.1 SOURCE TERM - RELEASE ACTIVITY

Assumption 4.24 provides the core inventory and # of fuel assemblies in a core at shutdown. In Table 6.1 the inventory is adjusted for decay (at 1 year for unconsolidated assemblies and at 5 years for consolidated assemblies) and then divided by the number of fuel assemblies (217) to get an isotopic inventory/ assembly determination[a]. This is summarized in Table 6.1. As can be seen in the table, the only isotopes of interest are I-129 (0.02 Ci/ assembly) and Kr-85 (5159 Ci/ assembly). All other isotopes have decayed to less than 7E-10 times their original activity (at least 30 half-lives will have passed) and are not considered to provide any significant dose contribution.

TABLE 6.1 Fuel Assembly Isotopic Activity Determination

ISOTOPE	INVENTORY, Ci	$T_{1/2}$, hours FROM RAD. HEALTH HANDBOOK, TABLE OF NUCLIDES 1/1970	1 YEAR INVENTORY PER FUEL ASSEMBLY	5 YEAR INVENTORY PER FUEL ASSEMBLY
I129	4.137E+00	1.40E+11	0.02	0.02
I130	5.101E+06	1.24E+01	not significant	not significant
I131	7.719E+07	1.94E+02	not significant	not significant
I132	1.105E+08	2.28E+00	not significant	not significant
I133	1.504E+08	2.08E+01	not significant	not significant
I134	1.666E+08	8.73E-01	not significant	not significant
I135	1.407E+08	6.70E+00	not significant	not significant
I136	6.673E+07	2.36E-02	not significant	not significant
KR83M	1.080E+07	1.86E+00	not significant	not significant
KR85	1.194E+06	9.41E+04	5159	3985
KR85M	2.451E+07	4.40E+00	not significant	not significant
KR87	4.860E+07	1.27E+00	not significant	not significant
KR88	6.865E+07	2.79E+00	not significant	not significant
KR89	8.593E+07	5.30E-02	not significant	not significant
KR90	8.522E+07	8.97E-03	not significant	not significant
XE131M	8.615E+05	2.87E+02	not significant	not significant
XE133	1.569E+08	1.26E+02	not significant	not significant
XE133M	4.808E+06	5.42E+01	not significant	not significant
XE135	5.658E+07	9.16E+00	not significant	not significant
XE135M	3.104E+07	2.62E-01	not significant	not significant
XE137	1.316E+08	6.37E-02	not significant	not significant
XE138	1.316E+08	2.37E-01	not significant	not significant
XE139	1.050E+08	1.11E-02	not significant	not significant

Of the 4 scenarios described in Assumption 4.1, Cases 2 and 4 represent the worst cases based on 21 feet from the edge of the cask laydown area (an adjustment will be made later

[a] It should be noted that no credit is taken for the actual age of the fuel bundles. Considering that there are almost 9 complete reactor cores used in this analysis, much of the fuel would be older than 5 years.

in this section to address the number of assemblies that are 21 feet from the center of the laydown area). It should be noted that these cases represent a potential configuration which would allow more spent fuel storage in the spent fuel pool than currently allowed, but it is conservative.

The first scenario (Case 2) assumes the current pool inventory of 3 consolidated fuel canisters (3 CFCs), and 1111 regular assemblies. A consolidated fuel canister contains 2 fuel assemblies that have decayed at least 5 years. The source term for the 1117 assemblies (including the 3 consolidated boxes containing 6 assemblies) is as follows:

1111 unconsolidated assemblies (1 year decay)
6 consolidated assemblies (5 year decay)

Based upon Table 6.1, the only isotopes of any significance, after a 1 year decay are I-129 and Kr-85. I-129 and Kr-85 activity for the 1117 assemblies is determined by multiplying the number of 1 or 5 year decayed assemblies by the corresponding activity per assembly from Table 6.1. Based on Table 6.2, the activity available for release for this scenario is 22 Ci I-129, and 5.8E06 Ci Kr-85.

Table 6.2 - Scenario 1/ Case 2 isotopic inventory

	I-129 Activity, Ci	Kr-85 Activity, Ci
1111 assemblies, 1 year decay	1111 x 0.02 = 22.2 Ci	1111 x 5159 = 5.73E06 Ci
6 assemblies, 5 year decay	6 x 0.02 = 0.1Ci	6 x 3985 = 2.39E04 Ci
TOTALS	22 Ci I-129	5.8E+06 Ci Kr-85

The second scenario (Case 4) assumes the maximum number of consolidated fuel assemblies. There are a total of 1802 assemblies, 1376 of which have been consolidated. The source term for the 1802 assemblies (assuming the licensed 688 consolidated fuel boxes containing 1376 assemblies) is based on Table 6.3, the activity available for release for this scenario is 36 Ci I-129, and 7.7E06 Ci Kr-85.

426 unconsolidated assemblies (1 year decay)
1376 consolidated assemblies (5 year decay)

Table 6.3- Scenario 2/ Case 4 isotopic inventory

	I-129 Activity, Ci	Kr-85 Activity, Ci
426 assemblies, 1 year decay	426 x 0.02 = 8.5 Ci	426 x 5159 = 2.2E06 Ci
1376 assemblies, 5 year decay	1376 x 0.02 = 27.5 Ci	1376 x 3985 = 5.5E06 Ci
TOTALS	36.0 Ci I-129	7.7E06 Ci Kr-85

The second scenario (Table 6.3) provides the larger source term.

At this point, clarification should be made with regard to the number of damaged fuel bundles. Reference 3.1 provided the number of damaged fuel bundles within 21' of the edge (as opposed to center) of the cask laydown area (per request of the author of this calculation). This resulted in an extremely conservative number (1802) of damaged fuel bundles. Preliminary calculations showed that this amount of damage may potentially result in exceeding skin dose criteria for control room habitability. The amount of fuel damage will be redetermined based on 21 ft from the center of the cask laydown area which is consistent with Tech. Spec 3.9.16.1. From Reference 27 it can be shown that the center of the cask laydown area removes approximately an additional 5.5 ft of radius from the fall pattern of the cask drop. This effectively results in damaging all fuel within 15.5' (21' - 5.5') of the edge of the cask laydown area instead of 21 ft. As a result, using Reference 3.1, it can be shown that Modules N14, N15 and N16 will not be damaged. These modules contain a total of 242 assemblies (Ref. 3.1). Table 6.3 will be adjusted below to account for the removal of 242 assemblies.

Table 6.4 - Worst case fuel damage @ 21 ft from center of cask laydown area.

	I-129 Activity, Ci	Kr-85 Activity, Ci
184 assemblies, 1 year decay	184 x 0.02 = 3.7 Ci	184 x 5159 = 9.5E05 Ci
1376 assemblies, 5 year decay	1376 x 0.02 = 27.5 Ci	1376 x 3985 = 5.5E06 Ci
TOTALS	31.2 Ci I-129	6.5E06 Ci Kr-85

The released activity is based on release fractions . Release fractions are required for the 98% EBFS/ 2% bypass. Using the method in Reference 3.25, release fractions (F_c) are based on peaking factor (PF =1, Assumption 4.6), DF (100 for iodine, 1 for noble gas, Assumption 4.9.b), fraction available for release from the core, F_r (10% iodine and 30% Kr-85, Assumption 4.9.a) and fraction for release path, F_b (98% through EBFS and 2% bypass).

$F_c = PF * F_r * F_b / DF$

The resulting release fractions are

98% EBFS operation	(I-129):	1.00 * 0.1 * 0.98 / 100 =	9.80E-04
	(KR-85):	1.00 * 0.3 * 0.98 / 1 =	2.94E-01
2% Bypass	(I-129):	1.00 * 0.1 * 0.02 / 100 =	2.00E-05
	(KR-85):	1.00 * 0.3 * 0.02 / 1 =	6.00E-03

The corresponding source terms used in the TACT runs are

98% EBFS operation	(I-129):	9.80E-04 * 31.2 =	3.1E-02 Ci
	(KR-85):	2.94E-01 * 6.5E+06 =	1.9E+06 Ci
2% Bypass	(I-129):	2.00E-05 * 31.2 =	6.2E-04 Ci
	(KR-85):	6.00E-03 * 6.5E+06 =	3.9E+04 Ci

6.2 RAD. MONITOR RESPONSE TO INITIATE CONTROL ROOM ISOLATION

6.2.1 Control Room Ventilation Intake Rad. Monitors

The possibility for control isolation via the control room rad. monitors (Control Room Ventilation Intake RM 9799 A&B) is evaluated here. The scenario evaluated here assumes AES mode of operation.

The release path via the 2% bypass is from the elevator shaft smoke hole to the Main Exhaust system, which vents out of the MP2 Stack. From Reference 3.36, the MP2 Stack is located 41.5 m from the MP2 Control Room ventilation intake with an associated windspeed of 2.5 m/sec. Based on the distance and windspeed, the transit time for the release to get from the stack to the intake is 16.6 seconds

The 98% EBFS portion of the release is vented through the MP1 Stack. The MP1 stack is further from the control room air inlet than the MP2 stack and therefore transit time will be greater than 16.6 seconds.

The catastrophic failure of almost 1600 fuel bundles, would result in a release to the air in the Spent Fuel Building of 1.939E+06 Ci of Kr-85 and 3.162E-02 Ci I-129 (determined in Section 6.1). The building volume that this release is estimated to envelope is 443,434 ft^3 (Reference 34). The volume does include the railroad access area and as such provides a larger dilution volume and subsequently a lower airborne activity which would reduce the activity that the radiation monitors will see. Because I-129 is not included in CRADLE (nor is it significant from a dose point of view - see Section 6.4.1), it will not be credited here as providing any dose rate input to control room rad. monitors.

The dose rate that the control room rad. monitors (Control Room Ventilation Intake RM 9799 A&B) would see is calculated below. For simplicity, a constant release rate is assumed. This is conservative because it results in a lower dose rate seen by the control room inlet rad. monitors than if a puff release occurred (lower dose rates reduce the probability of control room isolation on a high radiation signal). Data inputs are :

Kr-85 activity released: 1.939E+06 Ci Kr-85

Release time: 2 hours (7200 seconds)

U2 Control Room Chi/ Q for U1 stack release: 2.51E-4 sec/ m^3

U2 Control Room Chi/ Q for U2 Stack release: 2.92E-03 sec/ m^3

Bypass fraction : 2%

The resultant activity at the control room monitor from 98% EBFS is:

1.939E+06/ 7200 * 2.51E-04 * 98% = 6.7E-02Ci/m^3 = 6.7E-02 uCi/cc

The resultant activity at the control room monitor from 2% Bypass is :

1.939E+06/ 7200 * 2.92E-03 * 2% = 1.6E-02 Ci/m^3 = 1.6E-02 uCi/cc

The rad. monitors are set to respond to a Xe-133 dose equivalent. To convert the Kr-85 to a Xe-133 equivalent , the responses must be multiplied by the Kr-85: Xe-133 average gamma energy ratio. From Assumption 3.20, the Kr-85 and Xe-133 average gamma energies per disintegration are 0.002 and 0.046 Mev/ dis, respectively. The resultant ratio is 4.4E-02. The Xe-133 equivalent seen at the control room rad. monitors is :

from 98% EBFS: 4.4E-02 x 6.7E-02 = 3.0E-03 uCi/cc Xe-133 equivalent

from 2% EBFS: 4.4E-02 x 1.6E-02= 7.0E-04 uCi/cc Xe-133 equivalent

Based on Assumption 4.21, the detector response factor is 1 mr/hr = 0.009 uCi/cc DEQ Xe-133. This means that the detectors will have a response of 0. 08 mr/hr from the 2% bypass portion and 0.3 mr/hr from the 98% EBFS portion.

The monitor , which has an alarm setpoint of 2 mR/hr (Assumption 4.22), will not alarm because the resultant combined dose rate is 0.4 mR/hr which is less than the Tech Spec alarm setpoint (and current procedure alarm setpoint), therefore the control room will not isolate based on these ventilation inlet radiation monitors.

6.2.2 SFP Area Rad. Monitors

If it is assumed that the Kr-85 activity is instantaneously released from the spent fuel pool and homogeneously dispersed in the Spent Fuel Building air volume (after which it is fully released over a 2 hour period to the EBFS and smoke hole). The resultant specific activity in air is 1.939E+06 Ci/ 443,434 ft^3, which is equivalent to 4.4 Ci/ft^3. I-129 is not included because it is not a significant contributor.

The possibility of control isolation by the spent fuel pool area rad. monitors (RM 8139/ 8142/ 8156/ 8157) is evaluated.

A modified semi-infinite cloud dose model will be used to determine the dose rate. The modification to the semi-infinite cloud model basically halves the semi-infinite cloud model because the radiation monitors are located near the floor and wall. If they were not located near the wall, then a semi-infinite model would be appropriate.

The semi-infinite cloud dose rate model for gamma radiation , as described in Reference 22 (Reg. Guide 1.4) is D = 0.25 * E * X

where

D is dose rate in rad/sec

E is average gamma energy per disintegration

X is specific activity of cloud in Ci/m^3

Because the rad. monitors use a conversion factor based on Xe-133, a Kr-85 equivalent must be determined for use as a source term. The specific activity of the cloud is 4.4 Ci/ ft^3, which is equivalent to 155 Ci/ m^3. Based upon Assumption 4.20, the average gamma energy per disintegration for Xe-133 is 0.046 and 0.002 for Kr-85. To convert the Kr-85 activity into a Xe-133 equivalent, it is multiplied by the ratio 0.002/ 0.046 for a Xe-133 equivalent of 6.8 Ci/ m^3.

Applying this activity to the semi-infinite cloud model results in a dose rate of 7.8E-02 rad/sec or 282 rad/hr. Compensating for the finite volume of the building requires that the finite volume correction factor used by CRADLE, be applied here. The semi-infinite cloud dose rate is divided by the correction factor (of 1091/ V $^{0.3386}$) from Reference 3.26, where V is in cubic feet. With the SF Building volume of 443, 434 ft^3, the correction factor is 13.4. The resulting dose rate at the monitors is 282 rad/hr divided by 13.4 resulting in a dose rate of 21.1 R/hr. . Procedurally these rad. monitors are set to alarm and isolate at 50 mr/hr, although Tech. Spec. 3/4.3.3 has a rad monitor setpoint of 100 mr/hr. Since the monitors have an alarm setpoint of 50 - 100 mR/hr, it is highly likely that all 4 area radiation monitors in the spent fuel pool area will alarm because of the 21.1 R/hr dose rate. This will result in an AEAS alarm which will isolate the Control Room.

The smoke hole in the elevator shaft exhausts out the Main Exhaust System which vents out of the Unit 2 stack. The transit time from the Unit 2 stack to the Control Room Ventilation Intake is based on Reference 3.36, which identifies a distance of 41.5 m from the Unit 2 stack to the Control Room Intake. At the highest wind speed listed (2.5 m/sec), it would take approximately 17 seconds (41.5 m/ 2.5 m/sec) to reach the intake. Based upon Reference 3.31, the control room isolation damper closing time on AEAS signal is 5 seconds. Detector response time to the 21.1 R/hr dose rate would be small and the delay time for control room isolation is 5 seconds. The delay time for control room isolation is based upon Reference 3.31 and includes the delay from detector response and damper closing time (5 seconds). Since detector response time is negligible, delay time for control room isolation is 5 seconds. Detector response time is negligible due to the large difference between detector alarm (50 to 100 mR/hr) and the general area dose rate (21.1 R/hr). Assume for conservatism that detector response time is 5 seconds, therefore total response time for control room isolation is 10 seconds.

With the release taking 17 seconds to travel to the control room, the control room would be isolated prior to seeing the release, therefore, after isolation, the control room would only see unfiltered inleakage at 130 CFM.

6.3 TACT INPUTS

The inputs used for the TACT runs are listed below. It should be noted the 2% bypass (with 98% EBFS pass through) represents a bounding situation. The 2% bypass represents the elevator shaft smoke hole in the spent fuel building is a Unit 2 Stack release path.

In the TACT runs described below, it is noted that an instantaneous puff is modeled by using a 1 cubic foot volume and a 10,000 CFM release rate as evidenced in cards 3 and 10 of the TACT runs.

6.3.1 Inputs for EBFS Operation with 2% Bypass

98% EBFS Portion via Unit 1 Stack (charcoal filtration)

```
MP2 SPENT FUEL POOL CASK DROP ANALYSIS - INSTANTANEOUS PUFF - 98% EBFS
1 1 1 11 6 11 99 0
2754 0 1.0 1.0 0.0 0.75 0.25 0.00
1 2 0 0.0 2.0
2 1 1 3.5E-2
2 1 11 2.3E6
3 1 0 1
10 1 1 10000
12 1 1  90
13 1 1  70
17 6 0 1.00E-4 2.69E-5 3.47E-4 0.0 0.0 0.0
0/
```

2% EBFS Bypass Portion via Unit 2 Stack Release

```
MP2 SPENT FUEL POOL CASK DROP ANALYSIS - INSTANTANEOUS PUFF- 2% Bypass
1 1 1 11 6 11 99 0
2754 0 1.0 1.0 0.0 0.75 0.25 0.00
1 2 0 0.0 2.0
2 1 1 7.2E-4
2 1 11 4.6E4
3 1 0 1
10 1 1 10000
12 1 1 0.0
13 1 1 0.0
17 6 0 3.66E-4 4.80E-5 3.47E-4 0.0 0.0 0.0
0/
```

6.4 CRADLE INPUTS

6.4.1 I-129 Dose Calculation Because CRADLE doesn't do I-129

CRADLE does not perform dose calculations with I-129. Dose from I-129 will be determined below.

Reference 3.9 (R.G. 1.25) does not address I-129 for dose purposes and does not allow credit for decay in transit (for which decay would be negligible due to the long half-life of I-129, 1.4E+11 hours, Section 6.1). The long half-life and low abundance of this isotope make the dose consequences from I-129 not significant, as determined below.

The overall release fraction for I-129 is 9.8E-04 + 2.0E-05 for a total of 1.0E-03 (Section 6.1). This means that the total amount of I-129 released from the spent fuel pool is 1.0E-03 * 31.2 Ci I-129 = 3.2E-02 Ci, I-129.

Assuming no filtration (from EBFS or control room systems) of the effluent, of this amount, and based on the breathing rate of 3.47E-04 m^3/sec (Ass. 4.11), a Unit 2 Stack release Chi/Q of 2.92E-03 sec/ m^3 (Ass. 4.17) and a thyroid DCF of 5.92+06 rem/ Ci inhaled (Ass. 4.7), a dose would be determined as follows.

The 3.2E-02 Ci is released over 2 hours and is equivalent to 4.4E-06 Ci/sec release rate. This is multiplied by the Chi/Q to get the specific activity of the air at the control room intake. For conservatism and simplicity, it will be assumed that the control room air activity concentration is the same as the intake concentration during the 2 hour release period. This is conservative because no credit is taken for mixing with control room air. At T = 2 hours, it will be assumed that the activity remaining in the control room is at the same concentration as the intake but now it is being exhausted out at 130 cfm.

4.4E-06 Ci/ sec * 2.92E-03 sec/ m^3 = 1.3E-08 Ci/ m^3
(release rate) * (chi/q) = (intake concentration)

The specific activity is multiplied by the breathing rate to get the I-129 inhalation rate.

1.3E-08 Ci/ m^3 * 3.47E-04 m^3/ sec = 4.5E-12 Ci/ sec.
(intake concentration) * (breathing rate) = (inhalation rate)

To determine a dose rate using the thyroid DCF

4.5E-12 Ci/sec * 5.92E+06 rem/ Ci inhaled = 2.66E-05 rem/sec (9.59E-02 rem/hour or 1.60E-03 rem/ min).

The dose for the first two hours, post-accident is based on a 2 hour release time multiplied by the dose rate of 9.59E-02 rem/hour and is equivalent to 1.92E-01 rem.

The residual I-129 in the control room air after the 2 hour release must be accounted for. To account for this activity and related dose from the control room air activity, the following method is used. The control room air intake of 130 CFM is divided by the control room volume (35650 ft^3) to determine the air turnover rate. The resultant air turnover rate which is exponential by nature is 3.65E-03 min^{-1} . The integration of the dose rate (which is being reduced by control room ventilation) from 2 hours post-accident to 30 days (43200 minutes) using the 3.65E-03 min^{-1} reduction factor and 1.60E-03 rem/min result in the integral listed below.

$$f(x) = \int_{120}^{43200} (1.60E(-03)) * e^{-3.65E-03*x} dx = 0.283 rem$$

Solving the integral results in a post-release thyroid dose attributable to the iodine in the control room of 0.283 rem.

<u>The total control room thyroid dose is 0.192 rem + 0.283 rem which is equivalent to 0.475 rem to the thyroid.</u>

In the event that the control room does not isolate, the inlet flow rate of 800 cfm would be appropriate to use. In this case, the air change rate is 800 cfm/ 35650 ft^3 which is equivalent to 0.0224 min^{-1}. Applying this to the integral above results in a total dose from the buildup of iodine in the control room of 0.005 rem. When compared to the dose resulting from control room isolation at 130 cfm of 0.283 rem, the use of 130 cfm is conservative.

It should be re-iterated that the method used above is extremely conservative for the following reasons.

- for the first 2 hours it is assumed that control room iodine concentration is the same as at the control room air intake. No credit is taken for mixing
- after the release stops, the control room iodine concentration is assumed to be the same as that at the air intake during the release but there is no more iodine input to the control room.
- no credit is taken for filtration

6.4.2 CRADLE Inputs for EBFS Operation with 2% Bypass (Blowing it out the Smoke Hole)

The CRADLE output will provide the dose consequences attributable to Kr-85 and many other isotopes, but only the Kr-85 portions will be used. In addition the release fraction used in the following input sets results in a Kr-85 activity in Node 1 (from CRADLE) that is comparable to the activity released from the fuel for the 98% and 2% release paths identified in Section 6.1. This release fraction is used so that the activity in the node is consistent with what is presented in this calculation. The release fraction can be set to any number including 1 but then the output must be adjusted to reflect the activity in the node. Also, the containment leak rate input was set at 7.0 hr^{-1} so that essentially all activity is exhausted over a 2 hour period. This value was arbitrarily chosen but in actuality results in an exponential removal process over 2 hours of $e^{-7.0*2}$ which means that only 8E-07 of the original activity remains in the spent fuel building. This input represents the exhaust rate from the spent fuel building and credits control room isolation as a result of SFP area radiation monitor alarms.

2% EBFS Bypass Portion

```
MP2 SFP CASK DROP ANALYSIS - KR-85 ONLY, 2% BYPASS                    1
   2.754E3  3.565E4        1.         0         5
   0.000E0  2.000E0  8.000E0  2.400E1  9.600E1  7.200E2
       0.0  0.03452
       0.0      0.0
       0.0      0.0
       0.0      0.0
       0.0      0.0
     1.3E2    1.3E2    1.3E2    1.3E2    1.3E2
     0.000    0.000    0.000    0.000    0.000
     1.3E2    1.3E2    1.3E2    1.3E2    1.3E2
       0.0      0.0      0.0
       0.0      0.0      0.0
       0.0      0.0      0.0
       0.0      0.0      0.0
       0.0      0.0      0.0
       0.0      0.0      0.0
   2.92E-3      0.0      0.0      0.0      0.0
    7.0E00      0.0      0.0      0.0      0.0
       0.0      0.0      0.0
       0.0      0.0      0.0
       0.0      0.0      0.0
       0.0      0.0      0.0
       0.0      0.0      0.0
       0.0      0.0      0.0
```

98% EBFS Portion

```
MP2 SFP CASK DROP ANALYSIS - KR-85 ONLY, 98%EBFS, U1 STACK RLS        1
   2.754E3  3.565E4        1.         0         5
   0.000E0  2.000E0  8.000E0  2.400E1  9.600E1  7.200E2
       0.0  1.68231
       0.0      0.0
       0.0      0.0
       0.0      0.0
       0.0      0.0
     1.3E2    1.3E2    1.3E2    1.3E2    1.3E2
     0.000    0.000    0.000    0.000    0.000
     1.3E2    1.3E2    1.3E2    1.3E2    1.3E2
       0.0      0.0      0.0
       0.0      0.0      0.0
       0.0      0.0      0.0
       0.0      0.0      0.0
```

0.0	0.0	0.0		
0.0	0.0	0.0		
2.51E-4	0.0	0.0	0.0	0.0
7.0E0	0.0	0.0	0.0	0.0
0.0	0.0	0.0		
0.0	0.0	0.0		
0.0	0.0	0.0		
0.0	0.0	0.0		
0.0	0.0	0.0		
0.0	0.0	0.0		

It should be noted that in the CRADLE output, only the dose from Kr-85 is used since it is the only nuclide present.

The control room dose will also be evaluated assuming no control room isolation. The only difference in CRADLE inputs is that the 130 cfm inleakage rate is replaced with the 800 cfm normal intake (Assumption 3.18).

6.5 RESULTS OF TACT AND CRADLE COMPUTER RUNS

The following outputs from TACT and CRADLE runs are provided. CRADLE results are listed for the two cases where the control room is 1) isolated and 2) not isolated. Doses corrected to ICRP 30 DCFs are listed in parentheses. The correction method is described in the text following Table 6.5.1.

Table 6.5.1 - Summary of Output

Code	Run #	Run Date	Description	thyroid, rem	whole body, rem	beta skin, rem
TACT	14849	7/12/99	98% EBFS - EAB	1.01E-03 (1.08E-03)	1.17E-01	NA
TACT	14849	7/12/99	98% EBFS - LPZ	2.72E-04 (2.91E-04)	3.16E-02	NA
TACT	14814	7/12/99	2% Bypass - EAB	5.07E-04 (5.42E-04)	8.59E-03	NA
TACT	14814	7/12/99	2% Bypass - LPZ	6.65E-05 (7.10E-05)	1.13E-03	NA
CRADLE	15631	7/12/99	Kr-85 only, 98% EBFS (semi-infinite	NA	2.62E-01*	2.03E+01

			cloud) w/ Control Room isolated			
CRADLE	14997	7/16/99	Kr-85 only, 2% bypass (semi-infinite cloud) w/ Control Room isolated	NA	6.26E-02*	4.84E+00
CRADLE	13287	9/2/99	Kr-85 only, 98% EBFS (semi-infinite cloud) w/ Control Room unisolated	NA	2.62E-01*	2.03E+01
CRADLE	13615	9/2/99	Kr-85 only, 2% bypass (semi-infinite cloud) w/ Control Room unisolated	NA	6.26E-02*	4.84E+00

* semi-infinite cloud dose - not adjusted for volume reduction due to limited size of control room

TACT and ICRP 30 use the following I-129 Dose Conversion Factors (DCFs) for the thyroid

	TACT DCF, Thyroid	ICRP 30 DCF, Thyroid
I-129	5.542E+06	5.92E+06 (Assumption 4.7)

The TACT thyroid results will be adjusted using the ICRP 30 (Ref. 3.19) thyroid dose conversion factors. The adjustment uses the equation below

$$D_{Thy}^{ICRP30} = D_{Thy}^{TACT} * R$$

Where

D_{Thy}^{ICRP30} = thyroid dose adjusted for ICRP 30, I-129 value

D_{Thy}^{TACT} = thyroid dose from TACT, based on I-129

R = ratio of ICRP 30 DCF to TACT DCF

= 5.92E+06 / 5.542E+06

= 1.068

The TACT thyroid doses are multiplied by 1.068 to determine an ICRP 30 based equivalent.

6.6 RESULTS

All doses listed below have been summarized in Table 6.5.1 or Section 6.4.1 and modified as described below. They will be based upon 2% bypass, which provides a greater dose than 100% EBFS.

Whole body doses are:
EAB = 1.17E-01 rem (98%) + 8.59E-03 rem (2%) = 1.26E-01 rem, whole body
LPZ = 3.16E-02 rem (98%) + 1.13E-03 rem (2%) = 3.27E-02 rem, whole body
Control Room = 2.62E-01 rem (98%) + 6.26E-02 rem (2%) =3.25E-01 rem, whole body

Beta (Skin) dose from Table 6.5.1 is:
Control Room = 2.51E+01 rem, skin.

Thyroid doses from Table 6.5.1 and section 6.4.1 are listed below.
EAB = 1.08E-03 rem (98%) + 5.42E-04 rem (2%) =1.62E-03 rem, thyroid
LPZ = 2.91E-04 rem (98%) + 7.10E-05 rem (2%) =3.62E-04 rem, thyroid
Control Room = 4.75E-01 rem, thyroid (no correction is required here, see Section 6.4.1)

The unisolated control room doses do not appear to differ from the isolated control room dose as shown in Table 6.5.1. It should be noted that there are differences but only when looking at the results to 3 significant figures.

The dose consequences from the drop of a consolidated fuel canister (CFC) onto a stored fuel assembly was not evaluated in this calculation, but the consequences are bounded by the results of the cask drop analysis. Reference 3.1 states that the maximum allowed consolidation is 1376 assemblies into 688 storage locations. This implies 2 fuel bundles per CFC. Since the cask drop analysis includes more than 1 fuel assembly and 1 CFC, then this analysis bounds the CFC drop event.

7. DESIGN VERIFICATION

A design review was performed in accordance with Design Control Manual Chapter 4 "Design Inputs and Design Verification". All design inputs were verified and assumptions were validated.

8. ATTACHMENTS

Attachment A - Reviewer Comments

Calculation Review Comment and Resolution Form

Attachment A

(Sheet 1 of 1)

Calculation Number: CASKDOSE-02686-R2 Revision: 0 CCN NA

Calculation Title: MP2 Spent Fuel Cask Drop- Dose Analysis

Calc. Originator: S. M. Torf Reviewer (PRINT): J. L. Wheeler

This form is intended to document significant comments and their resolutions. Typographical errors and other editorial recommendations may be marked up in the calculation text and presented to the originator

Review Type ☞ ☐**Interdiscipline** ☒**Independent**

Reviewer (SIGN) Date: 9/9/99

(signature signifies all comments have been resolved to your satisfaction)

Item	Page/Section	Comments	Response
1	general	the scope of the calculation should use consistent language throughout the calculation regarding cask movement and condition of the smoke hole	done
2	general	the use of 21 feet from the edge of the cask laydown area is very conservative but inconsistent with Tech specs. Should use 21 feet from center of cask laydown area.	done
3	4.17	a ground release pathway for smoke hole releases is possible and should be evaluated instead of the U2 stack release pathway	done
4	4.20	Xe-133 average gamma energy is 0.046 mev/dis, not 0.03	done
5	4.22	procedure uses 1 mr/hr alarm setpoint for CR intake rad. monitors, TS uses 2 mr/hr	the evaluation considered both setpoints and associated consequences
6	6.2.2	what is the basis for the semi-infinite cloud correction factor	basis was added to calculation
7	4.17	Use of U2 stack x/q may be more appropriate than ground x/q	Agreed. Ground x/q results in extreme conservative