Entergy

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0CAN080701

August 2, 2007

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Supplement to Amendment Request
To Delete the Fuel Handling Area Ventilation System and
Associated Filter Testing Program Requirements
Arkansas Nuclear One, Unit 1 and Unit 2
Docket Nos. 50-313 and 50-368
License Nos. DPR-51 and NPF-6

REFERENCES: 1. Entergy letter dated April 24, 2007, "License Amendment Request to Delete the Fuel Handling Area Ventilation System and Associated Filter Testing Program Requirements" (0CAN040701) (TAC NOs: MD5379 and MD 5380)

Dear Sir or Madam:

By letter (Reference 1), Entergy Operations, Inc. (Entergy) proposed a change to the Arkansas Nuclear One, Unit 1 (ANO-1) and Unit 2 (ANO-2) Technical Specifications (TSs) to delete Fuel Handling Area Ventilation System (FHAVS) and associated Ventilation Filter Testing Program (VFTP) requirements from the TSs.

On July 23, 2007, Entergy received requests for additional information (RAIs). A conference call was held with associated NRC staff members on July 24 and July 25, 2007, to ensure clear understanding of the information being requested. Attachment 1 includes Entergy's response to these RAIs. Attachments 2 and 3 are the current Fuel Handling Accident analyses for ANO-1 and ANO-2, respectively.

Attachment 4 includes an additional mark up page of TS 3.8.10, Distribution Systems – Shutdown, which deletes reference to TS 3.7.12. The removal of this reference is appropriate since the proposed amendment (Reference 1) deletes TS 3.7.12.

There are no technical changes proposed that impact the original no significant hazards consideration included in Reference 1. There are no new commitments contained in this letter.

A001

NRR

If you have any questions or require additional information, please contact David Bice at 479-858-5338.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 2, 2007.

Sincerely,

Timothy G. Mitchell

TGM/dbb

Attachments:
1. Response to Request for Additional Information
2. ANO-1 Fuel Handling Accident Analysis
3. ANO-2 Fuel Handling Accident Analysis
4. Markup of Additional Affected Technical Specification Page

cc: Dr. Bruce S. Mallett
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Attachment 1

To

0CAN080701

Response to Request for Additional Information

Response to Request for Additional Information Related to License Amendment Request to Delete the Fuel Handling Area Ventilation System and Associated Filter Testing Program Requirements

Question 1:

Please explain if any non-safety systems were credited in the analysis of the postulated design basis fuel handling accident (FHA) in the spent fuel pool (SFP) area, which concluded that the Fuel Handling Ventilation System (FHAVS) is not needed to perform a safety function required to mitigate the consequences of an accident, and if so please justify crediting such systems in this design basis accident (DBA) analysis.

Response 1:

Non-safety systems were not credited in the FHA analyses.

Question 2:

A) *Please provide the dose consequence analyses or all equations, assumptions, and input parameters used to determine the offsite and control room doses at ANO-1 and ANO-2 following the postulated design basis FHA, which supports the assertion that the FHAVS does not perform a safety function.*

B) *Please describe the bases for all equations, assumptions, and input parameters used in the FHA dose consequence analysis for ANO-1 and ANO-2.*

Response 2:

No new analyses were performed in support of the proposed amendment. The analyses of record were used to support a previous amendment permitting containment building openings during the movement of irradiated fuel. These previous submittals provided assumptions based on regulatory guidance and the dose consequences of a fuel handling accident inside containment with the containment open. The NRC performed their own calculations during their review of these submittals and verified Entergy's conclusions were acceptable. The NRC subsequently approved the requests in Arkansas Nuclear One, Unit 1 (ANO-1) Technical Specification (TS) Amendment 184 (letter dated September 20, 1996) and ANO, Unit 2 (ANO-2) TS Amendment 166 (letter dated September 28, 1995). Additional containment openings were later approved by the NRC based on the aforementioned Amendment 184 and 166 Safety Evaluations (example: ANO-2 TS Amendment 203, dated April 16, 1999). Therefore, Entergy believes the previously approved analyses are acceptable for use as a basis for the current proposed amendment. The response to Question 4 below further discusses the bounding nature of the analyses of record with respect to the difference in location of the fuel handling accident (i.e., the spent fuel handling area as opposed to the containment building).

Notwithstanding the above, the aforementioned amendments involved several submittals and conversations. Therefore, in order to simplify the NRC review of the proposed amendment, the current ANO-1 and ANO-2 analysis are included in Attachments 2 and 3 of this response, respectively. This information includes the calculations, assumptions, and input parameters associated with the FHA analyses.

Question 3:

Please provide the dose acceptance criterion, and the justification for this criterion, that was used for the post DBA dose consequence to the ANO-1 & ANO-2 control rooms following the postulated FHA.

Response 3:

The acceptance criteria for the post Design Basis Accident (DBA) dose consequences to the ANO-1 and ANO-2 control rooms following the postulated FHA are contained in General Design Criteria (GDC) 19 of 10 CFR 50, Appendix A. The equivalent thyroid dose is taken to be 30 rem, consistent with the current ANO-1 and ANO-2 licensing bases.

Question 4:

Please verify that the atmospheric dispersion associated with the containment release point, as analyzed in the current design basis FHA, bounds the atmospheric dispersion expected from the SFP area and Fuel Handling Building.

Response 4:

The ANO-1 spent fuel pool and fuel handling area are ventilated by fans that are mounted on the roof of the ANO-1 auxiliary building. These fans exhaust through a duct that is mounted on the outside of the ANO-1 reactor building (containment) wall. The effluents are then directed to a release point at elevation 533' 6" at the top of the ANO-1 reactor building. Similarly, the ventilation air from the ANO-2 fuel handling floor radwaste area is exhausted out a containment flute on the outer wall of the ANO-2 containment at elevation 533' 3". Therefore, since both units direct and exhaust any releases at the top of their respective containment structures, it is concluded that the atmospheric dispersion of a release due to a fuel handing accident would be the same whether the release is from the containment or from the fuel handling area. In summary, the current design basis FHA analyses for the containment building releases is bounding for the postulated respective fuel handling area releases.

Additional Changes

Attachment 4 includes an additional TS mark-up page of TS 3.8.10, Distribution Systems – Shutdown, which deletes reference to TS 3.7.12. The removal of this reference is appropriate since the proposed amendment (Reference 1) deletes TS 3.7.12. There are no technical changes proposed that impact the original no significant hazards consideration included in the original submittal.

Attachment 2

To

0CAN080701

ANO-1 Fuel Handling Accident Analysis

ARKANSAS NUCLEAR ONE
CALCULATION COVER SHEET

Calc. No.: 95-E-0030-01 **Rev. No.:** 0(2)

Calc. Title: Fuel Handling Accident Dose Analysis

Unit: 1 **Category:** Q
System (s):
Calc. Type: NS

Component No (s): NA

Topic (s): FHAC
Plt Area: **Bldg.** ______ **Elev.** ______
Room ______ **Wall** ______
Coordinates: ______

Config. Checklist (per 5010.004) completed? (Y or N) Y

Abstract (Included Purpose/Results): An evaluation of the offsite and Control Room dose consequences of a fuel handling accident occurring in either the containment or the spent fuel handling area. / The resulting site boundary doses for a FHA in containment (with no containment closure) is 63.599 Rem to the thyroid and 0.27 Rem to the Whole Body and 0.902 Rem to the skin. The Control Room results for this case is 3.283 Rem to the thyroid, 0.026 Rem to the Whole Body, and 2.045 Rem to the skin. The resulting site boundary doses for a FHA in the Spent Fuel Handling area (crediting ESF filtration) is 9.537 Rem to the thyroid, 0.261 Rem to the Whole Body, and 0.888 Rem to the skin. The Control Room results for this case is 0.493 Rem to the thyroid, 0.026 Rem to the Whole Body, and 2.045 Rem to the skin. These doses are reported in Attachment 6 of the calculation.

Pages Revised and/or Added: Attachment 6, Pages 1-2

Purpose of Revision: Calculation 95-E-0030-02, Rev. 0, "ANO-1 Fuel Assembly Drop Damage", has increased the number of fuel pins assumed to be damaged in a fuel handling accident to 82 (6 rows of pins) from 56 pins. This revision adds Attachment 6 to assess the doses resulting from this increased source term.

Initiating Documents	Resulting Document(s)	Key Design Input Docs.
		Calculation CEO-95-00067 Calculation 95-E-0030-02

Verification Method: **Design Review** X **Alternate Calculation** ______ **Qualification Testing** ______

Amends Calc(s): ______

Supersedes Calc(s): M-4135-6, Rev. 0, FSAR Accident Analysis Fuel Handling

Computer Software Used: TRANSACT Computer Code, CDP 93-C-0003-01, Math cad 3.1 for Windows

	(Print Name)	(Initials)	(Date)		(Print Name)	(Initials)	(Date)
By:	John W. Cotton	JWC	5/30/96	**Rvw'd:**	—		
Chk'd:	Daniel W. Fouts	DWF	5/31/96	**Apv'd:**	Karen M. Head	KMCKM	5/31/96

Check if Additional Revisions: ______

Table of Contents

1 PURPOSE

The purpose of this calculation is to evaluate the offsite (Exclusion Area Boundary, EAB) and control room dose consequences of a fuel handling accident (FHA). This accident is postulated to occur either in the containment or in the spent fuel pool area.

The following cases are analyzed:

Case 1: FHA in Containment (or in Fuel Handling Building without ESF Filtration) – 100 hours decay, 82 rods failed.

Case 2: FHA in Fuel Handling Building With ESF Filtraion (or in Containment with Purge Filtration) (100 hours decay, 82 rods failed)

Case 3: FHA in Containment (or in Fuel Handling Building without ESF Filtration) – 100 hours decay, 1 rod failed.

Case 4: FHA in Fuel Handling Building With ESF Filtraion (or in Containment with Purge Filtration) (100 hours decay, 1 rod failed)

Case 5: FHA in Containment (or in Fuel Handling Building without ESF Filtration) – 100 hours decay, 82 rods failed, maximizing CR inleakage.

2 Cycle dependent Parameters

Some of the parameters utilized in this calculation are based on the current fuel type and cycle exposure. Consequently, this calculation should be reviewed for adequacy each cycle. The following table lists the calculation parameters that are cycle-dependent along with the bounding values used in the analysis.

Calculation Parameter	Rationale
Rod internal pressure <1200 psig	Required by Regulatory Guide 1.25 (Position 1.b)
Core radial power peaking factor ≥1.65	Required by Regulatory Guide 1.25 (Position 1.e)
Peak linear power density ≤20.5 kW/ft	Required by Regulatory Guide 1.25 (Position 1)
Maximum fuel centerline temperature<4500°F	Required by Regulatory Guide 1.25 (Position 1)
Fuel batch average burnup for peak assembly ≤60,000 MWD/ton. (peak local burnup of 60,000 MWD/ton). [enrichment 5w/0, 465 Kg bundle]	Used in development of radionuclide source terms with the ORIGEN-II code. [Note: Fuel batch average burnup for peak assembly ≤25,000 MWD/ton is required by Regulatory Guide 1.25 (Position 1)]
Maximum number of fuel rods damaged during a fuel handling accident = 82	Reference 28, Calculation 95-E-0030-02, Rev. 0
Reactor and spent fuel pool water level sufficient to ensure 23 feet water coverage above any damaged fuel	Used in the basis for a decontamination factor of 100 for iodine per Regulatory Guide 1.25 (Position 1.c)
χ/Q at exclusion area boundary (0-2 hrs) ≤ 6.5×10^{-4} sec/m^3	Used in calculating the dose consequences of fuel failures
Spent Fuel Building and containment exhaust filter removal efficiency for iodine = 90% for inorganic and 70% for organic	Removal efficiency for iodine per Regulatory Guide 1.25 (Position 1.j). Based on 2-inch charcoal bed depth.

3 RESULTS

Standard Review Plan (SRP) 15.7.4 requires that the dose consequences of a FHA must be well within the limits given in 10CFR100. "Well within" is further defined in this SRP as 25% of the limits of 10CFR100. This results in an offsite dose limit of 6 rem to the whole body and 75 rem to the thyroid for the duration of the accident.

Table 1.
CASE 1 and Case 5- ANO UNIT 1 FHA
FHA in Containment (or in Fuel Handling Building Without ESF Filtration)
100 hours delay, 82 rods failed (Attachments 3 and 7)

Dose Category	EAB (2hr) Rem	Control Room Rem (10 cfm inleakage)	Control Room Rem(98 cfm inleakage)
Whole Body	2.948E-01	4.644E-02	5.279E-02
Skin[1]	1.166	3.751	4.257
Thyroid	6.912E+01	4.097	2.981E+01

Table 2.
CASE 2 - ANO UNIT 1 FHA
FHA in Fuel Handling Building with ESF Filtration (or in Containment with Purge Filtration
100 hours decay, 82 rods failed (Attachment 4)

Dose Category	EAB (2 hr) Rem	Control Room Rem
Whole Body	2.850E-01	4.642E-02
Skin[1]	1.146	3.751
Thyroid	1.037E+01	6.146E-01

[1] The "total" skin dose is reported, that is, the skin dose resulting from gamma radiation plus that resulting from beta radiation.

Table 3.
CASE 3 - ANO UNIT 1 FHA
FHA in Containment (or in Fuel Handling Building Without ESF Filtration)
100 hours delay, 1 rod failed (Attachment 5)

Dose Category	EAB (2hr) Rem	Control Room Rem
Whole Body	3.595E-03	5.662E-04
Skin[1]	1.422E-02	4.574E-02
Thyroid	8.427E-01	4.995E-02

Table 4.
CASE 4 - ANO UNIT 1 FHA
FHA in Fuel Handling Building with ESF Filtration (or in Containment with Purge Filtration
100 hours decay, 1 rod failed (Attachment 6)

Dose Category	EAB (2 hr.) Rem	Control Room Rem
Whole Body	3.475E-03	5.660E-04
Skin[1]	1.398E-02	4.573E-02
Thyroid	1.264E-01	7.493E-03

4 REFERENCES

1. Regulatory Guide 1.25, (Safety Guide 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, March 23, 1972.
2. NUREG-0800 (Standard Review Plan), Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents", Revision 1, July 1981.
3. 10CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance", last amendment date - June 24, 1975.
4. Annals of the ICRP, ICRP 30 supplement to Part 1, July 1978.
5. NUREG-800, Standard Review Plan 2.3.4, "Short Term Dispersion Estimates for Accidental Atmospheric Releases", revision 1, July 1981.
6. NUREG/CR-5106, User's Guide for the TACT5 Computer Code, Appendix e, June 1988.
7. Operating License No. DPR-51, ANO Unit 1
8. NUREG/CR-5009, Assessment of the Use of Extended Burnup Fuel in Light Water Reactors, February 1988.
9. Chart of the Nuclides, Knolles Atomic Power Laboratory, Tenth Edition, Norman E. Holden and F. William Walker
10. Table of Isotopes, C. M. Lederer, J. M. Hollander, and I. Perlman, 6th Edition, John Wiley and Sons, New York, 1967
11. Regulatory Guide 1.52, "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants", Rev. 2, 1978
12. ANO Unit-1, Technical Specifications
13. ANO Unit 1 FSAR Table 14-24
14. Standard Review Plan 6.4, "Control Room Habitability System", Rev. 2, July 1981
15. ANO Unit 1 SAR Section 14.2.2.3, Amendment No. 17
16. Murphy and Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19", August 1974
17. TRANSACT Computer Code, SP 93-C-0003-01, Rev. 2
18. Calc CEO95/00067, ORIGEN "Airlock" Isotopic Activities
19. ANO Unit 1 Technical Specification 2.1.2
20. ANO Unit 2 FSAR, Amendment 12, Table 1.3-1
21. ANO Unit 1 FSAR, Amendment 12, Table 1-1
22. Interoffice Memorandum, CEO95/00095, dated 3/21/95, from D.L. Smith to J.W. Cotton, "Maximum Cold Internal Pressure of a PWR Fuel Rod"
23. Calc 91-E-0117-01, Control Room Habitability Considering Containment Leakage and ESF Leakage, Rev. 4.
24. Calculation 00-R-1001-01, Cycle 17 Ground Rules

25. Calculation 00-R-1001-02, Cycle 17 Reload Technical Document
26. Entergy Operations Arkansas Nuclear One – Unit One Cycle 17 Core Operating Limits Report
27. Calculation 88-E-0130-01, CR LOCA Dose Calc Using Revised Source Terms, Rev. 2
28. Calculation 95-E-0030-02, ANO-1 Fuel Assembly Drop Damage, Rev. 0

5 CURRENT LICENSING BASIS

The current licensing basis for a FHA is given in Reference 15, ANO Unit 1 SAR Section 14.2.2.3, Amendment 17. SAR Section 14.2.2.3 provides a discussion of the FHA, methods of analysis and analytical results. The Original licensing basis for ANO-1 was compliance with the limits of 10CFR100. There was no original commitment to Regulatory Guide 1.25 or Standard Review Plan 15.7.4. The current FHA analyses considered the failure of six rows of fuel rods (82 rods out of 208 in the entire assembly). The SER evaluated the failure of an entire assembly (208) rods with the resulting doses "well within" the 10CFR100 limits. The original analysis also assumed a water depth of 9.5 feet resulting in the removal of 99% of the released iodines.

6 ASSUMPTIONS AND DATA

6.1 General Regulatory Requirements

Reference 2 gives the acceptance criteria related to the radiological consequences of a fuel handling accident:

1. The calculated whole-body and thyroid doses at the exclusion area and low population zone boundaries are to be "well within" the guidelines of 10CFR100.11 (Reference 3: 300 Rem to the thyroid, 25 Rem to the whole body). Reference 2 states that "well within" means ≤25% of the 10CFR100.11 limits. This is given as 75 Rem to the thyroid and 6 Rem[2] to the whole-body.

2. The calculated dose shall incorporate the appropriate conservative assumptions stated in Regulatory Guide 1.25 (Reference 1) with the exception of the atmospheric dispersion factors (i.e., χ/Q values) which should be determined in accordance with Standard Review Plan 2.3.4 (Reference 5).

A discussion of the Regulatory Guide 1.25 assumptions is provided in Sections 6.2 through 6.5. A discussion of the application of Standard Review Plan 2.3.4 methodology is given is Section 6.6.

The acceptance criteria for a fuel handling accident with regard to control room doses is given in 10CFR50, Appendix A, General Design Criterion (GDC) 19. The regulatory guidance given in GDC 19 states that adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures to any part of

[2]Actually, 25% of 25 Rem is 6.25 Rem. However, since Reference 2 defines 25% of 25 Rem as 6 Rem, this analysis will use 6 Rem as the limit.

the body in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Standard Review Plan 6.4 (Reference 14) further defines the control room dose limit as : 5 rem whole body and 30 rem thyroid or skin dose.

6.2 Regulatory Guide 1.25: Overview

The basic assumptions given in Regulatory Guide 1.25 are related to the following:

1. The manner of calculating the quantity of activity (both iodines and noble gases) released to the containment, spent fuel building, or the environment. Guidance is provided on calculating the following:

   - the fission product inventory within the fuel pellets
   - the fraction of the fuel pellet fission product inventory which is released from the fuel pellets to the pellet/cladding gap and consequently available for release to the pool water following cladding failure
   - the pool decontamination factors for the fission products
   - the iodine removal efficiencies for plant adsorbers/filters

   Additional details of these assumptions are given in Section 6.3.

2. The manner of calculating the thyroid dose based on the curies of iodine released into the environment. Details of these calculations are given in Section 6.4.

3. The manner of calculating the whole-body dose based on the curies of noble gas released into the environment. Details of these calculations are given in Section 6.5.

6.3 Regulatory Guide 1.25: Activity Releases

The major guidance and assumptions related to the quantity of activity released are as follows:

1. *Requirement:* The accident occurs at the earliest time that fuel handling can begin as allowed by the Technical Specifications. Radioactive decay of the fission product inventory during the interval between shutdown and the commencement of fuel handling is taken into consideration.

   Response: Based on the current revision of Technical Specification 3.8.11, the minimum subcritical time before fuel movement is 100 hours.

2. *Requirement:* The maximum fuel rod pressurization (after the above decay time) is 1200 psig.

Response: Reference 22 states that the maximum fuel rod pressure during refueling will be less than 1,200 psig if the hot internal pressure is less than the reactor operating pressure.

3. *Requirement:* The minimum water depth between the top of the damaged fuel rods and the fuel pool surface is 23 feet.

Response: This water level is required to allow the use of an overall decontamination factor for iodine of 100 (i.e., 1% of the iodine released to the pool will escape from the pool and 99% will be retained in the pool). The pool decontamination factor for noble gases is 1 (i.e., no noble gases retained in the pool water). The minimum depth of water above the potentially damaged fuel rods is as follows:

Containment	23 ft	
Fuel Pool	23 ft	TS 3.8.6 (ITS 3.9.6)

Consequently the use of a water coverage of 23 feet is applicable and the iodine decontamination factors given in Regulatory Guide 1.25 are conservative.

4. *Requirement:* All of the gap activity in the damaged rods is released and consists of 10% of the total noble gas other than Kr-85, 30% of the Kr-85, and 10% of the total radioactive iodine in the rods at the time of the accident. For the purpose of sizing filters for the fuel handling accident addressed in this guide, 30% of the I-127 and I-129 inventory is assumed to be released from the damaged fuel.

Response: NUREG/CR-5009 (Reference 8) discusses the variations in plenum inventories for extended burnup fuel. Table 3.6 of Reference 8 indicates that the I-131 inventory is 20% higher than in Regulatory Guide 1.25 for rod burnups of 60 GWD/MTU. Therefore the gap activity for iodine is assumed to be 12%.

Noble Gas Release (except Kr-85) 10%

Kr-85 Release	30%
Radioactive Iodine Release	12%

5. *Requirement:* The values used for individual fission product inventories are calculated assuming full power operation at the end of core life immediately proceeding shutdown and such calculation should include an appropriate radial peaking factor. The minimum acceptable radial peaking factor is 1.65 for PWR's.

   Response: This analysis uses the source terms developed by use of the ORIGEN-II code. Rated power at ANO Unit 1 is 2568 Mwt (Reference 7). The FHA source terms were generated assuming a power level of 2855 Mwt (103% of the proposed upgraded power level of 2772 Mwt) and a radial peaking factor of 1.65. The power level assumed is conservative with regard to 10CFR50.46, Appendix K and Standard Review Plan 15.6.5, which both assume 102% of rated power. The radial peaking factor is consistent with Regulatory Guide 1.25. However, in subsequent cycles (for cycle 17 see references 24 and 25), a maximum radial peaking factor of 1.8 has been used from the COLR (reference 26). Therefore, the source term will be multiplied by the ratio of 1.8/1.65 to obtain the source term at a radial peaking factor of 1.8.

6. *Requirement:* The fission products which are assumed to be released to the containment/auxiliary building atmosphere are assumed to escape into the environment within two hours.

   Response: All releases of fission products to the environment are assumed to occur over a two hour time period. Also, this calculation conservatively assumes that there is no decay of the fission products during residency in the containment or auxiliary building. Two-hour χ/Q factors are used to calculate the offsite doses.

7. *Requirement:* If it can be shown that the building atmosphere is exhausted through adsorbers designed to remove iodine, the removal efficiency is 90% for inorganic iodine and 70% for organic species. [This is based on a 2-inch charcoal bed depth with 1/4-second residence time.]

   Response: Iodine filter efficiencies of 90% for inorganic iodine and 70% for organic species are used in this calculation. This assumption is

consistent with ANO Technical Specifications (3.15.1 and 3.22.1) which require filter efficiencies of ≥99% when tested in-place with a halogenated hydrocarbon refrigerant test gas. The Technical Specifications also require testing prior to fuel handling and these tests are performed in accordance with the criterion of Regulatory Guide 1.52 (Reference 11) for activated charcoal beds of 2 inches or more.

This calculation will use the following iodine filter removal efficiencies:

Inorganic	90%
Organic	70%

8. *Requirement:* The iodine gap inventory is composed of inorganic species (99.75%) and organic species (0.25%).

   Response: The analysis will use these assumptions:

Inorganic Iodine	99.75%
Organic Iodine	0.25%

9. *Requirement:* The pool decontamination factors for the inorganic and organic species are 133 and 1, respectively, giving an overall effective decontamination factor of 100 (i.e., the pool water retains 99% of the total iodine released from the damaged rods). This difference in decontamination factors for inorganic and organic iodine species results in the iodine above the fuel pool being composed of 75% inorganic and 25% organic species.

   Response: The analysis will use these assumptions:

Inorganic Iodine DF	133
Organic Iodine DF	1

10. *Requirement:* The retention of noble gases in the pool is negligible (i.e., decontamination factor of 1).

    Response: The analysis will use this assumption. [Noble Gas Pool DF = 1]

11. *Requirement:* The effluent from the filter system passes directly to the emergency exhaust system without mixing (credit for mixing will be allowed in some cases: the amount of credit will be evaluated on an individual case basis) in the surrounding building atmosphere and is then released (as an

elevated plume for those facilities with stacks). Credit for an elevated release will be given only if the point of release is (a) more than two and one-half times the height of any structure close enough to affect the dispersion of the plume or (2) located far enough from any structure which could affect the dispersion of the plume.

Response: Credit for mixing in the containment and spent fuel building atmosphere is not taken. Releases are assumed to be ground level releases.

The guidance provided in Regulatory Guide 1.25 states that the above assumptions are only applicable if the following three conditions are met. These conditions are primarily related to the fraction of fission products that are released to the fuel/cladding gap. If any condition is not met, the impact of this on the above assumptions will need assessment. As shown below, all conditions are met for ANO-1.

1. *Requirement:* The peak linear heat generation rate (LHGR) is not to exceed 20.5 kW/ft.

   Response: The ANO Unit 1 Technical Specifications 2.1.2 (Reference 19) limits the LHGR to 20.5 kW/ft. This value represents the maximum overpower (accident) thermal output. During normal operations, the maximum thermal output, which is appropriate for a fuel handling accident, is 17.63 kW/ft (Reference 20). Even this value would only be reached for low burnup fuel. Therefore, fuel centerline melting is not a concern and the assumptions of Regulatory Guide 1.25 are applicable.

2. *Requirement:* The maximum fuel centerline temperature is to be less that 4500°F.

   Response: The peak centerline temperature is 4220°F (Reference 21) for a 100% power hot spot. This is acceptable relative to the regulatory guidance.

3. *Requirement:* The average burnup of the peak bundle is less than 25,000 MWd/MTU.

   Response: The average burnup of the peak bundle is assumed to be 60 MWd/MTU. For this burnup level, there may be an increase in fuel rod gap inventory.

   NUREG/CR-5009 (Reference 8) discusses the variations in plenum inventories for extended burnup fuel. Table 3.6 of Reference 8 indicates that the I-131 inventory is 20% higher than in Regulatory Guide 1.25 for rod burnups of 60 GWd/t. For rod burnups of 33

GWd/t, the NUREG predicts an I-131 release fraction of 0.04 which is well below the Regulatory Guide 1.25 guidance. A linear interpolation between the values reported in the NUREG, indicates that the release fractions given in Regulatory Guide 1.25 are applicable to peak rod burnups of up to 53.25 MWd/MTU. Since the peak rod burnup is assumed to be 60 MWd/MTU (60 GWd/t), the 20% multiplier given in NUREG/CR-5009 will be used.

6.4 Regulatory Guide 1.25: Thyroid Dose Calculations

The methodology given in Regulatory Guide 1.25 for determining the thyroid dose is to calculate the dose for each individual iodine isotope using the formula given below. The total thyroid dose is obtained by summing the dose due to the individual isotopes. The TRANSACT code will be used to perform this calculation. The following equations form the bases for the TRANSACT code.

The Thyroid dose is given by:

$$D_{thy} = \sum_i D_{thy,i} = \frac{F_g * I_i * F * P}{DF_{p,i} * DF_{f,i}} * B * R_i * \frac{\chi}{Q}$$

Or,

$$D_{thy,i} = \text{Curies Released to the environment} * B * DCF_i * \frac{\chi}{Q}$$

Where:

$D_{thy,i}$ = Thyroid dose for isotope i

F_g = Fraction of fuel rod inventory in the fuel rod void space (0.1 per RG 1.25)

I_i = Core iodine inventory, of isotope i, at time of accident (curies)

F = Fraction of core damaged so as to release void space iodine

P = fission product peaking factor

B = Breathing rate (3.47×10^{-4} m^3/sec)

DCF_i = Dose conversion factor for isotope i (rads/Ci)

$\frac{\chi}{Q}$ = atmospheric diffusion factor at receptor location (sec/m^3)

$DF_{p,i}$ = effective iodine decontamination factor for pool water for isotope i

$DF_{f,i}$ = effective iodine decontamination factor for filters for isotope i

The site boundary thyroid dose is given by:

$$D_{thy,i} = \frac{\chi}{Q} A_i * BR * DCF_i$$

Where :

$D_{thy,i}$ = Thyroid dose for isotope i (rem)

$\frac{\chi}{Q}$ = atmospheric diffusion factor at receptor location (sec/ m^3)

A_i = Activity of radionuclide i released from pool (Ci)

BR = Breathing rate (3.47×10^{-4} m^3/sec)

DCF_i = Dose conversion factor for isotope i (rem/Ci)

$$D_{thy,i} = 6.5 * 10^{-4} \frac{\text{sec}}{m^3} * R_i(Ci) * 3.47 * 10^{-4} \frac{m^3}{\text{sec}} * DCF_i(\frac{rem}{Ci})$$

The control room thyroid dose is determined as follows:

$$D_{CR,thy,i} = A_{CR,i} * BR * DCF_i$$

Where :

$D_{CR,thy,i}$ = Thyroid dose for isotope i (rem)

$A_{CR,i}$ = Activity of isotope i in control room (Ci)

BR = Breathing rate (3.47×10^{-4} m^3/sec)

DCF_i = Dose conversion factor for isotope i (rem/Ci)

The concentration of isotope i at the control room intake is given by:

$$A_{intake,i} = \frac{\chi}{Q} * R_i$$

Where :

$\frac{\chi}{Q}$ = control room atmospheric diffusion factor

R_i = release rate of radionuclide i from pool

The Activity in the control room is dependent on the Control Room Emergency Ventilation system intake flow.

This FHA analysis will use the dose conversion factors given in the TACT5 User's Guide (Reference 6). These dose conversion factors are consistent with ICRP 30 and are given as follows:

TACT5 Thyroid Dose Conversion Factors

Isotope	Dose Conversion Factors - Rem/Ci Inhaled
	TACT5
I-130	7.40E+04[3]
I-131	1.10E+06
I-132	6.30E+03
I-133	1.80E+05
I-134	1.10E+03
I-135	3.10E+04

[3]The dose conversion factor for I-130 is not given in Reference 6. This factor is taken from Reference 4 with the following conversion:

$$2 * 10^{-8} \frac{\text{Sievert}}{\text{Bq}} * \frac{100\,\text{Rem}}{\text{Sievert}} * \frac{1\,\text{Bq}}{2.703 * 10^{-11}\,\text{Ci}} = 7.40 * 10^{4} \frac{\text{Rem}}{\text{Ci}}$$

6.5 Regulatory Guide 1.25: Whole Body Dose Calculation

The methodology given in Regulatory Guide 1.25 for determining the whole-body dose is to calculate the dose for each individual isotope using the formula given below. The total whole-body dose is obtained by summing the dose due to the individual isotopes. The TRANSACT code will be used to perform this calculation.

The Whole-body dose is given by:

$$\beta \text{ Whole Body Dose (rad)} = 0.23 * \overline{E}_\beta * \chi$$

$$\gamma \text{ Whole Body Dose (rad)} = 0.25 * \overline{E}_\gamma * \chi$$

Where :

$\overline{E}_\beta$ = Average β energy per disintegration (Mev/dis)

$\overline{E}_\gamma$ = Average γ energy per disintegration (Mev/dis)

χ = Concentration time integral (curies * sec/ m^3)

$= \text{curies of Activity} * \frac{\chi}{Q}$

$\frac{\chi}{Q}$ = atmospheric diffusion factor at receptor location (sec/ m^3)

The whole-body dose at the exclusion area boundary can be found by use of the following formula:

$$Dose\ (Rem) = \frac{\chi}{Q} * (Dose\ Conversion Factors) * (Curies\ Released)$$

The control room whole-body dose can be found by use of the following formulas:

$$Dose_{CR\text{-}WB}(Rem) = \left(\frac{DCF_{Gamma}}{GF}\right) * A_{CR} * \frac{Occ}{V_{CR}} * \delta t$$

$$\text{Where: } GF = \frac{1173}{V^{0.338}} = \frac{1173}{(40{,}000)^{0.338}} = 32.643877 \text{ See ref. } 16$$

Occ = Control Room Occupancy

V_{CR} = Volume of Control Room

A_{CR} = Average Activity in Control Room

δt = Change in time

$$Dose_{CR\text{-}Skin}(Rem) = \left(\frac{DCF_{Gamma}}{GF} + DCF_{Beta}\right) * A_{CR} * \frac{Occ}{V_{CR}} * \delta t$$

$$Dose_{EAB\text{-}WB}(Rem) = \frac{\chi}{Q} * (DCF_{Gamma}) * (CuriesReleased)$$

$$Dose_{EAB\text{-}Skin}(Rem) = \frac{\chi}{Q} * (DCF_{Beta}) * (CuriesReleased)$$

This analysis will use the dose conversion factors given in the Tact5 User's Manual (Reference 6). This is consistent with ICRP 30 and are as follows:

TACT5 (ICRP 30) Dose Conversion Factors

Isotope	Dose Conversion Factors		
	Whole Body Gamma Rem-m^3/Ci-sec	Skin Beta Rem-m^3/Ci-sec	Skin Gamma Rem-m^3/Ci-sec
Kr-83m	1.27E-05	0.00E+00	1.36E-04
Kr-85m	2.31E-02	4.97E-02	3.20E-02
Kr-85	3.31E-04	4.84E-02	4.75E-04
Kr-87	1.33E-01	3.36E-01	1.85E-01
Kr-88	3.38E-01	7.76E-02	4.69E-01
Kr-89	3.03E-01	3.47E-01	4.21E-01
Xe-131m	1.25E-03	1.33E-02	2.71E-03
Xe-133m	4.29E-03	2.96E-02	7.00E-03
Xe-133	4.96E-03	9.67E-03	7.89E-03
Xe-135m	6.37E-02	2.14E-02	9.16E-02
Xe-135	3.59E-02	6.32E-02	5.07E-02
Xe-137	2.83E-02	4.59E-01	4.02E-02
Xe-138	1.87E-01	1.47E-01	2.61E-01
I-130	Not used	Not used	Not used
I-131	5.59E-02	3.07E-02	7.95E-02
I-132	3.55E-01	1.10E-01	5.07E-01
I-133	9.11E-02	8.90E-02	1.31E-01
I-134	4.11E-01	1.42E-01	5.86E-01
I-135	2.49E-01	7.86E-02	3.52E-01

6.6 Atmospheric Dispersion Factors

Standard Review Plan 2.3.4 (Reference 5) states that the atmospheric dispersion factor (i.e., χ/Q) can be determined on a probabilistic basis using the χ/Q value which will not be exceeded 95% of the time. Since the fuel handling accident assumes the release of all fission products to the environment over a two hour period, the two hour χ/Q values are used in this analysis. The two hour χ/Q values are as follows:

Control Room (0-2 hr)	5.6×10^{-3} sec/m^3	Reference 23
Exclusion Area Boundary ,EAB, (0-2 hr)	6.5×10^{-4} sec/m^3	Reference 13

Since the offsite dose limits of 75 Rem thyroid and 6 Rem whole-body are to be applied to both the exclusion area boundary and the low population zone, the offsite dose consequences of a fuel handling accident are most limiting at the exclusion area boundary due to the higher χ/Q value. On this basis, this calculation will consider only the dose consequences for the control room and the EAB.

6.7 Data

In the event that radioactivity is detected in the Unit 1 control room and/or the Unit 2 control room normal ventilation intake (Ref. 4), both Unit 1 and Unit 2 emergency ventilation systems are initiated. One of the two systems is subsequently shutdown as one is sufficient to maintain habitability. Although both systems meet the design intent for control room habitability, the Unit 1 ventilation system is limiting since it has a 2 inch recirculation charcoal bed compared to a 4 inch bed on the Unit 2 system. Therefore, in the control room analysis, the Unit 1 system was assumed to be operating and per the guidelines of Regulatory Guide 1.52 a 95% recirculation filter efficiency was used. Control room doses resulting from the FHA assumes that the control room recirculation charcoal absorber filter efficiency is 95% based on the limiting unit, Unit 1, which has a 2 inch charcoal absorber.

The data used in the evaluation of control room doses is given below:

Parameter	Input	Reference
Control Room Volume	4.00×10^4 ft^3	Ref. 27, 88E-0130-01
CR unfiltered inleakage	10 cfm	Regulatory Guide 1.78
CR filtered inleakage	333 cfm	ITS 3.7.9 Bases
CR out leakage	343 cfm [333 + 10 cfm]	N/A
CR recirculation	1667 cfm	
CR occupancy factor		

0-24 hrs.	1.0	Murphy and Campe (Ref. 16)
Breathing Rate (CR)	$3.47x10^{-4}$	
Control Room χ/Q		
0-2 hrs.	$5.6x10^{-3}$	Reference 23
CR filter efficiency		
recirc (2" filter)	95%	Regulatory Guide 1.52
intake (2-2" filters)	99%	Regulatory Guide 1.52

Additional FHA Input Data

Parameter	Input	Reference
Dose evaluation points	EAB and Control Room	N/A
Power level	2855 MWt	
Fuel Release Fraction		
Noble Gases (except Kr-85)	10%	Regulatory Guide 1.25
Kr-85	30%	Regulatory Guide 1.25
Halogens	12%	NUREG/CR 5009
Plate out	0%	not considered
Iodine Form		
Inorganic	99.75%	Regulatory Guide 1.25
Organic	0.25% "	Regulatory Guide 1.25
Offsite χ/Q		
EAB 0-2 hrs.	$6.5x10^{-4}$	FSAR Table 14-24
Breathing Rate (offsite)		Regulatory Guide 1.4
0-8 hrs.	$3.47x10^{-4}$	
Pool Decontamination Factors		
Inorganic Iodine	133	Regulatory Guide 1.25
Organic Iodine	1	Regulatory Guide 1.25
Fuel Handling Building (or Containment) filter iodine removal efficiencies (2 inch charcoal)		
Inorganic	90 %	(Regulatory Guide 1.25)
Organic	70 %	

7 CALCULATIONS

The fraction of activity remaining as a function of time after shutdown (relative to activity at shutdown) is shown below. As seen, the only significant isotopes remaining at the minimum time after shutdown (100 hrs) at which fuel movement could occur are I-131, I-133, Kr-85, Xe-131m, Xe-133m, and Xe-133.

Table 5.

Isotope	Half-life[4]	Decay Constant (hr^{-1})	Days After Shutdown			
			3	6	9	12
I-130	12.4 hr	5.5899e-02	1.79e-02	3.19e-04	5.70e-06	1.02e-07
I-131	***8.065 d***	***3.5810e-03***	***7.73e-01***	***5.97e-01***	***4.61e-01***	***3.57e-01***
I-132	2.284 hr	3.0348e-01	3.24e-10	1.05e-19	3.40e-29	1.10e-38
I-133	***20.8 hr***	***3.3324e-02***	***9.08e-02***	***8.24e-03***	***7.48e-04***	***6.79e-05***
I-134	52.3 m	7.9520e-01	1.36e-25	1.86e-50	2.54e-75	3.46e-100
I-135	6.7 hr	1.0345e-01	5.82e-04	3.39e-07	1.97e-10	1.15e-13
Kr-83m	1.86 hr	3.7266e-01	2.22e-12	4.95e-24	1.10e-35	2.45e-47
Kr-85m	4.4 hr	1.5753e-01	1.19e-05	1.41e-10	1.67e-15	1.98e-20
Kr-85	***10.74 yr***	***7.3674e-06***	***9.99e-01***	***9.99e-01***	***9.98e-01***	***9.98e-01***
Kr-87	76 m	5.4722e-01	7.74e-18	5.99e-35	4.64e-52	3.59e-69
Kr-88	2.79 hr	2.4844e-01	1.70e-08	2.90e-16	4.95e-24	8.43e-32
Kr-89	3.18 m	1.3078e+01	0.00e+00	0.00e+00	0.00e+00	0.00e+00
Xe-131m	***11.96 d***	***2.4148e-03***	***8.40e-01***	***7.06e-01***	***5.94e-01***	***4.99e-01***
Xe-133m	***2.26 d***	***1.2779e-02***	***3.98e-01***	***1.59e-01***	***6.33e-02***	***2.52e-02***
Xe-133	***5.27 d***	***5.4803e-03***	***6.74e-01***	***4.54e-01***	***3.06e-01***	***2.06e-01***
Xe-135m	15.7 m	2.6490e+00	1.48e-83	2.18e-166	3.21e-249	0.00e+00
Xe-135	9.16 hr	7.5671e-02	4.30e-03	1.85e-05	7.97e-08	3.43e-10
Xe-137	3.82 m	1.0887e+01	0.00e+00	0.00e+00	0.00e+00	0.00e+00
Xe-138	14.2 m	2.9288e+00	2.62e-92	6.89e-184	1.81e-275	0.00e+00

As shown above, the only significant isotopes that need to be considered at 100 hrs after shutdown are I-131, I-133, Kr-85, Xe-131m, Xe-133m, and Xe-133.

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[4] Data from "Chart of the Nuclides", Tenth Edition, (Reference 9)

Table 2 lists the half-lives and decay chains which need to be considered in this evaluation.

Table 2

Isotope	Half-life[5]	Daughter 1	Fraction 1[6]	Daughter 2	Fraction 2[4]
I-130	12.4 hr				
I-131	*8.065 d*	Xe-131m	100.00%		
I-132	2.284 hr				
I-133	*20.8 hr*	Xe-133	97.6%	Xe-133m	2.40%
I-134	52.3 m				
I-135	6.7 hr				
Kr-83m	1.86 hr				
Kr-85m	4.4 hr	Kr-85	23.00%		
Kr-85	*10.74 yr*				
Kr-87	76 m				
Kr-88	2.79 hr				
Kr-89	3.18 m				
Xe-131m	*11.96 d*				
Xe-133m	*2.26 d*	Xe-133	100.00%		
Xe-133	*5.27 d*				
Xe-135m	15.7 m				
Xe-135	9.16 hr				
Xe-137	3.82 m				
Xe-138	14.2 m				

Based on the above, the isotopes which will be considered in this calculation, based on half-life and importance as parent iotopes, are I-131, I-133, Kr-85, Xe-131m, Xe-133m and Xe-133. Although Kr-85m is a parent of Kr-85, this isotope is not included in the calculation since its short half-live (4.4 hours) will not result in any significant production of Kr-85 during the delay time being considered.

[5]Data from "Chart of the Nuclides", Tenth Edition, (Reference 9)

[6] Data from "Table of Isotopes", Lederer and Hollander, (Reference 10)

7.1 Calculation Procedure

Based on the source terms given in Section 7.2, the activity in the fuel is determined for 100 hours after shutdown. Based on the above discussion, this determination will consider decay of parent isotopes to obtain the fuel activity at the time of the fuel handling accident. The offsite and control room doses are calculated using the TRANSACT computer code. TRANSACT (Reference 6 and 17) is a Fortran computer code designed to model the transport of radionuclides through various compartments, the release to the environment, and the resulting onsite and offsite doses.

7.2 Radionuclide Source Terms

Fuel Gap activity is determined as follows:

$A_{gap,i} = A_{fuel,i} * 12\%$ For Iodines
$A_{gap,i} = A_{fuel,i} * 10\%$ For Noble Gases except Kr 85
$A_{gap,i} = A_{fuel,i} * 30\%$ For Kr 85

Since the Technical Specifications restrict fuel handling operations until 100 hours after shutdown, the only significant isotopes which will be considered are Kr-85, I-131, I-133, Xe-131m, XE-133m and Xe-133.

Table 6.
Fuel Gap Activity at 100 hours Delay, 82 Rods Failed (See Attachment 2)

Isotope	Gap Activity at 100 hrs (Ci) (Assembly)	Pool DF	Activity Released From Pool
Kr-85	1.341E+03	1	1.341E+03
Xe-131m	1.071E+04	1	1.071E+04
Xe-133m	1.265E+03	1	1.265E+03
Xe-133	8.399E+04	1	8.399E+04
I-131	2.742E+04	133	2.062E+02
I-133	2.655E+03	133	1.996E+01

Table 7.
Fuel Gap Activity at 100 hours Delay, 1 Rod Failed (See Attachment 2)

Isotope	Gap Activity at 100 hrs (Ci) (1 rod)	Pool DF	Activity Released From Pool
Kr-85	1.635E+01	1	1.635E+01
Xe-131m	1.306E+02	1	1.306E+02
Xe-133m	1.542E+01	1	1.542E+01
Xe-133	1.024E+03	1	1.024E+03
I-131	3.343E+02	133	2.514E+00
I-133	3.237E+01	133	2.434E-01

To account for the pool decontaimination factor, the TRANSACT input is as follows:

$$\text{TRANSACT Input Activity(Activity released from pool)} = \frac{\text{GapActivity}}{\text{Pool DF}}$$

This activity is multiplied within TRANSACT, by the inorganic species fraction (0.9975) and the organic species fraction (0.0025) to give the inorganic and organic iodine released.

In section 6.7, the Pool DF is given as follows:
Inorganic Iodine DF = 133
Organic Iodine DF = 1

To provide the correct organic iodine released activity, the TRANSACT input activity for the iodines (activity released from pool) is modified as follows:

$$\text{Organic Iodine released} = (\text{Iodine activity released from pool}) * 133 * 0.0025$$

$$\text{Organic Iodine released} = (\text{Iodine activity released from pool}) * 0.3325$$

This is accomplished in TRANSACT by setting the organic iodine species fraction to 0.3325

Therefore, the species fraction for the iodines is:

Inorganic Iodines	0.9975
Organic Iodines	0.3325

7.3 FHA inside Containment or Spent Fuel Building Without Filtration (Case 1)

This case evaluates the consequences of a fuel handling accident in either the fuel handling building or the containment without building integrity. A puff (2 hour) release is assumed with equipment hatch open in the spent fuel building or the containment equipment hatch open. This analysis will be based on typical licensing basis assumptions for EAB and Control Room dose consequences. Since this case is based on a puff release and typical regulatory guide assumptions, the results are applicable to a fuel handling accident either inside the spent fuel building or the containment. Input and output files for this case are given in Attachment 3.

7.4 FHA inside Containment or Spent Fuel Building With Filtration (Cases 2)

This case evaluates the release of radioisotopes following a fuel handling accident with the equipment hatch open. This is the same basic scenario as Case 1 except that credit for fuel pool ventilation and filtration system or the containment purge system, is utilized. The radioisotopes released from the fuel gap of the damaged fuel pass through the pool water and are then released at the pool surface. At this point, the pool sweep system directs the airflow, and the radioisotopes released from the pool, to the fuel handling area ventilation system. This system is designed to remove radioiodine from the airflow to minimize the offsite dose consequences of a fuel handling accident. The fuel handling area ventilation system is designed to meet the requirements of Regulatory Guide 1.52. An iodine decontamination efficiency of 90% for elemental iodine and 70% is assumed based on the guidance in Regulatory Guide 1.25.

In the spent fuel building, the pool sweep system filters any activity released from the damaged fuel as it evolves from the pool. The fuel handling area ventilation system is required to be OPERATIONAL whenever irradiated fuel is being moved in the storage pool and during crane operation with loads over the storage pool (Reference 19, Technical specification 3.15.1). Activity removed by the pool sweep system is directed to the filtration system before it enters the atmosphere.

Inclusion of credit for the filtration system only affects the EAB and Control Room thyroid doses since the filters are ineffective in removal of noble gases. Due to the design and operation of the pool sweep system, no credit for mixing within the fuel handling building is considered. Input and output files for this case are given in Attachment 4.

7.5 FHA With 1 Rod Failed Without Filtration (Cases 3)

This case is the same as Case 1, except that only one fuel rod is assumed to fail. Input and output files for this case are given in Attachment 5.

7.6 FHA With 1 Rod Failed With Filtration (Case 4)

This case is the same as Case 2, except that only one fuel rod is assumed to fail. Input and output files for this case are given in Attachment 6.

7.7 FHA With 82 Rod Failed Without Filtration Maximum CR Inleakage (Case 5)

One final case will be run. This case will determine the maximum control room inleakage that can occur while still meeting GDC 19. Input and output files for this case are given in Attachment 7.

The results of this case show that the requirements of GDC-19 are met with up to 98 cfm control room inleakage.

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ATTACHMENT 1
Decay Chains

The decay chain for a single parent-daughter chain is given below:

$$\boxed{\text{Isotope 1}} \longrightarrow \boxed{\text{Isotope 2}}$$

For a single isotope, the time rate of change is given by:

$$\frac{d\ N_1}{dt} = -\ \lambda_1\ N_1$$

Let η *be the Laplace transform of* N, *then,*

$$p\eta_1 - \eta_1(0) = -\lambda_1\eta_1$$

Or,

$$\eta_1 = \frac{\eta_1(0)}{p+\lambda_1}$$

For a decay chain of two isotopes, the time rate of change of isotope 2 is:

$$\frac{d\ N_2}{dt} = \lambda_1\ N_1 - \lambda_2\ N_2$$

Let η_2 *be the Laplace transform of* N_2 *then,*

$$p\eta_2 - \eta_2(0) = \lambda_1\eta_1 - \lambda_2\eta_2$$

Substituting for η_1 *gives:*

$$\eta_2\ (p+\lambda_2) = \lambda_1\frac{\eta_1(0)}{p+\lambda_1} + \eta_2(0)$$

$$\eta_2 = \frac{\lambda_1\eta_1(0)}{(p+\lambda_2)(p+\lambda_1)} + \frac{\eta_2(0)}{(p+\lambda_2)}$$

Taking the inverse Laplace transform gives:

$$N_2(t) = \frac{F_1\lambda_1 N_1(0)(e^{-\lambda_1 t} - e^{-\lambda_2 t})}{\lambda_2 - \lambda_1} + N_2(0)e^{-\lambda_2 t}$$

Where: F_1 *is the branching fraction for isotope 1*

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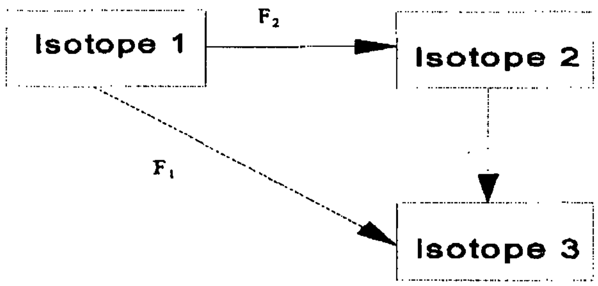
For a chain of three isotopes, the decay scheme is as follows:



For this decay chain, the time rate of change of isotope N_3 is given by:

$$\frac{d\,N_3}{dt} = F_1\lambda_1\,N_1 + F_2\lambda_2\,N_2 - \lambda_3\,N_3$$

Let η be the Laplace transform of N, then,

$p\eta_3 - \eta_3(0) = \lambda_1\eta_1 + \lambda_2\eta_2 - \lambda_3\eta_3$

Substituting the following for η_1 and η_2 from the previous page gives:

$$\eta_1 = \frac{F_1\eta_1(0)}{(p+\lambda_1)}$$

$$\eta_2 = \frac{F_2\lambda_1\eta_1(0)}{(p+\lambda_1)(p+\lambda_2)} + \frac{\eta_2(0)}{(p+\lambda_2)}$$

Where F_1 is the branching fraction for isotope 1 to isotope 3

And F_2 is the branching fraction for isotope 1 to isotope 2

Rearranging gives:

$$\eta_3 = \frac{\eta_3(0)}{p+\lambda_3} + F_1\frac{\lambda_1\eta_1(0)}{(p+\lambda_1)(p+\lambda_3)} + F_2\frac{\lambda_1\lambda_2\eta_1(0)}{(p+\lambda_1)(p+\lambda_2)(p+\lambda_3)} + \lambda_2\frac{\eta_2(0)}{(p+\lambda_2)(p+\lambda_3)}$$

Taking the inverse Laplace transform gives:

Note: The inverse transform of the third term can be found by the method of residues.

$$N_3(t) = N_3(0)e^{-\lambda_3 t} + F_1\frac{\lambda_1 N_1(0)(e^{-\lambda_1 t}-e^{-\lambda_3 t})}{(\lambda_3-\lambda_1)}$$

$$+ F_2\lambda_1\lambda_2 N_1(0)\left[\frac{e^{-\lambda_1 t}}{(-\lambda_1+\lambda_2)(-\lambda_1+\lambda_3)} + \frac{e^{-\lambda_2 t}}{(-\lambda_2+\lambda_1)(-\lambda_2+\lambda_3)} + \frac{e^{-\lambda_3 t}}{(-\lambda_3+\lambda_1)(-\lambda_3+\lambda_2)}\right]$$

$$+ \frac{\lambda_2 N_2(0)(e^{-\lambda_2 t}-e^{-\lambda_3 t})}{(\lambda_3-\lambda_2)}$$

Attachment 2
Decay of Isotopes

This attachment is used to calculate the activity at the end of the time in question.

The input data is as follows:

Decay constants for Radioisotopes:
Isotope number n is:
0 = Kr-85
1 = Xe-131m
2 = Xe-133m
3 = Xe-133
4 = I-131
5 = I-133

$Ci := 3.7 \cdot 10^{10} Bq$

$n := 0, 1..5$ $\quad t := 100 \cdot hr$

Decay Constant:
Reference 6

$\lambda_n :=$

$7.367 \cdot 10^{-6} \cdot hr^{-1}$
$2.415 \cdot 10^{-3} \cdot hr^{-1}$
$1.278 \cdot 10^{-2} \cdot hr^{-1}$
$5.480 \cdot 10^{-3} \cdot hr^{-1}$
$3.581 \cdot 10^{-3} \cdot hr^{-1}$
$3.332 \cdot 10^{-2} \cdot hr^{-1}$

Activity per Fuel Assembly at t=0, Reference 18
(used the most limiting activity per isotope)
(1.65 peaking factor)

$A_n :=$

$1.04 \cdot 10^{4} \cdot Ci$
$8.84 \cdot 10^{3} \cdot Ci$
$4.69 \cdot 10^{4} \cdot Ci$
$1.41 \cdot 10^{6} \cdot Ci$
$7.60 \cdot 10^{5} \cdot Ci$
$1.44 \cdot 10^{6} \cdot Ci$

Activity per Fuel Assembly at t=0 corrected to 1.8 peaking factor

$$A_n := \frac{(A_n \cdot 1.8)}{1.65}$$

$A_n =$

$1.135 \cdot 10^{4}$	Ci
$9.644 \cdot 10^{3}$	
$5.116 \cdot 10^{4}$	
$1.538 \cdot 10^{6}$	
$8.291 \cdot 10^{5}$	
$1.571 \cdot 10^{6}$	

Fraction in gap:

$F_n :=$

0.3
0.1
0.1
0.1
0.12
0.12

Activity in gap per fuel assembly:

$A_n := A_n \cdot F_n$

$A_n =$

$3.404 \cdot 10^{3}$	Ci
$9.644 \cdot 10^{2}$	
$5.116 \cdot 10^{3}$	
$1.538 \cdot 10^{5}$	
$9.949 \cdot 10^{4}$	
$1.885 \cdot 10^{5}$	

Activity in Fuel Gap (per fuel assembly) at start of fuel movement (t=100 hrs):

(See Attachment 1 for derivation of decay chains)

$$Agap_0 := A_0 \cdot \exp(-\lambda_0 \cdot t)$$ **Kr-85**

$$Agap_1 := \frac{\lambda_4 \cdot A_4 \cdot (\exp(-\lambda_4 \cdot t) - \exp(-\lambda_1 \cdot t))}{\lambda_1 - \lambda_4} + A_1 \cdot \exp(-\lambda_1 \cdot t)$$ **Xe-131m, daughter of I-131**

$$Agap_2 := 0.024 \cdot \frac{\lambda_5 \cdot A_5 \cdot (\exp(-\lambda_5 \cdot t) - \exp(-\lambda_2 \cdot t))}{(\lambda_2 - \lambda_5)} + A_2 \cdot \exp(-\lambda_2 \cdot t)$$

$$C_3 := A_3 \cdot \exp(-\lambda_3 \cdot t) + 0.976 \cdot \frac{\lambda_5 \cdot A_5 \cdot (\exp(-\lambda_5 \cdot t) - \exp(-\lambda_3 \cdot t))}{(\lambda_3 - \lambda_5)} + \lambda_2 \cdot A_2 \cdot \frac{\exp(-\lambda_2 \cdot t) - \exp(-\lambda_3 \cdot t)}{(\lambda_3 - \lambda_2)}$$

$$Agap_3 := C_3 + 0.024 \cdot \lambda_5 \cdot \lambda_2 \cdot A_5 \cdot \left[\frac{\exp(-\lambda_5 \cdot t)}{(-\lambda_5 + \lambda_2) \cdot (-\lambda_5 + \lambda_3)} + \frac{\exp(-\lambda_2 \cdot t)}{(-\lambda_2 + \lambda_5) \cdot (-\lambda_2 + \lambda_3)} + \frac{\exp(-\lambda_3 \cdot t)}{(-\lambda_3 + \lambda_5) \cdot (-\lambda_3 + \lambda_2)}\right]$$

Xe-133, daughter of I-133 and Xe-133m

$$Agap_4 := A_4 \cdot \exp(-\lambda_4 \cdot t)$$ **I-131**

$$Agap_5 := A_5 \cdot \exp(-\lambda_5 \cdot t)$$ **I-133**

$Agap_n =$

$3.401 \cdot 10^3$
$2.717 \cdot 10^4$
$3.208 \cdot 10^3$
$2.131 \cdot 10^5$
$6.954 \cdot 10^4$
$6.734 \cdot 10^3$

Ci

Gap activity at t=100 hours for fuel assembly

$Agap_n := Agap_n$

Gap activity at t=100 hours for one fuel rod:

$rods := 208$

$$Agap1 := \frac{Agap}{rods}$$

$Agap1_n =$

$1.635 \cdot 10^1$
$1.306 \cdot 10^2$
$1.542 \cdot 10^1$
$1.024 \cdot 10^3$
$3.343 \cdot 10^2$
$3.237 \cdot 10^1$

Ci

Gap Activity for one rod at t=100 hours

Gap activity for 82 rods

$failed := 82$

$Agap82 := Agap1 \cdot failed$

$Agap82_n =$

$1.341 \cdot 10^3$
$1.071 \cdot 10^4$
$1.265 \cdot 10^3$
$8.399 \cdot 10^4$
$2.742 \cdot 10^4$
$2.655 \cdot 10^3$

Ci

Attachment 3

To

0CAN080701

ANO-2 Fuel Handling Accident Analysis

PROC./WORK PLAN NO.	PROCEDURE/WORK PLAN TITLE:		
5010.015	ENGINEERING CALCULATIONS	CHANGE:	001-02-1

ARKANSAS NUCLEAR ONE
CALCULATION COVER SHEET

Calc. No.: 95-E-0031-01 **Rev. No.:** 2(2)

Calc. Title: Fuel Handling Accident Dose Analysis

Unit: 2 **Category:** Q
System(s): RB,FB

Calc. Type: NS

Components: NA

Comp Tag	Suffix	Comp Code

Topics:
Plt Area: **Bldg.** 2RB,2FB **Elev.** N/A
Room N/A **Wall**
Coordinates: N/A

Config. Checklist (per 5010.004) completed? (Y or N) Y
Document Comment/Resolution Form completed? (Y or N) Y

Abstract (Included Purpose/Results):
This Calculation: Determines the offsite and Control Room dose consequences of a fuel handling accident occurring in either the containment or the spent fuel handling area. The results establish the minimum time delay for fuel movement. Revision 2 of this calculation revises the previous results for steam generator replacement and power uprate. Revision 2(1) incorporates the use of a new Control Room χ/Q (0-8 hour) value in the calculation and gives the skin dose in terms of "total" skin dose. Revision 2(2) incorporates the use of a new Control Room χ/Q (0-8 hour) value in the calculation and updated fuel source terms.

Pages Revised and/or Added: Rev. 2(2) changes marked w/ revision bars.

Purpose of Revision:
The purpose of revision 2 is to reevaluate the fuel handling accident with a power level of 3087 Mwt for the Unit 2 power uprate and to include the updated Control Room χ/Q. Revision 2(1) incorporates the use of a new Control Room χ/Q (0-8 hour) value in the calculation and gives the skin dose in terms of "total" skin dose. Revision 2(2) incorporates the use of a new Control Room χ/Q (0-2 hour) value in the calculation.

APA 7/25/01

Initiating Documents	Resulting Document(s)	Key Design Input Documents
~~ER-980528 E202, "Revise ANO2 Dose Calculations to Support Power Uprate".~~	SAR Section 15.1.23	Calculation 98-E-0029-01, Rev. 0
ER 002344 E201		Calculation 95-E-0030-10, Rev. 0(2)
		Engineering Report 98-R-2005-01

Verification Method: **Design Review** X **Alternate Calculation** **Qualification Testing**
Amends Calc(s): N/A
Supersedes Calc(s): N/A
Computer Software Used: TRANSACT

	(Print Name)	(Initials)	(Date)		(Print Name)	(Initials)	(Date)
By:	Marvin Morris	MMM	9/21/2000	**Rvw'd:**	Chad Creese	CC	10/26/00
Chk'd :	Joanne Morris	JM	9/21/2000	**Apv'd**	D.W. Fouts	DWF	10/26/00

Check if Additional Revisions: ✓

Table of Contents

1 PURPOSE

The purpose of this calculation is to evaluate the offsite (Exclusion Area Boundary, EAB) and control room dose consequences of a fuel handling accident (FHA). This accident is postulated to occur either in the containment or in the spent fuel pool area. This calculation consolidates design basis data and addresses issues that were not addressed in previous evaluations of the fuel handling accident. These include:

- proposed relaxation of containment integrity requirements during fuel handling
- proposed relaxation of spent fuel building integrity requirements during fuel handling
- use of ICRP 30 Dose Conversion Factors
- use of power uprate (3087 Mwt) source terms obtained from the ORIGEN-II code
- updated Control Room χ/Q values (Calculation 95-E-0030-10, Revision 0(2)).

The following cases are analyzed:

Case 1: This case evaluates the limiting fuel handling event for informational purposes: 1) Failure of an entire assembly (236 rods) in the fuel building with equipment hatch open, or 2) Failure of an entire assembly in the containment with the containment personnel or equipment hatch open. Analysis will be based on typical licensing basis assumptions for EAB and Control Room dose consequences. Offsite (EAB) and onsite doses will be calculated based on a puff release and Regulatory Guide 1.25 assumptions, the results are applicable to a fuel handling accident either inside the spent fuel building or the containment. Since the Technical Specifications restrict fuel handling operations until 100 hours after shutdown, a decay time of 100 hours is assumed.

Case 2: Failure of an entire assembly (236 rods, 100 hours decay) in the containment or fuel building with credit for fuel pool ventilation and filtration system or containment purge filtration system. Calculation assumes that the equipment hatch is closed in the spent fuel building or the containment personnel airlock and equipment hatch are closed. Analysis will be based on typical licensing basis assumptions for EAB and Control Room dose consequences.

Case 2a: Same as Case 2, but with 61 cfm control room unfiltered inleakage.

Case 3: FHA in the fuel building with credit for the fuel pool ventilation and filtration system. Offsite and onsite doses will be calculated for failure of 60 fuel rods and a decay time 100 hrs.

Case 3a: Same as Case 3, but with 61 cfm control room unfiltered inleakage.

Case 4 FHA in the fuel building without credit for the fuel pool ventilation and filtration system. Offsite and onsite doses will be calculated for failure of 60 fuel rods and a decay time 100 hrs. Case 4 evaluates the ANO Unit 2 design basis fuel handling accident.

Case 4a Same as Case 4, but with 61 cfm control room unfiltered inleakage.

2 Cycle dependent Parameters

Some of the parameters utilized in this calculation are based on the current fuel type and cycle exposure. Consequently, this calculation should be reviewed for adequacy each cycle. The following table lists the calculation parameters that are cycle-dependent along with the bounding values used in the analysis.

Calculation Parameter	Rationale
Rod internal pressure <1200 psig	Required by Regulatory Guide 1.25 (Position 1.b)
Core radial power peaking factor ≥1.65	ANO-2 Technical Specification, Section 3/4 9.1. Core radial power peaking factor ≥1.65 required by Regulatory Guide 1.25 (Position 1.e)
Peak linear power density ≤20.5 kW/ft	Peak linear power density ≤20.5 kW/ft required by Regulatory Guide 1.25 (Position 1)
Maximum fuel centerline temperature<4500°F	Required by Regulatory Guide 1.25 (Position 1)
Fuel batch average burnup for peak assembly ≤65,000 MWD/MTU. [maximum isotopic inventory for 4 w/o <u>or</u> 5 w/o enrichment used]	Used in development of radionuclide source terms with the ORIGEN-II code. [Note: Fuel batch average burnup for peak assembly ≤25,000 MWD/ton is required by Regulatory Guide 1.25 (Position 1)]
Maximum number of fuel rods damaged during a fuel handling accident = 236	Assumed value based on failure of all rods within a single assembly
Reactor and spent fuel pool water level sufficient to ensure 23 feet water coverage above any damaged fuel	Used in the basis for a decontamination factor of 100 for iodine per Regulatory Guide 1.25 (Position 1.c)
χ/Q at exclusion area boundary (0-2 hrs) ≤ 6.5×10^{-4} sec/m^3	Used in calculating the dose consequences of fuel failures
Spent Fuel Building and containment exhaust filter removal efficiency for iodine = 90% for inorganic and 70% for organic	Removal efficiency for iodine per Regulatory Guide 1.25 (Position 1.j). Based on 2-inch charcoal bed depth.

3 RESULTS

Standard Review Plan (SRP) 15.7.4 requires that the dose consequences of a FHA must be well within the limits given in 10CFR100. "Well within" is further defined in this SRP as 25% of the limits of 10CFR100. This results in an offsite dose limit of 6 rem to the whole body and 75 rem to the thyroid for the duration of the accident.

Case 1

The results for the limiting FHA: 1) FHA (236 rods, 100 hrs. decay) in the fuel building with equipment hatch open, or 2) FHA (236 rods, 100 hrs. decay) in the containment with the containment personnel or equipment hatch open are given below. Case 1 was evaluated for informational purposes. It is an event that is beyond the design basis and is not expected to meet dose limits given above.

Table 1.
CASE 1 - ANO UNIT 2 FHA
FHA with 100 hour delay and no ESF filtration

Dose Category	EAB (2hr) Rem	Control Room Rem
Whole Body	3.928E-01	**1.278E-02**
Skin[1]	1.475E+00	**9.921E-01**
Thyroid	2.081E+02	**2.645E+00**

Case 2 and 2a

This case evaluates a FHA (236 rods, 100 hours decay) in the containment or fuel building with credit for filtration. The results for Case 2 are given in Table 2

Table 2.
CASE 2 and 2a- ANO UNIT 2 FHA
FHA with filtration, 100 hour decay

Dose Category	EAB (2 hr.) Rem	Control Room Rem (10 cfm inleakage)	Control Room Rem (61 cfm inleakage)
Whole Body	3.377E-01	9.329E-03	1.020E-2
Skin[1]	1.316E+00	7.247E-01	7.920E-1
Thyroid	2.904E+01	3.481E-01	1.641

Case 3 and 3a

The results for the 60-rod failure FHA in the fuel building with credit for the fuel pool ventilation and filtration system are given below.

Table 3.
CASE 3 and 3a –ANO UNIT 2 FHA – 60 Rod Failure
FHA with filtration, 100 hour decay

Dose Category	EAB (2 hr.) Rem	Control Room Rem (10 cfm inleakage)	Control Room Rem (61 cfm inleakage)
Whole Body	8.586E-02	2.372E-03	2.593E-3
Skin[1]	3.345E-01	1.842E-01	2.013E-01
Thyroid	7.381E+00	8.848E-02	4.171E-1

Case 4 and 4a

The onsite and offsite doses for a 60-rod failure FHA in the fuel building without credit for the fuel pool ventilation and filtration system are given below. Case 4 results constitute the ANO Unit 2 design basis fuel handling accident:

Table 4.
CASE 4 and 4a- ANO UNIT 2 FHA – 60 Rod Failure
FHA without filtration, 100 hour decay

Dose Category	EAB (2 hr.) Rem	Control Room Rem (10 cfm inleakage)	Control Room Rem (61 cfm inleakage)
Whole Body	**9.985E-02**	**3.245E-03**	**3.519E-3**
Skin[1]	**3.750E-01**	**2.519E-01**	**2.724E-1**
Thyroid	**5.289E+01**	**6.722E-01**	**3.162**

1 The "total" skin dose is reported, that is, the skin dose resulting from gamma radiation plus that resulting from beta radiation.

For the design basis FHA in which 60 rods are damaged, the EAB doses are within the regulatory limits without credit for filtration. For a case in which fewer rods are damaged, the resultant onsite and offsite doses are a direct ratio of the above results. For 16 failed rods with ESF filtration, the EAB Thyroid dose would be: $\text{EAB Dose} = \frac{16}{60} * 52.89\ \text{rem} = 14.1\ \text{rem}$

4 REFERENCES

1. Regulatory Guide 1.25, (Safety Guide 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, March 23, 1972.
2. NUREG-0800 (Standard Review Plan), Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents", Revision 1, July 1981.
3. 10CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance", last amendment date - June 24, 1975.
4. Annals of the ICRP, ICRP 30 supplement to Part 1, July 1978.
5. Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis", Revision 1, December 1975.
6. 10CFR, Part 50, Appendix A, Criterion 60, "Control of Releases to the Environment", January 1, 1987.
7. 10CFR, Part 50, Appendix A, Criterion 61, "Fuel Storage and Handling and Radioactivity Control", January 1, 1987.
8. NUREG-800, Standard Review Plan 2.3.4, "Short Term Dispersion Estimates for Accidental Atmospheric Releases", revision 1, July 1981.
9. NUREG/CR-5106, *User's Guide for the TACT5 Computer Code*, Appendix e, June 1988.
10. AP&L Memorandum entitled "Fuel Handling Accident (FHA) Re-analysis, NO-94-360, dated August 22, 1984, from Tony Mansell to A. B. McGregor.
11. NUREG/CR-5009, *Assessment of the Use of Extended Burnup Fuel in Light Water Reactors*, February 1988.
12. *Chart of the Nuclides*, Knolles Atomic Power Laboratory, Tenth Edition, Norman E. Holden and F. William Walker
13. *Table of Isotopes*, C. M. Lederer, J. M. Hollander, and I. Perlman, 6th Edition, John Wiley and Sons, New York, 1967
14. Regulatory Guide 1.52, "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants", Rev. 2, 1978
15. Regulatory Guide 1.109, "Calculation of Annual doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance With 1 CFR 50, Appendix I", Revision 1, 1977
16. ANO Calculation No. 91-E-0017-01
17. ANO Unit-2, Technical Specifications
18. Procedure 1104.034, Rev. 20, Control Room Air Conditioning, 9/18/92
19. Procedure 2104.034, Rev. 14, Control Room Air Conditioning and Ventilation, 7/11/91
20. Procedure 2104.007, Rev. 12, Control Room Emergency Air Conditioning and Ventilation, 9/18/92

21. Procedure 5120.421, Rev. 0, "Inplace Leak Testing of Ventilation Systems Containing HEPA and Carbon Filters".
23. Report No. 92-R-2018-01, Justification for a Mixed Containment, 7/31/92
24. Calc. A-22, RB Net Free Volume, Rev. 1, 5/2/75
25. MCS-88-0717, AP&L letter to Bill Watson, from Bill Eaton, Subject: Revised Control Room LOCA Doses, 10/6/88
26. ANO2 SAR Table 15.1.23-1 and 15.1.23-2, Amendment No. 12
27. Standard Review Plan 6.4, "Control Room Habitability System", Rev. 2, July 1981
28. Calculation No. J-20, "Site Boundary Dose For Revised Fuel Handling Accident".
29. Calc. A-28, "RBSS Sprayed Volume", Rev. 0, 7/22/77
30. Calc. J-27, "Fuel Handling Accident", Rev. 0, 4/12/79 (Mech/Nuclear Calc. 111)
31. ANO2 SAR Section 15.1.23, Amendment No. 12
32. Murphy and Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19", August 1974
33. TRANSACT Computer Code, CDP 93-C-0003-01, Rev. 1
34. Calculation No. 98-E-0029-01, "ANO-2 Power Uprate Source Terms for LOCA, Non-LOCA, and FHA".
35. ANO Calculation No. 88-E-0130-01
36. ANO Unit 2 FSAR, Amendment 12, Table 1.3-1
37. Interoffice Memorandum, CEO-95/00095, dated 3/21/95, from D. L. Smith to J. W. Cotton, "Maximum Cold Internal Pressure of a PWR Fuel Rod"
38. Calculation 95-E-0030-10, Rev. 0(2), "Control Room Atmospheric Dispersion Factors for Non-LOCA Accidents at ANO Units 1 & 2, August 23,1999.
39. Engineering Report 98-R-2005-01, Rev. 5, "ANO-2 Cycle 15/16 Safety Analysis Groundrules".
40. ANO-2 Technical Specification, Section 3/4 9.1.

5 CURRENT LICENSING BASIS

The current licensing basis for a FHA is given in Revision 33 to the ANO-2 FSAR. FSAR Section 15.1.23 provides a discussion of the FHA, methods of analysis and analytical results. The primary assumptions used in this analysis were obtained from Regulatory Guide 1.25. The FHA analyses considered two basic cases: (1) A postulated fuel drop which results in the damage to 60 fuel rods [four rows], and (2) A postulated fuel drop which results in the damage to 16 fuel rods [failure of outer row]. An additional case was evaluated to demonstrate compliance with the requirements of Regulatory Guide 1.13. This case assumed the failure of 236 rods (failure of all rods in an assembly).

6 ASSUMPTIONS AND DATA

6.1 General Regulatory Requirements

Reference 2 gives the acceptance criteria related to the radiological consequences of a fuel handling accident:

1. The calculated whole-body and thyroid doses at the exclusion area and low population zone boundaries are to be "well within" the guidelines of 10CFR100.11 (Reference 3: 300 Rem to the thyroid, 25 Rem to the whole body). Reference 2 states that "well within" means ≤25% of the 10CFR100.11 limits. This is given as 75 Rem to the thyroid and 6 Rem[1] to the whole-body.

2. The calculated dose shall incorporate the appropriate conservative assumptions stated in Regulatory Guide 1.25 (Reference 1) with the exception of the atmospheric dispersion factors (i.e., χ/Q values) which should be determined in accordance with Standard Review Plan 2.3.4 (Reference 8).

A discussion of the Regulatory Guide 1.25 assumptions is provided in Sections 6.2 through 6.5. A discussion of the application of Standard Review Plan 2.3.4 methodology is given is Section 6.6.

The acceptance criteria for a fuel handling accident with regard to control room doses is given in 10CFR50, Appendix A, General Design Criterion (GDC) 19. The regulatory guidance given in GDC 19 states that adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures to any part of the body in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

[1]Actually, 25% of 25 Rem is 6.25 Rem. However, since Reference 2 defines 25% of 25 Rem as 6 Rem, this analysis will use 6 Rem as the limit.

Standard Review Plan 6.4 (Reference 27) further defines the control room dose limit as 5 rem whole body and 30 rem thyroid or skin dose.

6.2 Regulatory Guide 1.25: Overview

The basic assumptions given in Regulatory Guide 1.25 are related to the following:

1. The manner of calculating the quantity of activity (both iodines and noble gases) released to the containment, spent fuel building, or the environment. Guidance is provided on calculating the following:

   - the fission product inventory within the fuel pellets
   - the fraction of the fuel pellet fission product inventory which is released from the fuel pellets to the pellet/cladding gap and consequently available for release to the pool water following cladding failure
   - the pool decontamination factors for the fission products
   - the iodine removal efficiencies for plant adsorbers/filters

   Additional details of these assumptions are given in Section 6.3.

2. The manner of calculating the thyroid dose based on the curies of iodine released into the environment. Details of these calculations are given in Section 6.4.

3. The manner of calculating the whole-body dose based on the curies of noble gas released into the environment. Details of these calculations are given in Section 6.5.

6.3 Regulatory Guide 1.25: Activity Releases

The major guidance and assumptions related to the quantity of activity released are as follows:

1. *Requirement:* The accident occurs at the earliest time that fuel handling can begin as allowed by the Technical Specifications. Radioactive decay of the fission product inventory during the interval between shutdown and the commencement of fuel handling is taken into consideration.

   Response: Based on the current revision of Technical Specification 3.9.3.a, the minimum subcritical time before fuel movement is 100 hours.

2. *Requirement:* The maximum fuel rod pressurization (after the above decay time) is 1200 psig.

Response: FSAR Section 15.1.23.2.2 states that the maximum fuel rod pressure 100 hours after a refueling shutdown will be less than 1,200 psig. This statement is supported by the Fuel Handling Accident Re-analysis given in memorandum NO-84-360 (Reference 10). This re-analysis was performed to determine if the dose consequences of a fuel handling accident were increased for fuel assemblies with an average burnup of 60,000 MWD/MTU rather than the 25,000 MWD/MTU burnup given in Regulatory Guide 1.25. This evaluation concluded that the maximum fuel rod pressure would be 1,192 psig with an average burnup of 60,000 MWD/MTU. Conservatisms in the calculation methodology are given in the referenced memo. Reference 37 states that the maximum fuel rod pressure during refueling will be less than 1,200 psig if the hot internal pressure is less than the reactor operating pressure. Fuel rod pressure during refueling for CY-16 (power uprate) fuel would also be less than 1200 psig based on a burnup of 65,000 MWD/MTU.

3. *Requirement:* The minimum water depth between the top of the damaged fuel rods and the fuel pool surface is 23 feet.

Response: This water level is required to allow the use of an overall decontamination factor for iodine of 100 (i.e., 1% of the iodine released to the pool will escape from the pool and 99% will be retained in the pool). The pool decontamination factor for noble gases is 1 (i.e., no noble gases retained in the pool water). The minimum depth of water above the potentially damaged fuel rods is as follows:

Containment	23 ft	Technical Specification 3.9.9
Fuel Pool	23 ft	Technical Specification 3.9.10

Consequently the use of a water coverage of 23 feet is conservative relative to the actual water coverage. Therefore, the iodine decontamination factor given in the Regulatory Guide is also conservative.

4. *Requirement:* All of the gap activity in the damaged rods is released and consists of 10% of the total noble gas other than Kr-85, 30% of the Kr-85, and 10% of the total radioactive iodine in the rods at the time of the accident. For the purpose of sizing filters for the fuel handling accident addressed in this guide, 30% of the I-127 and I-129 inventory is assumed to be released from the damaged fuel.

Response: NUREG/CR-5009 (Reference 11) discusses the variations in plenum inventories for extended burnup fuel. Table 3.6 of Reference 11 indicates that the I-131 inventory is 20% higher than in Regulatory Guide 1.25 for rod burnups of 60 GWD/MTU. Therefore the gap activity for iodine is assumed to be 12%. The use of 30% of the I-127 and I-129 inventory is not used since this is specified in Regulatory Guide 1.25 only for filter sizing.

Noble Gas Release (except Kr-85)	10%
Kr-85 Release	30%
Radioactive Iodine Release	12%

5. *Requirement:* The values used for individual fission product inventories are calculated assuming full power operation at the end of core life immediately proceeding shutdown and such calculation should include an appropriate radial peaking factor. The minimum acceptable radial peaking factor is 1.65 for PWR's.

Response: This analysis uses the source terms developed by use of the ORIGEN-II code [34]. Uprate (CY-16) power at ANO Unit 2 is 3026 Mwt [39]. The FHA source terms are based on 102% of rated (*i.e.*, 3087 Mwt) and a radial peaking factor of 1.70. The radial peaking factor is consistent with Regulatory Guide 1.25, which specifies a minimum radial peaking factor of 1.65.

6. *Requirement:* The fission products which are assumed to be released to the containment/auxiliary building atmosphere are assumed to escape into the environment within two hours.

Response: All releases of fission products to the environment are assumed to occur over a two hour time period. Also, this calculation conservatively assumes that there is no decay of the fission products during residency in the containment or auxiliary building. Two-hour χ/Q factors are used to calculate the offsite doses.

7. *Requirement:* If it can be shown that the building atmosphere is exhausted through

adsorbers designed to remove iodine, the removal efficiency is 90% for inorganic iodine and 70% for organic species. [This is based on a 2-inch charcoal bed depth with 1/4-second residence time.]

Response: Iodine filter efficiencies of 90% for inorganic iodine and 70% for organic species are used in this calculation. This assumption is conservative since the ANO Technical Specifications (3.9.4 and 3.9.11) require filter efficiencies of ≥99.95 when tested in-place with a halogenated hydrocarbon refrigerant test gas. The Technical Specifications also require testing on an 18 month interval and these tests are performed in accordance with the criterion of Regulatory Guide 1.52 (Reference 14) for activated charcoal beds of 2 inches or more.

This calculation will use the following iodine filter removal efficiencies:

Inorganic	90%
Organic	70%

8. *Requirement:* The iodine gap inventory is composed of inorganic species (99.75%) and organic species (0.25%).

Response: The analysis will use these assumptions:

Inorganic Iodine	99.75%
Organic Iodine	0.25%

9. *Requirement:* The pool decontamination factors for the inorganic and organic species are 133 and 1, respectively, giving an overall effective decontamination factor of 100 (i.e., the pool water retains 99% of the total iodine released from the damaged rods). This difference in decontamination factors for inorganic and organic iodine species results in the iodine above the fuel pool being composed of 75% inorganic and 25% organic species.

Response: The analysis will use these assumptions:

Inorganic Iodine DF	133
Organic Iodine DF	1

10. *Requirement:* The retention of noble gases in the pool is negligible (i.e., decontamination factor of 1).

Response: The analysis will use this assumption. [Noble Gas Pool DF = 1]

11. *Requirement:* The effluent from the filter system passes directly to the emergency exhaust system without mixing (credit for mixing will be allowed in some cases: the amount of credit will be evaluated on an individual case basis) in the surrounding building atmosphere and is then released (as an elevated plume for those facilities with stacks). Credit for an elevated

release will be given only if the point of release is (a) more than two and one-half times the height of any structure close enough to affect the dispersion of the plume or (2) located far enough from any structure which could affect the dispersion of the plume.

Response: Credit for mixing in the containment and spent fuel building atmosphere is taken for the FHA inside the containment (Case 3). This case was performed to determine the effect of mixing on the offsite and control room doses. For all other cases evaluated, credit for mixing is not taken. Releases are assumed to be ground level releases.

The guidance provided in Regulatory Guide 1.25 states that the above assumptions are only applicable if the following three conditions are met. These conditions are primarily related to the fraction of fission products that are released to the fuel/cladding gap. If any condition is not met, the impact of this on the above assumptions will need assessment. As shown below, all conditions are met for ANO Unit 2 or suitable adjustments in the release fractions are provided.

1. *Requirement:* The peak linear heat generation rate (LHGR) is not to exceed 20.5 kW/ft.

   Response: The ANO Unit 2 Technical Specifications 2.1.1.2 limits the LHGR to 21.0 kW/ft. This value represents the maximum overpower (accident) thermal output. During normal operations, the maximum thermal output, which is appropriate for a fuel handling accident, is 12.7 kW/ft (Reference 36). Even this value would only be reached for low burnup fuel. For the CY-16 (power uprate) fuel, the target linear heat rate limit of the hot rod at hot full power is 13.7 kW/ft. Therefore, fuel centerline melting is not a concern and the assumptions of Regulatory Guide 1.25 are applicable.

2. *Requirement:* The maximum fuel centerline temperature is to be less that 4500°F.

   Response: The peak centerline temperature is 2000°F (Reference 10, page 4) for a fuel burnup of 60 GWD/MTU. Reference 36 gives a maximum fuel centerline temperature of 3420°F at 100% power. For CY-16, the fuel maximum centerline temperature should be bounded by the Regulatory Guide condition of 4500°F. Therefore, the Regulatory Guide assumptions related to fission products that are released to the fuel/cladding gap are valid for this calculation.

3. *Requirement:* The average burnup of the peak bundle is less than 25,000 MWd/t.

   Response: The average burnup of the peak bundle is assumed to be 65 MWd/MTU

for the CY-16 (power uprate) fuel. Higher fuel burnup has been shown to have no effect on the internal fuel rod pressure up to a burnup of 60 MWd/MTU (Reference 10). However, there may be an increase in fuel rod gap inventory for high burnup fuel.

NUREG/CR-5009 (Reference 11) discusses the variations in plenum inventories for extended burnup fuel. Table 3.6 of Reference 11 indicates that the I-131 inventory is 20% higher than in Regulatory Guide 1.25 for rod burnups of 60 GWd/t. For rod burnups of 33 GWd/t, the NUREG predicts an I-131 release fraction of 0.04 which is well below the Regulatory Guide 1.25 guidance. A linear interpolation between the values reported in the NUREG, indicates that the release fractions given in Regulatory Guide 1.25 are applicable to peak rod burnups of up to 53.25 MWd/MTU. Since the peak rod burnup is assumed to be 65 MWd/MTU (65 GWd/t), a release fraction of 0.135 will be used (see calculation below).

NUREG /CR-5009 iodine release fractions as a function of burnup. [See Table 3.6, page 3-12, of NUREG /CR-5009]

$i := 0..1$

$Release_i :=$ $\begin{array}{|c|}\hline 0.04 \\ \hline 0.12 \\ \hline \end{array}$ $\quad Burnup_i :=$ $\begin{array}{|c|}\hline 33 \\ \hline 60 \\ \hline \end{array}$

For the RG 1.25 iodine release fraction of 0.1, the corresponding burnup is:

$\mathrm{linterp}(Release, Burnup, 0.1) = 53.25$

The iodine release fraction for a burnup of 65 Gwd/t (rod average) is:

$\mathrm{linterp}(Burnup, Release, 65) = 0.135$

6.4 Regulatory Guide 1.25: Thyroid Dose Calculations

The methodology given in Regulatory Guide 1.25 for determining the thyroid dose is to calculate the dose for each individual iodine isotope using the formula given below. The total thyroid dose is obtained by summing the dose due to the individual isotopes.

The Thyroid dose is given by:

$$D_{thy} = \sum_i D_{thy,i} = \frac{F_g * I_i * F * P}{DF_{p,i} * DF_{f,i}} * B * R_i * \frac{\chi}{Q}$$

Or,

$$D_{thy,i} = \text{Curies Released to the environment} * B * R_i * \frac{\chi}{Q}$$

Where:

$D_{thy,i}$ = Thyroid dose for isotope i

F_g = Fraction of fuel rod inventory in the fuel rod void space (0.1 per RG 1.25)

I_i = Core iodine inventory, of isotope i, at time of accident (curies)

F = Fraction of core damaged so as to release void space iodine

P = fission product peaking factor

B = Breathing rate (3.47×10^{-4} m^3/sec)

R_i = Dose conversion factor for isotope i (rads/Ci)

$\frac{\chi}{Q}$ = atmospheric diffusion factor at receptor location (sec/ m^3)

$DF_{p,i}$ = effective iodine decontamination factor for pool water for isotope i

$DF_{f,i}$ = effective iodine decontamination factor for filters for isotope i

The site boundary thyroid dose is given by:

$$D_{thy,i} = \frac{\chi}{Q} R_i * BR * DCF_i$$

Where :

$D_{thy,i}$ = Thyroid dose for isotope i (rem)

$\frac{\chi}{Q}$ = atmospheric diffusion factor at receptor location (sec/m^3)

R_i = Activity of radionuclide i released from pool (Ci)

BR = Breathing rate ($3.47 x 10^{-4}$ m^3/sec)

DCF_i = Dose conversion factor for isotope i (rem/Ci)

$$D_{thy,i} = 6.5 * 10^{-4} \frac{sec}{m^3} * R_i(Ci) * 3.47 * 10^{-4} \frac{m^3}{sec} * DCF_i(\frac{rem}{Ci})$$

The control room thyroid dose is determined as follows:

$$D_{CR,thy,i} = A_{CR,i} * BR * DCF_i$$

Where :

$D_{CR,thy,i}$ = Thyroid dose for isotope i (rem)

$A_{CR,i}$ = Activity of isotope i in control room (Ci)

BR = Breathing rate ($3.47 x 10^{-4}$ m^3/sec)

DCF_i = Dose conversion factor for isotope i (rem/Ci)

The concentration of isotope i at the control room intake is given by :

$$A_{intake,i} = \frac{\chi}{Q} * R_i$$

Where :

$\frac{\chi}{Q}$ = control room atmospheric diffusion factor

R_i = release rate of radionuclide i from pool

The revised FHA analysis will use the dose conversion factors given in the TACT5 User's Guide (Reference 9). These dose conversion factors are consistent with ICRP 30 and the dose conversion factors used in the maximum hypothetical accident (Reference 16). The Thyroid dose conversion factors used are as follows:

TACT5 Thyroid Dose Conversion Factors

Isotope	Dose Conversion Factors - Rem/Ci Inhaled
	TACT5
I-130	7.40e+04[2]
I-131	1.10e+06
I-132	6.30e+03
I-133	1.80e+05
I-134	1.10e+03
I-135	3.10e+04

[2]The dose conversion factor for I-130 is not given in Reference 9. This factor is taken from Reference 4 with the following conversion:

$$2 * 10^{-8}\,\frac{\text{Sievert}}{\text{Bq}} * \frac{100\,\text{Rem}}{\text{Sievert}} * \frac{1\,\text{Bq}}{2.703 * 10^{-11}\,\text{Ci}} = 7.40 * 10^{4}\,\frac{\text{Rem}}{\text{Ci}}$$

6.5 Regulatory Guide 1.25: Whole Body Dose Calculation

The methodology given in Regulatory Guide 1.25 for determining the whole-body dose is to calculate the dose for each individual isotope using the formula given below. The total whole-body dose is obtained by summing the dose due to the individual isotopes.

The Whole-body dose is given by:

$$\beta \text{ Whole Body Dose (rad)} = 0.23 * \overline{E}_\beta * \chi$$

$$\gamma \text{ Whole Body Dose (rad)} = 0.25 * \overline{E}_\gamma * \chi$$

Where :

$\overline{E}_\beta$ = Average β energy per disintegration (Mev/dis)

$\overline{E}_\gamma$ = Average γ energy per disintegration (Mev/dis)

χ = Concentration time integral (curies * sec/ m^3)

$$= \text{curies of Activity} * \frac{\chi}{Q}$$

$\frac{\chi}{Q}$ = atmospheric diffusion factor at receptor location (sec/ m^3)

The whole-body dose at the exclusion area boundary can be found by use of the following formula:

$$\textit{Dose (Rem)} = \frac{\chi}{Q} * \textit{(Dose Factors)} * \textit{(Curies Released)}$$

The control room whole-body dose can be found by use of the following formulas:

$$Dose_{CR\text{-}WB}\,(Rem) = \left(\frac{DCF_{Gamma}}{GF}\right) * A_{CR} * \frac{Occ}{V_{CR}} * \delta t$$

$$Where:\ GF = \frac{1173}{V^{0.338}} = \frac{1173}{(40{,}000)^{0.338}} = 32.643877 \quad See\ ref.\ 32$$

Occ = Control Room Occupancy

V_{CR} = Volume of Control Room

A_{CR} = Average Activity in Control Room

$$Dose_{CR\text{-}Skin}\,(Rem) = \left(\frac{DCF_{Gamma}}{GF} + DCF_{Beta}\right) * A_{CR} * \frac{Occ}{V_{CR}} * \delta t$$

$$Dose_{EAB\text{-}WB}\,(Rem) = \frac{\chi}{Q} * (DCF_{Gamma}) * (Curies\ Released)$$

$$Dose_{EAB\text{-}Skin}\,(Rem) = \frac{\chi}{Q} * (DCF_{Beta}) * (Curies\ Released)$$

The original ANO Fuel Handling Accident dose analysis used dose conversion factors obtained from Regulatory Guide 1.109 (Reference 15). The revised analysis will use the dose conversion factors given in the Tact5 User's Manual (Reference 9). This is consistent with ICRP 30 and the dose conversion factors used in the maximum hypothetical accident (Reference 16).

The ICRP 30 whole body gamma and beta dose conversion factors from TACT5 are as follows:

TACT5 (ICRP 30) Dose Conversion Factors

Isotope	Dose Conversion Factors		
	Whole Body Gamma Rem-m^3/Ci-sec	Skin Beta Rem-m^3/Ci-sec	Skin Gamma Rem-m^3/Ci-sec
Kr-83m	1.27e-05	0.00e+00	1.36e-04
Kr-85m	2.31e-02	4.97e-02	3.20e-02
Kr-85	3.31e-04	4.84e-02	4.75e-04
Kr-87	1.33e-01	3.36e-01	1.85e-01
Kr-88	3.38e-01	7.76e-02	4.69e-01
Kr-89	3.03e-01	3.47e-01	4.21e-01
Xe-131m	1.25e-03	1.33e-02	2.71e-03
Xe-133m	4.29e-03	2.96e-02	7.00e-03
Xe-133	4.96e-03	9.67e-03	7.89e-03
Xe-135m	6.37e-02	2.14e-02	9.16e-02
Xe-135	3.59e-02	6.32e-02	5.07e-02
Xe-137	2.83e-02	4.59e-01	4.02e-02
Xe-138	1.87e-01	1.47e-01	2.61e-01
I-130			
I-131	5.59e-02	3.07e-02	7.95e-02
I-132	3.55e-01	1.10e-01	5.07e-01
I-133	9.11e-02	8.90e-02	1.31e-01
I-134	4.11e-01	1.42e-01	5.86e-01
I-135	2.49e-01	7.86e-02	3.52e-01

6.6 Atmospheric Dispersion Factors

Standard Review Plan 2.3.4 (Reference 8) states that the atmospheric dispersion factor (i.e., χ/Q) can be determined on a probabilistic basis using the χ/Q value which will not be exceeded 95% of the time. Since the fuel handling accident assumes the release of all fission products to the environment over a two hour period, the two hour χ/Q values are used in this analysis. The two hour χ/Q values are as follows:

Control Room (0-2 hr)	1.20×10^{-3} sec/m^3	[Reference 38]
Exclusion Area Boundary ,EAB, (0-2 hr)	6.5×10^{-4} sec/m^3	(Unit 2 FSAR, Table 15.1.0-5)

Since the offsite dose limits of 75 Rem thyroid and 6 Rem whole-body are to be applied to both the exclusion area boundary and the low population zone, the offsite dose consequences of a fuel handling accident are most limiting at the exclusion area boundary due to the higher χ/Q value. On this basis, this calculation will consider only the dose consequences for the control room and the EAB.

6.7 Data

In the event that radioactivity is detected in the Unit 1 control room and/or the Unit 2 control room normal ventilation intake (Ref. 4), both Unit 1 and Unit 2 emergency ventilation systems are initiated. One of the two systems is subsequently shutdown as one is sufficient to maintain habitability. Although both systems meet the design intent for control room habitability, the Unit 1 ventilation system is limiting since it has a 2 inch recirculation charcoal bed compared to a 4 inch bed on the Unit 2 system. Therefore, in the control room analysis, the Unit 1 system was assumed to be operating and per the guidelines of Regulatory Guide 1.52 a 95% recirculation filter efficiency was used. Control room doses resulting from the FHA assumes that the control room recirculation charcoal absorber filter efficiency is 95% based on the limiting unit, Unit 1, which has a 2 inch charcoal absorber.

The data used in the evaluation of control room doses is given below:

Parameter	Input	Reference
Control Room Volume	4.00×10^4 ft^3	88E-0130-01
CR unfiltered inleakage	10 cfm	Regulatory Guide 1.78
CR filtered inleakage	333 cfm	2104.007
CR out leakage	343 cfm [333 + 10 cfm]	N/A
CR recirculation	1667 cfm	2104.007
CR occupancy factor		
0-24 hrs.	1.0	Murphy and Campe (Ref. 32)
Breathing Rate (CR)	3.47×10^{-4}	

For Cases 2a, 3a, and 4a, the CR unfiltered inleakage was increased to 61 cfm and the outleakage was increased to 394 cfm [333 + 61].

Control Room χ/Q		
0-2 hrs.	1.20×10^{-3}	Reference 38
CR filter efficiency		
recirc (2" filter)	95%	Regulatory Guide 1.52
intake (2-2" filters)	99%	Regulatory Guide 1.52

Additional FHA Input Data

Parameter	Input	Reference
Dose evaluation points	EAB and Control Room	N/A
Power level	3087 MWt	
Fuel Release Fraction		
Noble Gases (except Kr-85)	10%	Regulatory Guide 1.25
Kr-85	30%	Regulatory Guide 1.25
Halogens	13.5%	NUREG/CR 5009
Plate out	0%	not considered
Iodine Form		
Inorganic	99.75%	Regulatory Guide 1.25
Organic	0.25% "	Regulatory Guide 1.25
Offsite χ/Q		
EAB 0-2 hrs.	6.5×10^{-4}	FSAR Table 15.1.0-5
Breathing Rate (offsite)		Regulatory Guide 1.4
0-8 hrs.	3.47×10^{-4}	
Pool Decontamination Factors		
Inorganic Iodine	133	Regulatory Guide 1.25
Organic Iodine	1	Regulatory Guide 1.25
Fuel Handling Building (or Containment) filter iodine removal efficiencies (2 inch charcoal)		
Inorganic	90 %	(Regulatory Guide 1.25)
Organic	70 %	

7 CALCULATIONS

The fraction of activity remaining as a function of time after shutdown (relative to activity at shutdown) is shown below. As seen, the only significant isotopes remaining at the minimum time after shutdown (100 hrs) at which fuel movement could occur are I-131, I-133, Kr-85, Xe-131m, Xe-133m, and Xe-133.

Table 5.

Isotope	Half-life[3]	Decay Constant (hr^{-1})	Days After Shutdown			
			3	6	9	12
I-130	12.4 hr	5.5899e-02	1.79e-02	3.19e-04	5.70e-06	1.02e-07
I-131	***8.065 d***	***3.5810e-03***	***7.73e-01***	***5.97e-01***	***4.61e-01***	***3.57e-01***
I-132	2.284 hr	3.0348e-01	3.24e-10	1.05e-19	3.40e-29	1.10e-38
I-133	***20.8 hr***	***3.3324e-02***	***9.08e-02***	***8.24e-03***	***7.48e-04***	***6.79e-05***
I-134	52.3 m	7.9520e-01	1.36e-25	1.86e-50	2.54e-75	3.46e-100
I-135	6.7 hr	1.0345e-01	5.82e-04	3.39e-07	1.97e-10	1.15e-13
Kr-83m	1.86 hr	3.7266e-01	2.22e-12	4.95e-24	1.10e-35	2.45e-47
Kr-85m	4.4 hr	1.5753e-01	1.19e-05	1.41e-10	1.67e-15	1.98e-20
Kr-85	***10.74 yr***	***7.3674e-06***	***9.99e-01***	***9.99e-01***	***9.98e-01***	***9.98e-01***
Kr-87	76 m	5.4722e-01	7.74e-18	5.99e-35	4.64e-52	3.59e-69
Kr-88	2.79 hr	2.4844e-01	1.70e-08	2.90e-16	4.95e-24	8.43e-32
Kr-89	3.18 m	1.3078e+01	0.00e+00	0.00e+00	0.00e+00	0.00e+00
Xe-131m	***11.96 d***	***2.4148e-03***	***8.40e-01***	***7.06e-01***	***5.94e-01***	***4.99e-01***
Xe-133m	***2.26 d***	***1.2779e-02***	***3.98e-01***	***1.59e-01***	***6.33e-02***	***2.52e-02***
Xe-133	***5.27 d***	***5.4803e-03***	***6.74e-01***	***4.54e-01***	***3.06e-01***	***2.06e-01***
Xe-135m	15.7 m	2.6490e+00	1.48e-83	2.18e-166	3.21e-249	0.00e+00
Xe-135	9.16 hr	7.5671e-02	4.30e-03	1.85e-05	7.97e-08	3.43e-10
Xe-137	3.82 m	1.0887e+01	0.00e+00	0.00e+00	0.00e+00	0.00e+00
Xe-138	14.2 m	2.9288e+00	2.62e-92	6.89e-184	1.81e-275	0.00e+00

As shown above, the only significant isotopes that need to be considered at 100 hrs after shutdown are I-131, I-133, Kr-85, Xe-131m, Xe-133m, and Xe-133.

[3] Data from "Chart of the Nuclides", Tenth Edition, (Reference 12)

7.1 Calculation Procedure

The fuel gap activity has been determined for the CY-16 fuel [Ref. 34]. From this data, the activity released to the environment is determined based on the isotopic form in the fuel, the isotopic pool decontamination factors and building filter efficiencies as applicable. The offsite and control room doses are calculated using the TRANSACT computer code. TRANSACT (Reference 9 and 33) is a Fortran computer code designed to model the transport of radionuclides through various compartments, the release to the environment, and the resulting onsite and offsite doses.

7.2 Radionuclide Source Terms

Fuel Gap activity is determined as follows:

$A_{gap,i} = A_{fuel,i} * 13.5\%$ For Iodines
$A_{gap,i} = A_{fuel,i} * 10\%$ For Noble Gases except Kr 85
$A_{gap,i} = A_{fuel,i} * 30\%$ For Kr 85

Since the Technical Specifications restrict fuel handling operations until 100 hours after shutdown, the only significant isotopes which will be considered are Kr-85, I-131, I-133, Xe-131m, XE-133m and Xe-133.

Table 6.
Fuel Gap Activity at 100 hours Delay

Isotope	Fuel Activity[6] at t=100 hr (Ci)	Gap Activity at 100 hrs (Ci)
Kr-85	1.076E+04	3.228E+03
Xe-131m	9.543E+03	9.543E+02
Xe-133m	2.121E+04	2.121E+03
Xe-133	1.088E+06	1.088E+05
I-131	6.117E+05	8.258E+04
I-133	5.856E+04	7.906E+03

[6]Reference 34

The gap activity for 60 failed fuel rods is obtained as a ratio of the gap activity for 236 rods.

$$\text{Gap activity (60 rods)} = \text{Gap Activity (236 rods)} * \frac{60}{236}$$

Table 7.
Fuel Gap Activity
60 Rods, 100 hour Delay

Isotope	Fuel Activity[6] at t=100 hr for 236 rods (Ci)	Gap Activity at 100 hrs for 236 rods (Ci)	Gap Activity at 100 hrs for 60 rods (Ci)
Kr-85	1.076E+04	3.228E+03	8.207E+02
Xe-131m	9.543E+03	9.543E+02	2.426E+02
Xe-133m	2.121E+04	2.121E+03	5.392E+02
Xe-133	1.088E+06	1.088E+05	2.766E+04
I-131	6.117E+05	8.258E+04	2.099E+04
I-133	5.856E+04	7.906E+03	2.010E+03

Releases from the pool are given by:

$$R_{pool,i} = \frac{A_{gap,i}\, D_{gap,i}}{DF_{pool,i}}$$

Where:

$R_{pool,i}$ = Activity of isotope i released from pool (Ci)

$A_{gap,i}$ = Activity of isotope i in gap(Ci)

$D_{gap,i}$ = Distribution of isotope i (or isotopic form) in gap

The activity released from the pool is used as input to the TRANSACT Computer code. This activity is determined as follows:

$$A(released) = \frac{\text{Gap Activity}}{\text{Pool DF}} * Species\,Fraction$$

For TRANSACT input, the inorganic I_{131} and I_{133} activity released is :

$$A_{inorganic}(released) = \frac{\text{Gap Activity of Iodine}}{133} * 0.9975 = \text{Gap Activity of Iodine} * 7.5 * 10^{-3}$$

Where the inorganic species fraction is 0.9975

For organic Iodines, the TRANSACT input iodine species fraction is determined as follows :

$$A_{organic}(released) = \frac{\text{Gap Activity of Iodine}}{\text{Pool DF}} * 0.0025 = \frac{\text{Gap Activity of Iodine}}{1} * 0.0025 = \text{Gap Activity of Iodine} * 0.0025$$

Where the organic species fracion is 0.0025

For noble gases, all activity in the fuel-cladding gap is released from the pool.

7.3 FHA inside Containment or Spent Fuel Building Without Filtration (Case 1)

This case evaluates the consequences of the limiting fuel handling accident; failure of an assembly (236 rods, 100 hrs. decay) in either the fuel handling building or the containment without building integrity. A puff (2 hour) release is assumed with equipment hatch open in the spent fuel building or the containment equipment hatch open. This analysis will be based on typical licensing basis assumptions for EAB and Control Room dose consequences. Since this case is based on a puff release and typical regulatory guide assumptions, the results are applicable to a fuel handling accident either inside the spent fuel building or the containment. Input and output files for this case are given in Attachment 1.

7.4 FHA inside Containment or Spent Fuel Building With Filtration (Cases 2 and 2a)

This case evaluates the release of radioisotopes following a fuel handling accident with the equipment hatch open. This is the same basic scenario as Case 1 except that credit for fuel pool ventilation and filtration system or the containment purge system, is utilized. The radioisotopes released from the fuel gap of the damaged fuel pass through the pool water and are then released at the pool surface. At this point, the pool sweep system directs the airflow, and the radioisotopes released from the pool, to the fuel handling area ventilation system. This system is designed to remove radioiodine from the airflow to minimize the offsite dose consequences of a fuel handling accident. The fuel handling area ventilation system is designed to meet the requirements of Regulatory Guide 1.52. An iodine decontamination efficiency of 90% for elemental iodine and 70% is assumed based on the guidance in Regulatory Guide 1.25.

In the spent fuel building, the pool sweep system filters any activity released from the damaged fuel as it evolves from the pool. The fuel handling area ventilation system is required to be OPERATIONAL

whenever irradiated fuel is being moved in the storage pool and during crane operation with loads over the storage pool (Reference Technical specification 3.9.11). Activity removed by the pool sweep system is directed to the filtration system before it enters the atmosphere.

Inclusion of credit for the filtration system only affects the EAB and Control Room thyroid doses since the filters are ineffective in removal of noble gases. Due to the design and operation of the pool sweep system, no credit for mixing within the fuel handling building is considered. Input and output files for this case are given in Attachment 2. The input and output files for case 2a (61cfm inleakage) are given in Attachment 6.

7.5 FHA With 60 Rods Failed With Filtration (Cases 3 and 3a)

This case evaluates the offsite and onsite dose consequences of a fuel handling accident in which 60 fuel rods are damaged. The ANO Unit 2 Upper Level Document, ULD-2-TOP-02, Revision 0 (Reference 25) states that the number of fuel rods damaged during a fuel handling accident is 60 (4 layers of fuel rods) of the 236 rods in an assembly. The number of rods damaged is also given in FSAR Section 15.1.23 (Amendment 12).

The FHA is assumed to occur in the fuel handling building or the containment after 100 hours decay. Due to the design and operation of the pool sweep system, no credit for mixing within the fuel handling building is considered. Since the charcoal adsorber iodine removal efficiency for the fuel handling building and containment building filters are the same, a FHA in the fuel building will be limiting relative to release rate (no holdup due to mixing) and will be evaluated here. In the spent fuel building, the pool sweep system filters any activity released from the damaged fuel as it evolves from the pool. The fuel handling area ventilation system is required to be OPERATIONAL whenever irradiated fuel is being moved in the storage pool and during crane operation with loads over the storage pool (Reference Technical Specification 3.9.11). Activity removed by the pool sweep system is directed to the filtration system before it enters the atmosphere.

The activity released from the pool at 100 hrs for 60 failed fuel rods used as input for the TRANSACT computer code. Input and output files for this case are given in Attachment 3. The input and output files for case 3a (61cfm inleakage) are given in Attachment 7.

7.6 FHA With 60 Rods Failed Without Filtration (Case 4 and 4a)

This case evaluates the offsite and onsite dose consequences of a fuel handling accident in which 60 fuel rods are damaged. The input for this case is the same as for case 3 except that credit for the fuel building ventilation system is not taken. The activity released from the pool escapes directly into the environment without filtration. Input and output files for this case are given in Attachment 4. The input and output files for case 4a (61cfm inleakage) are given in Attachment 8.

The input data file used for all cases is given in Attachment 5.

Attachment 4

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3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems - Shutdown

LCO 3.8.10 The necessary portion of AC, DC, and 120 VAC vital bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE by the following specifications:

LCO 3.3.9, "Source Range Neutron Flux,"
LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits,"
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled,"
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled,"
LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP) System,"
LCO 3.7.9, "Control Room Emergency Ventilation System (CREVS),"
LCO 3.7.10, "Control Room Emergency Air Conditioning System (CREACS),"
~~LCO 3.7.12, "Fuel Handling Area Ventilation System (FHAVS),"~~ |
LCO 3.9.2, "Nuclear Instrumentation," for one monitor,
LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation - High Water Level," and
LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

--NOTE--
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC, DC, or 120 VAC vital bus electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable. OR A.2.1 Suspend CORE ALTERATIONS. AND	Immediately Immediately