Entergy Operations, Inc.
1448 S.R. 333
Russellville, AR 72802
Tel 479-858-3110

Timothy G. Mitchell
Vice President, Operations
Arkansas Nuclear One

1CAN080801

August 14, 2008

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

SUBJECT: Response to Request for Additional Information
Regarding Technical Specification Changes and Analyses Relating to Use of Alternate Source Term
Arkansas Nuclear One, Unit 1
Docket No. 50-313
License No. DPR-51

REFERENCE:

1. Entergy letter dated October 22, 2007, "License Amendment Request: Technical Specification Changes and Analyses Relating to Use of Alternate Source Term" (1CAN100703) (TAC NO: MD7178)
2. Entergy letter dated March 13, 2008, "Supplement to Amendment Request: Technical Specification Changes and Analyses Relating to Use of Alternate Source Term" (1CAN030803) (TAC NO: MD7178)
3. Entergy letter dated April 3, 2008, "Supplement to Amendment Request: Technical Specification Changes and Analyses Relating to Use of Alternate Source Term" (1CAN040802) (TAC NO: MD7178)
4. NRC letter dated August 13, 2008, "Arkansas Nuclear One, Unit 1 – Re: Request for Additional Information Regarding Application of the Alternative Source Term (TAC No. MD7178)

Dear Sir or Madam:

By letter (Reference 1), Entergy Operations, Inc. (Entergy) proposed a change to the Arkansas Nuclear One, Unit 1 (ANO-1) Technical Specifications (TSs) to support adoption and use of Alternate Source Term (AST) in the ANO-1 Safety Analyses.

On July 16, 2008, Entergy was notified of a request for additional information (RAI) with regard to the subject letter (Reference 1). A conference call was held with the NRC on July 28, 2008, to ensure clear understanding of the additional information being requested. The formal RAI was received by Entergy on August 13, 2008 via fax (Reference 4). Attachment 1 contains the Entergy responses to the RAI.

Attachment 2 contains updated tables. All tables submitted in the Reference 3 letter above (beginning on Page 47 of 67 of Attachment 2 of the letter) are re-submitted, although two of the tables (Tables 2.1-2 and 3-1), did not require updating in response to this RAI.

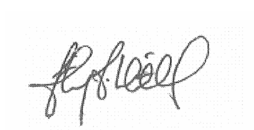
Attachment 3 provides a mark-up of the ANO-1 Safety Analysis Report (SAR). Again, the complete set of markups is re-submitted although changes were only made to pages associated with Sections 14.2 and 14.5 of the SAR in response to this RAI.

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.

If you have any questions or require additional information, please contact Dale James at 479-858-4619.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 14, 2008.

Sincerely,



TGM/dbb

Attachments:
1. Response to Request for Additional Information Regarding Technical Specification Changes and Analyses Relating to Use of Alternate Source Term
2. Updated Analysis Input Tables
3. Updated Markup of ANO-1 Safety Analysis Report

Enclosure: CD Rom containing data files

cc: Mr. Elmo E. Collins
Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

NRC Senior Resident Inspector
Arkansas Nuclear One
P. O. Box 310
London, AR 72847

U. S. Nuclear Regulatory Commission
Attn: Mr. Alan B. Wang
MS O-7 D1
Washington, DC 20555-0001

Mr. Bernard R. Bevill
Director Division of Radiation
Control and Emergency Management
Arkansas Department of Health & Human Services
P.O. Box 1437
Slot H-30
Little Rock, AR 72203-1437

Attachment 1

To

1CAN080801

Response to Request for Additional Information
Regarding Technical Specification Changes and Analyses Relating to Use of Alternate Source Term

Response to Request for Additional Information Regarding Technical Specification Changes and Analyses Relating to Use of Alternate Source Term

The U.S. Nuclear Regulatory Commission (NRC) staff determined that additional information is needed to complete its review of the license amendment request for Alternate Source Term (AST) for Arkansas Nuclear One, Unit 1 (ANO-1), sent by letter dated October 22, 2007, as supplemented by letter dated April 3, 2008 from Entergy Operations Inc. (Entergy or licensee). The following questions constitute the NRC request for additional information for the dose consequence portion of the review.

A. INTEGRITY OF FACILITY DESIGN-BASIS

1. *Paragraph 50.67(b) of Title 10 of the Code of Federal Regulations (10 CFR) Part 50.67, "Accident Source Term", requires that applications under this section contain an evaluation of the consequences of applicable design-basis accidents (DBAs) previously analyzed in the plant's safety analysis report (SAR). Also, Regulatory Guide (RG) 1.183 provides guidance to licensees of operating power reactors on acceptable applications of alternative source terms (AST); the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. Regulatory position 1.1.3, "Integrity of Facility Design-basis" of RG 1.183 states in principal that a complete re-assessment of all facility radiological analyses would be desirable.*

   For the dose consequence DBAs that have been evaluated for the proposed AST amendment, provide the basic parameters used in the analyses. For each parameter, indicate the current licensing basis (CLB) value, the revised value where applicable, as well as the basis for any changes to the CLB. The NRC staff requests that the licensee expand the information in the Attachment 3 tables of the ANO-1 amendment request to include CLB parameters whether or not the individual parameter has changed for this amendment (See regulatory position 1.3.2 and 1.3.4 of RG 1.183 and NRC RIS 2006-04).

   Note: This question is repeated from the February 27, 2008 RAI (A.2.b). The licensee in its April 3, 2008, supplement stated, "A response did not appear to add significant benefit since each [AST licensing basis] value illustrated would require a discussion of its origins." The NRC staff believes that ANO-1's response was insufficient for the NRC staff to complete its review. Since the proposed design-basis parameters conformance with NRC regulations and regulatory guidance will form the basis for NRC staff approval or denial of a licensee's AST, a discussion of the origins of these major parameters is essential for the NRC staff to evaluate the major design-basis assumptions of an applicant's AST implementation.

   Response:

   The analysis input tables included in Attachment 2 (beginning on page 47 of 67) of the Entergy Operations, Inc. supplemental letter dated March 13, 2008 (1CAN030803), have been expanded and are presented in Attachment 2. These tables now include the bases for the input values chosen for each AST analysis and demonstrate that the postulated dose consequences of the analyses are conservative, consistent with the guidance of RG 1.183, Regulatory Position 5.1.3. Note that Tables 2.1-2 and 3-1 are not affected by this RAI, but included are in Attachment 2 for completeness.

B. ACCIDENT SOURCE TERM

2. *Please describe in detail why ANO-1 did not utilize the non-LOCA gap fractions as outlined in Table 3 of RG 1.183 for the fuel handling accident (FHA). In the supplement dated April 3, 2008, the licensee stated that RG 1.183 Table 3 was not "completely utilized." Instead, the licensee used RG 1.25 gap fractions for the FHA. In general, using a regulatory guidance other than RG 1.183 for AST applications is not acceptable to the NRC staff unless previous precedent can be cited or the licensee can show adequate detailed justification. Therefore, the NRC staff requests that the licensee cite precedent and include a justification for why that precedent is applicable to ANO-1, provide a detailed technical justification for using the RG 1.25 gap fractions in lieu of the RG 1.183 guidance, or revise its FHA dose consequence analyses employing the gap fractions in RG 1.183.*

   Table 4 of regulatory position 3.3 describes acceptable release phases for application of an AST for a loss of coolant accident (LOCA). For pressurized water reactors (PWRs) like ANO-1, the early in-vessel duration is 0.5-1.3 hr. In Table 2.1-1 of its April 3, 2008 supplement, ANO-1 listed the Early In-Vessel Release Phase as 0.5-1.8 hrs. Based on the NRC review of the licensee's amendment request, the NRC staff could not ascertain the basis for this deviation from the regulatory guidance. Therefore, the NRC staff requests that the licensee provide its detailed justification for deviation from the prescribed RG 1.183 Table 4 release phase value. As specified in regulatory position 3.3, a licensee may propose an alternative time for the onset of the gap release phase; however, in the absence of approved alternatives, the gap release phase onsets in Table 4 of RG 1.183 should be used.

   Response:

   RG 1.183, Table 3, is believed to underestimate the gap fractions that will exist following a FHA and therefore the more conservative fractions of RG 1.25 were used. In addition, the gap fraction for I-131 was further increased by 20% per Table 3.6 of NUREG/CR-5009 to accommodate extended burnup fuel. The following table compares the RG 1.183 fractions with those used in the ANO-1 AST FHA analysis and demonstrates that the ANO-1 analysis is conservative. Although the gap fraction of the alkali metals used in the ANO-1 analysis does not bound the RG 1.183 value, all alkali metals are particulate radionuclides, which are retained by the water in the fuel pool or reactor cavity, consistent with the guidance of RG 1.183, Appendix B, Regulatory Position 3, and therefore, do not contribute to the offsite or control room dose consequences.

Group	RG 1.183 Fraction	ANO-1 Analysis Fraction
I-131	0.08	0.12
Kr-85	0.10	0.30
Other Noble Gases	0.05	0.10
Other Halogens	0.05	0.10
Alkali Metals	0.12	0.10

With respect to the LOCA early in-vessel release period, the ANO-1 AST LOCA analysis has been performed consistent with the guidance provided in Table 4 of RG 1.183, which states that this phase should be assumed to begin at 0.5 hrs following accident initiation and have a duration of 1.3 hrs, that is, occur over the period of 0.5 to 1.8 hrs following the initiation of the LOCA.

C. DOSE CACULATIONAL METHODOLOGY

3. *The ANO-1 SAR section 14.2.2.6, "Maximum Hypothetical Accident" states that: "In order to demonstrate that the operation of ANO-1 does not produce undue risk to the public under any accident conditions, the dose that would be received at the exclusion distance and the low population zone from a release of radioactivity larger than any which could actually occur is calculated."*

This ANO-1 SAR description conforms to the postulated accident described in footnote 1 of 10 CFR 50.67, "Accident Source Term", which states, "The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products."

The worst credible accident for dose consequence purposes is not necessarily defined as the design-basis accident for meeting 10 CFR 50.46 criteria for exclusion area boundary systems. For dose consequence purposes, the deterministic source term is postulated from the worst credible event used to test a facility's engineered safety features and are intentionally conservative to compensate for uncertainties in accident progression, fission product transport, and atmospheric dispersion.

Please describe the ANO-1 accident that is considered the Maximum Hypothetical Accident for the AST analysis including the evaluation and any associated reanalysis that was done in order to ensure that ANO-1 has identified the most limiting accident under AST conditions. Also, since ANO-1 is a member of the PWR Owners Group that has requested formal review of BAW-2374 Revision 2 topical report, the NRC staff requests that ANO-1 clearly describe its AST analysis of the leakage pathway through failed steam generator tubes during a worst case dose consequence accident. If ANO-1 is susceptible to a consequential failure of the steam generator tubes during a large break LOCA, the NRC staff requests that the licensee describe how it meets the 10 CFR 50.67 regulatory criteria for protecting public health and safety during this credible design-basis accident. Furthermore, the NRC staff requests that the licensee describe the changes being made to its design and licensing basis and Technical Specifications as a result of this analysis.

Response:

The current ANO-1 Maximum Hypothetical Accident (MHA) analysis reported in the SAR is a LOCA that assumes the TID-14844 source term. The ANO-1 AST LOCA analysis is proposed to replace this analysis. The current ANO-1 LOCA analysis reported in the SAR assumes more realistic conditions than the MHA analysis, including a significantly lower fission product inventory, and is therefore bounded by the current ANO-1 MHA analysis.

Due to the bounding nature of the RG 1.183 LOCA assumptions, a separate "realistic" LOCA analysis using the alternate source terms would not provide any meaningful insights and thus is considered unnecessary. Therefore, Entergy proposes to replace the separate MHA and realistic LOCA dose analyses currently in the ANO-1 SAR with a single, bounding AST LOCA analysis. As shown in Table 3-1 of the March 13, 2008 supplement (1CAN030803), the worst credible ANO-1 AST accidents for dose consequence purposes are the LOCA for absolute Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) doses and the Control Rod Ejection Accident (CREA) (secondary release case) for control room dose. With AST, ANO-1 is proposing to delete use of the antiquated "Maximum Hypothetical Accident" terminology in the interest of maintaining a clear design basis per the guidance of RG 1.183, Regulatory Position 1.6.

Note the intent of any analysis is to ensure the "most limiting" accidents or conditions are used to verify accident consequences remain acceptable. The ANO AST submittal and proposed SAR changes have appropriately identified these most limiting accidents and use of the antiquated MHA term is, therefore, being retired.

The analyses on which BAW-2374 is based show that a LOCA will not result in any steam generator (SG) tube failures, unless the LOCA occurs at or near the top of the Reactor Coolant System (RCS) hot leg piping. However, the conclusion of the cladding rupture study documented in Appendix F of BAW-2374, Revision 2, is that the cladding integrity will not be compromised during the blowdown, refill, or reflood phase of the hot leg break event that produces the most limiting SG tube-to-shell temperature differential. Without a breach of the fuel cladding, no large source term exists and any radiation release would have to be due to the radioactive products contained in the reactor coolant, not the fissionable material or gap activity within the fuel itself. Therefore, the existence of an AST LOCA source term for a BAW-2374 event that results in SG tube failure is not credible and the bounding AST SG Tube Rupture (SGTR) source term should be considered applicable to the scenario.

Due to the small differential pressure following the LOCA, the leak rate through any failed tubes would be insignificant compared to the leak rate assumed in the ANO-1 AST SGTR analysis on which the ANO-1 submittal is based. In addition, unlike a SGTR, the BAW-2374 scenario does not produce the secondary pressure necessary to lift the Main Steam Safety Valves (MSSVs) or require use of the SGs to cooldown; therefore, there is no direct release to the atmosphere. Based on this information, the results of the ANO-1 AST LOCA and SGTR analyses, as shown in Table 3-1 of the March 13, 2008 supplement , are bounding for the BAW-2374 scenario and, since the results meet the applicable acceptance criteria, the analyses demonstrate that the 10 CFR 50.67 regulatory criteria for protecting public health and safety would be met during a BAW-2374 accident scenario.

It should be noted that the BAW-2374 scenario in question is actually an over-cooling event in which cold, maximum Emergency Core Cooling System (ECCS) flows are assumed in order to maximize thermal stresses in the SG tubes; the scenario is not considered the classic LOCA analysis that results in a more significant heatup and oxidation of the core. As such, the BAW-2374 scenario does not establish any design basis parameter, such as for ECCS performance or containment structural design, but rather demonstrates that the current plant design bases remain bounding. In summary, Entergy does not consider the BAW-2374 scenario to be a DBA LOCA. No changes to the ANO-1 current license basis (CLB) or Technical Specifications are currently planned as a result of the BAW-2374 analysis, pending final NRC review and approval of the document.

D. ADDITIONAL INFORMATION

4. *In addition to the NRC staff questions given above, provide the following. Please summarize all the new or revised CLB AST parameters and analysis methods including the control room dose consequence results for your analyzed dose consequence accidents. Regulatory position 1.6 of RG 1.183 outlines the FSAR update requirements including a reference to 10 CFR 50.71.*

Response:

The SAR markups provided in the March 13, 2008 (1CAN030803), supplement have been modified, as requested, and are included in Attachment 3. All markups are included for completeness. For ease in NRC review, the following SAR markup pages were revised with respect to those submitted in the March 13, 2008 supplement:

14.2-17	14.5-13
14.2-56	14.5-16
14.5-11	14.5-28
14.5-12	

Note that the SAR markup previously submitted includes discussions of primary and secondary activity, 10 CFR 50.67, RG 1.183, leak rates, event durations, iodine assumptions, source terms, Total Effective Dose Equivalent (TEDE), the computer models used, conversion factors, flow rates, and dose consequence results. However, the CEA ejection accident does not list control room dose consequences for which it is the most limiting accident; therefore, this information is added to the SAR markup.

Attachment 2

To

1CAN080801

Updated Analysis Input Tables

Updated Analysis Input Tables

Acronyms

AOR	Analysis of Record
AST	Alternate Source Term Analyses
CLB	Current Licensing Basis
CR	Control Room
CREA	Control Rod Ejection Accident
FHA	Fuel Handling Accident
LOCA	Loss of Coolant Accident
MSLB	Main Steam Line Break
RG	Regulatory Guide
RP	Regulatory Position
SGTR	Steam Generator Tube Rupture
TS	Technical Specification

REFERENCES:

(1) NRC Letter of May 6, 2004, "Final Safety Evaluation for Topical Report WCAP-16072-P, Revision 00, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs" TAC NO MB8721."

(2) Westinghouse Calculation CN-CRA-05-62, Rev. 0, "Ringhals Unit 3 LOCA Doses for Uprate Project (GREAT) – Conservative Case", U. Bachrach, January 31, 2006.

Table 1.6.3-1

Control Room Ventilation System Parameters

(Conforms to RG 1.183, Main Section, RP 4.2.4)

Parameter	**Value**	**Basis**
Control Room Volume	40,000 ft^3	Conservative internal calculation; consistent with AOR
Normal Operation		
Filtered Make-up Flow Rate	0 cfm	Conservative assumption; Particulate filters exist
Filtered Recirculation Flow Rate	0 cfm	Conservative assumption; Particulate filters exist
Unfiltered Make-up Flow Rate	35,200 cfm	Total design flow rate for combined ANO-1 & 2 CR
Emergency Operation		
Filtered Make-up Flow Rate	333 cfm	Consistent with AOR and CLB
Filtered Recirculation Flow Rate	1667 cfm	Consistent with AOR and CLB
Unfiltered Make-up Flow Rate	0 cfm	System design does not support unfiltered makeup flow
Unfiltered Inleakage:		
- LOCA, CREA	82 cfm	Consistent with proposed new TS limit
- MSLB, SGTR, FHA	85 cfm	Bounds proposed new TS limit
Filter Efficiencies		
Make-up Flow (two 2” filters):		
- Particulate	99%	Conservative assumption bounded by current TS 5.5.11
- Elemental	99%	Conservative assumption bounded by current TS 5.5.11
- Organic	99%	Conservative assumption bounded by current TS 5.5.11
Recirculation Flow (one 2” filter):		
- Particulate	99%	Conservative assumption bounded by current TS 5.5.11
- Elemental	95%	Conservative assumption bounded by current TS 5.5.11
- Organic	95%	Conservative assumption bounded by current TS 5.5.11

Table 1.6.5-1

Control Room Doses Due to Containment Shine

(Conforms to RG 1.183, Main Section, RPs 4.2.2 and 4.2.3)

Time After Accident (hrs)	Gamma Ray Dose Rate (Rem/hr)	Integrated Dose (Rem)	Time After Accident (hrs)	Gamma Ray Dose Rate (Rem/hr)	Integrated Dose (Rem)
0.0083	3.96E-02	1.10E-05	**3.31**	1.03E-02	3.78E-02
0.1083	6.53E-03	1.76E-03	**3.81**	7.90E-03	4.21E-02
0.2083	3.89E-03	2.26E-03	**5.81**	4.69E-03	5.44E-02
0.31	3.27E-03	2.62E-03	**7.81**	2.84E-03	6.18E-02
0.41	3.06E-03	2.93E-03	**9.81**	1.73E-03	6.62E-02
0.51	2.97E-03	3.23E-03	**11.81**	1.06E-03	6.90E-02
0.83	8.19E-03	4.90E-03	**16.50**	1.01E-03	7.38E-02
1.16	1.22E-02	8.17E-03	**24.00**	3.33E-04	7.83E-02
1.48	1.51E-02	1.26E-02	**64.00**	2.17E-07	8.00E-02
1.81	1.72E-02	1.78E-02	**72.00**	2.15E-07	8.00E-02
2.06	1.59E-02	2.20E-02	**100.00**	2.18E-07	8.00E-02
2.07	1.57E-02	2.21E-02	**168.00**	2.03E-07	8.00E-02
2.31	1.42E-02	2.57E-02	**720.00**	6.05E-08	8.01E-02
2.81	1.20E-02	3.23E-02			

Table 1.6.5-2

Control Room Attenuation Factors

(Conforms to RG 1.183, Main Section, RP 4.2.3)

Isotope	Attenuation Factor	Isotope	Attenuation Factor	Isotope	Attenuation Factor
Kr-83m	0.00E+00	**Sb-127**	2.47E-04	**Ce-144**	2.14E-06
Kr-85	1.89E-04	**Sb-129**	9.34E-04	**Np-239**	1.16E-05
Kr-85m	4.04E-05	**Te-127**	7.74E-05	**Pu-238**	1.19E-08
Kr-87	2.35E-03	**Te-127m**	5.92E-09	**Pu-239**	2.53E-07
Kr-88	3.57E-03	**Te-129**	1.85E-04	**Pu-240**	4.75E-08
Xe-131m	1.01E-06	**Te-129m**	8.82E-05	**Pu-241**	1.79E-03
Xe-133	6.33E-08	**Te-131m**	7.87E-04	**Am-241**	2.62E-08
Xe-133m	6.09E-06	**Te-132**	1.88E-05	**Cm-242**	7.33E-08
Xe-135	4.16E-05	**Sr-89**	5.25E-04	**Cm-244**	1.17E-06
Xe-135m	1.76E-04	**Sr-90**	0.00E+00	**La-140**	1.97E-03
Xe-138	2.49E-03	**Sr-91**	9.08E-04	**La-142**	3.41E-03
I-130	3.46E-04	**Sr-92**	1.28E-03	**Nb-95**	5.25E-04
I-131	1.00E-04	**Ba-139**	1.20E-04	**Nd-147**	7.69E-05
I-132	5.42E-04	**Ba-140**	1.14E-04	**Pr-143**	5.25E-04
I-133	3.16E-04	**Mo-99**	3.87E-04	**Y-90**	0.00E+00
I-134	7.87E-04	**Rh-105**	7.96E-05	**Y-91**	1.34E-03
I-135	1.64E-03	**Ru-103**	1.87E-04	**Y-92**	7.53E-04
Cs-134	3.91E-04	**Ru-105**	3.57E-04	**Y-93**	1.29E-03
Cs-136	8.12E-04	**Ru-106**	0.00E+00	**Zr-95**	5.25E-04
Cs-137	1.86E-04	**Tc-99m**	3.54E-06	**Zr-97**	1.01E-03
Cs-138	1.89E-03	**Ce-141**	3.21E-06		
Rb-86	1.34E-03	**Ce-143**	1.20E-04		

Table 1.7.2-1

Primary Coolant Source Term

(Conforms to RG 1.183, Appendix E, RP 2 and Appendix F, RP 2)

Nuclide	AST Nuclide Group	AST Activity (Ci)
Kr-85m	1 – Noble Gas	1.31E+03
Kr-85	1 – Noble Gas	4.80E+02
Kr-87	1 – Noble Gas	2.09E+03
Kr-88	1 – Noble Gas	2.93E+03
Xe-131m	1 – Noble Gas	3.78E+02
Xe-133m	1 – Noble Gas	7.64E+02
Xe-133	1 – Noble Gas	3.02E+04
Xe-135m	1 – Noble Gas	1.18E+03
Xe-135	1 – Noble Gas	1.30E+04
Xe-138	1 – Noble Gas	3.46E+03
I-130	2 – Iodine	6.82E+02
I-131	2 – Iodine	7.33E+01
I-132	2 – Iodine	1.00E+03
I-133	2 – Iodine	6.91E+02
I-134	2 – Iodine	1.48E+03
I-135	2 – Iodine	1.22E+03
Cs-134	3 – Alkali Metal	5.11E+02
Cs-136	3 – Alkali Metal	4.03E+01
Cs-137	3 – Alkali Metal	4.22E+02
Cs-138	3 – Alkali Metal	1.00E+04
Rb-86	3 – Alkali Metal	6.43E+01

Table 1.7.3-1

Secondary Side Source Term

(No RG 1.183 guidance; Values consistent with proposed new TS limit)

Nuclide	AST Nuclide Group	AST Activity (Ci)
I-130	2 – Iodine	1.29E+01
I-131	2 – Iodine	1.39E+00
I-132	2 – Iodine	1.90E+01
I-133	2 – Iodine	1.31E+01
I-134	2 – Iodine	2.81E+01
I-135	2 – Iodine	2.31E+01
Cs-134	3 – Alkali Metal	9.70E+00
Cs-136	3 – Alkali Metal	7.66E-01
Cs-137	3 – Alkali Metal	8.01E+00
Cs-138	3 – Alkali Metal	1.90E+02
Rb-86	3 – Alkali Metal	1.22E+00

Table 1.7.4-1

Core Isotopic Inventory for LOCA and CREA

(Conforms to RG 1.183, Main Section, RPs 3.1 and 3.4)

Isotope	AST Core Inventory [Curies]	Isotope	AST Core Inventory [Curies]	Isotope	AST Core Inventory [Curies]
Kr-83m	8.77E+06	**Sb-129**	2.01E+07	**Ce-141**	1.23E+08
Kr-85	9.61E+05	**Sb-131**	5.67E+07	**Ce-143**	1.12E+08
Kr-85m	1.90E+07	**Te-127**	6.52E+06	**Ce-144**	1.05E+08
Kr-87	3.73E+07	**Te-127m**	1.16E+06	**Np-239**	1.39E+09
Kr-88	5.01E+07	**Te-129**	1.88E+07	**Pu-238**	1.93E+05
Xe-131m	7.55E+05	**Te-129m**	3.66E+06	**Pu-239**	2.51E+04
Xe-133	1.48E+08	**Te-131**	6.10E+07	**Pu-240**	3.88E+04
Xe-133m	4.60E+06	**Te-131m**	1.40E+07	**Pu-241**	9.82E+06
Xe-135	3.51E+07	**Te-132**	1.02E+08	**Am-241**	1.02E+04
Xe-135m	3.09E+07	**Te-133**	7.89E+07	**Cm-242**	2.71E+06
Xe-138	1.27E+08	**Te-133m**	7.02E+07	**Cm-244**	1.99E+05
I-130	1.36E+06	**Sr-89**	7.25E+07	**La-140**	1.32E+08
I-131	7.22E+07	**Sr-90**	7.47E+06	**La-142**	1.15E+08
I-132	1.05E+08	**Sr-91**	8.78E+07	**Nb-95**	1.34E+08
I-133	1.48E+08	**Sr-92**	9.40E+07	**Nd-147**	4.70E+07
I-134	1.67E+08	**Ba-139**	1.32E+08	**Pr-143**	1.11E+08
I-135	1.41E+08	**Ba-140**	1.28E+08	**Y-90**	7.75E+06
Cs-134	1.46E+07	**Mo-99**	1.35E+08	**Y-91**	9.53E+07
Cs-136	2.98E+06	**Rh-105**	7.25E+07	**Y-92**	9.51E+07
Cs-137	9.88E+06	**Ru-103**	1.14E+08	**Y-93**	1.07E+08
Cs-138	1.38E+08	**Ru-105**	7.64E+07	**Zr-95**	1.29E+08
Rb-86	1.29E+05	**Ru-106**	4.19E+07	**Zr-97**	1.23E+08
Sb-127	6.56E+06	**Tc-99m**	1.18E+08		

Table 1.7.5-1

Fuel Handling Accident Source Term – 82 Damaged Rods

(Conforms to RG 1.183, Main Section, RPs 3.1 and 3.4)

Nuclide	RADTRAD Group	RADTRAD AST Core Activity (Ci)
Kr-85m	1 – Noble Gas	1.90E+07
Kr-85	1 – Noble Gas	9.61E+05
Kr-87	1 – Noble Gas	3.73E+07
Kr-88	1 – Noble Gas	5.01E+07
Xe-131m	1 – Noble Gas	7.55E+05
Xe-133m	1 – Noble Gas	4.60E+06
Xe-133	1 – Noble Gas	1.48E+08
Xe-135m	1 – Noble Gas	3.09E+07
Xe-135	1 – Noble Gas	3.51E+07
Xe-138	1 – Noble Gas	1.27E+08
I-130	2 – Iodine	1.36E+06
I-131	2 – Iodine	7.22E+07
I-132	2 – Iodine	1.05E+08
I-133	2 – Iodine	1.48E+08
I-134	2 – Iodine	1.67E+08
I-135	2 – Iodine	1.41E+08
Sb-131	4 – Tellurium Group	5.67E+07
Te-131	4 – Tellurium Group	6.10E+07
Te-131m	4 – Tellurium Group	1.40E+07
Te-132	4 – Tellurium Group	1.02E+08
Te-133	4 – Tellurium Group	7.89E+07
Te-133m	4 – Tellurium Group	7.02E+07

Table 1.8.1-1

Revised Onsite Atmospheric Dispersion Factors for ANO-1

(Conforms to RG 1.183, Main Section, RP 5.3)

Time Period	AST X/Q Value	Basis
ADV Releases to VPH-1		
0 to 2 hrs	1.89×10^{-3} sec/m^3	New ARCON96 analysis
2 to 8 hrs	1.39×10^{-3} sec/m^3	New ARCON96 analysis
8 to 24 hrs	6.00×10^{-4} sec/m^3	New ARCON96 analysis
1 to 4 days	4.13×10^{-4} sec/m^3	New ARCON96 analysis
4 to 30 days	3.28×10^{-4} sec/m^3	New ARCON96 analysis
ADV Releases to VPH-2		
0 to 2 hrs	4.10×10^{-3} sec/m^3	New ARCON96 analysis
2 to 8 hrs	2.59×10^{-3} sec/m^3	New ARCON96 analysis
8 to 24 hrs	1.12×10^{-3} sec/m^3	New ARCON96 analysis
1 to 4 days	8.32×10^{-4} sec/m^3	New ARCON96 analysis
4 to 30 days	5.91×10^{-4} sec/m^3	New ARCON96 analysis
MSSV Releases to VPH-1		
0 to 2 hrs	5.17×10^{-3} sec/m^3	New ARCON96 analysis
2 to 8 hrs	3.38×10^{-3} sec/m^3	New ARCON96 analysis
8 to 24 hrs	1.42×10^{-3} sec/m^3	New ARCON96 analysis
1 to 4 days	1.07×10^{-3} sec/m^3	New ARCON96 analysis
4 to 30 days	7.58×10^{-4} sec/m^3	New ARCON96 analysis
MSSV Releases to VPH-2		
0 to 2 hrs	1.90×10^{-2} sec/m^3	New ARCON96 analysis
2 to 8 hrs	1.23×10^{-2} sec/m^3	New ARCON96 analysis
8 to 24 hrs	5.83×10^{-3} sec/m^3	New ARCON96 analysis
1 to 4 days	3.80×10^{-3} sec/m^3	New ARCON96 analysis
4 to 30 days	3.10×10^{-3} sec/m^3	New ARCON96 analysis
Main Steam Pipe Release to VPH-1		
0 to 2 hrs	1.75×10^{-3} sec/m^3	New ARCON96 analysis
2 to 8 hrs	1.25×10^{-3} sec/m^3	New ARCON96 analysis
8 to 24 hrs	5.49×10^{-4} sec/m^3	New ARCON96 analysis
1 to 4 days	3.90×10^{-4} sec/m^3	New ARCON96 analysis
4 to 30 days	3.05×10^{-4} sec/m^3	New ARCON96 analysis
Main Steam Pipe Release to VPH-2		
0 to 2 hrs	3.15×10^{-3} sec/m^3	New ARCON96 analysis
2 to 8 hrs	2.16×10^{-3} sec/m^3	New ARCON96 analysis
8 to 24 hrs	8.90×10^{-4} sec/m^3	New ARCON96 analysis
1 to 4 days	6.61×10^{-4} sec/m^3	New ARCON96 analysis
4 to 30 days	5.01×10^{-4} sec/m^3	New ARCON96 analysis

Table 1.8.1-1 (continued)

Time Period	AST X/Q Value	Basis
Fuel Handling Area Releases to VPH-1		
0 to 2 hrs	1.48×10^{-3} sec/m^3	New ARCON96 analysis
2 to 8 hrs	1.07×10^{-3} sec/m^3	New ARCON96 analysis
8 to 24 hrs	4.37×10^{-4} sec/m^3	New ARCON96 analysis
1 to 4 days	3.04×10^{-4} sec/m^3	New ARCON96 analysis
4 to 30 days	2.44×10^{-4} sec/m^3	New ARCON96 analysis
Fuel Handling Area Releases to VPH-2		
0 to 2 hrs	3.46×10^{-3} sec/m^3	New ARCON96 analysis
2 to 8 hrs	1.80×10^{-3} sec/m^3	New ARCON96 analysis
8 to 24 hrs	8.46×10^{-4} sec/m^3	New ARCON96 analysis
1 to 4 days	6.27×10^{-4} sec/m^3	New ARCON96 analysis
4 to 30 days	4.42×10^{-4} sec/m^3	New ARCON96 analysis
PRVS Releases to VPH-1		
0 to 2 hrs	4.46×10^{-3} sec/m^3	New ARCON96 analysis
2 to 8 hrs	2.80×10^{-3} sec/m^3	New ARCON96 analysis
8 to 24 hrs	1.31×10^{-3} sec/m^3	New ARCON96 analysis
1 to 4 days	8.70×10^{-4} sec/m^3	New ARCON96 analysis
4 to 30 days	6.97×10^{-4} sec/m^3	New ARCON96 analysis
PRVS Releases to VPH-1		
0 to 2 hrs	4.36×10^{-3} sec/m^3	New ARCON96 analysis
2 to 8 hrs	3.05×10^{-3} sec/m^3	New ARCON96 analysis
8 to 24 hrs	1.36×10^{-3} sec/m^3	New ARCON96 analysis
1 to 4 days	8.66×10^{-4} sec/m^3	New ARCON96 analysis
4 to 30 days	7.36×10^{-4} sec/m^3	New ARCON96 analysis
Containment Releases to VPH-1		
0 to 2 hrs	2.80×10^{-3} sec/m^3	New ARCON96 analysis
2 to 8 hrs	1.75×10^{-3} sec/m^3	New ARCON96 analysis
8 to 24 hrs	7.24×10^{-4} sec/m^3	New ARCON96 analysis
1 to 4 days	5.98×10^{-4} sec/m^3	New ARCON96 analysis
4 to 30 days	4.34×10^{-4} sec/m^3	New ARCON96 analysis
Containment Releases to VPH-2		
0 to 2 hrs	3.55×10^{-3} sec/m^3	New ARCON96 analysis
2 to 8 hrs	2.49×10^{-3} sec/m^3	New ARCON96 analysis
8 to 24 hrs	9.85×10^{-4} sec/m^3	New ARCON96 analysis
1 to 4 days	8.30×10^{-4} sec/m^3	New ARCON96 analysis
4 to 30 days	6.31×10^{-4} sec/m^3	New ARCON96 analysis

Table 1.8.1-2

Release-Receptor Combination Parameters

(Basis: Plant Design Drawings)

Release Point	Receptor Point	Release Height (m)	Receptor Height (m)	Distance (m)	Direction with respect to true north
ADV	N CR Intake	32.61	28.62	33.51	212
ADV	S CR Intake	32.61	28.65	23.67	229
MSSV	N CR Intake	30.78	28.62	20.62	228
MSSV	S CR Intake	30.78	28.65	15.26	266
Steam Pipe	N CR Intake	19.58	28.62	35.01	206
Steam Pipe	S CR Intake	19.58	28.65	24.17	219
Fuel Handling Area	N CR Intake	54.53	28.62	32.52	218
Fuel Handling Area	S CR Intake	54.53	28.65	23.77	237
PRVS Exhaust	N CR Intake	55.63	28.62	19.61	258
PRVS Exhaust	S CR Intake	55.63	28.65	21.09	295
Containment	N CR Intake	4.22(1)	28.62	20.91	246
Containment	S CR Intake	4.22(1)	28.65	17.6	264

(1) Initial vertical diffusion coefficient; containment assumed to be a diffuse area with a height of 83 feet (above the auxiliary building). The diffuse area width (containment diameter) is 123.5 feet.

Table 1.8.2-1

Offsite Atmospheric Dispersion Factors for ANO-1

Time Period	AST X/Q Value	Basis
Exclusion Area Boundary		
0 to 30 days	6.8×10^{-4} s/m^3	Conforms to RG 1.183, Main Section, RP 5.3; Consistent with CLB
Low Population Zone		
0 to 8 hrs	1.1×10^{-4} sec/m^3	Conforms to RG 1.183, Main Section, RP 5.3; Consistent with CLB
8 to 24 hrs	1.1×10^{-5} sec/m^3	Conforms to RG 1.183, Main Section, RP 5.3; Consistent with CLB
1 to 4 days	4.0×10^{-6} sec/m^3	Conforms to RG 1.183, Main Section, RP 5.3; Consistent with CLB
4 to 30 days	1.3×10^{-6} sec/m^3	Conforms to RG 1.183, Main Section, RP 5.3; Consistent with CLB

Table 2.1-1

LOCA Input Parameters

Parameter	Input Value	Basis
Power level for analyses (102% of 2568 MWt)	2619.36 MWt	Conforms to RG 1.183, Main Section, RP 3.1; Licensed power = 2568 MWt
Core Average Fuel Burnup	41,045 MWD/MTU	Conforms to RG 1.183, Main Section, RP 3.1
Maximum Fuel Enrichment	4.1 w/o	Conforms to RG 1.183, Main Section, RP 3.1
Margin Added to ORIGEN Source Term Results	4%	Conservative addition to accommodate potential future variations in fuel enrichment
Core Fission Product Inventory	Table 1.7.4-1	Conforms to RG 1.183, Main Section, RPs 3.1 and 3.4
Gap Release Phase	30 sec – 0.5 hrs	Conforms to RG 1.183, Main Section, RP 3.3, Table 4
Early In-Vessel Release Phase	0.5 – 1.8 hrs	Conforms to RG 1.183, Main Section, RP 3.3, Table 4
Gap Release Fraction	0.05 for noble gases, halogens, and alkali metals only	Conforms to RG 1.183, Main Section, RP 3.2, Table 2
Early In-Vessel Release Fractions	0.95 noble gases 0.35 halogens 0.25 alkali metals 0.05 tellurium metals 0.02 strontium and barium 0.0025 noble metals 0.0005 cerium group 0.0002 lanthanides	Conforms to RG 1.183, Main Section, RP 3.2, Table 2
Iodine species distribution (%)	95.00 particulate 4.85 elemental 0.15 organic	Conforms to RG 1.183, Main Section, RP 3.5 and Appendix A, RP 2; Spray Ph greater than 7 during injection and recirculation due to inline NaOH tank
Containment Net Free Volume	1.81×10^6 ft^3	Conservative internal calculation
Containment Leak Rates	0.2%/day = 24 hrs 0.1%/day > 24 hrs	Conforms to RG 1.183, Appendix A, RP 3.7
Unsprayed Containment Volume	2.00×10^5 ft^3 (rounded up)	Conservative internal calculation
Sump Volume	54,918 ft^3	Conservative internal calculation

Table 2.1-1 (continued)

Parameter	Input Value	Basis
Sprayed Containment Volume	1.61×10^6 ft^3	Conservative internal calculation
Containment Sprayed Fractions	0.11 unsprayed 0.89 sprayed	Conservative internal calculation
Containment Mixing Rates	6270 cfm unsprayed to sprayed 6270 cfm sprayed to unsprayed	Bounds RG 1.183, Appendix A, RP 3.3; RG would allow 6637 cfm
Spray Removal Rates		
Elemental	20 hr^{-1} during injection, 10 hr^{-1} during recirculation until DF=200, then 0; DF=200 at 3.1 hours	Bounds RG 1.183, Appendix A, RP 3.3; Reduction to 10 during recirculation not required
Organic	No removal	Bounds RG 1.183, Appendix A, RP 3.3; Removal credit, same as particulate, allowed
Particulate	2.60 hr^{-1} until DF=50 at 3.3 hours, then 0.26 hr^{-1} until DF=1000 at 16.5 hours, then 0	Bounds RG 1.183, Appendix A, RP 3.3; Reduction to 0 when DF = 1000 not required by RG
Spray Initiation Time (no termination)	300 sec	No termination conforms to RG 1.183, Appendix A, RP 5.2
Natural Deposition in Unsprayed Region	No credit taken	Conforms to RG 1.183, Appendix A, RP 3.2; RADTRAD used
Amount of Containment Leakage into Penetration Rooms	50%	Consistent with CLB
Penetration Room Ventilation System Filter Efficiency	99% particulates 90% elemental and organic iodines 0% noble gases	Conservative assumption bounded by current TS 5.5.11; System conforms to RG 1.183, Main Section, RP 5.1.2
Offsite X/Q	Table 1.8.2-1	Consistent with CLB
Offsite Breathing Rates	Per RG 1.183 Sections 4.1.3	Conforms to RG 1.183, Main Section, RP 4.1.3
CR X/Q (used limiting worst-case PRVS exhaust release)	4.46E-3 s/m^3 0-2 hrs 3.05E-3 s/m^3 2-8 hrs 1.36E-3 s/m^3 8-24 hrs 8.70E-4 s/m^3 24-96 hrs 7.36E-4 s/m^3 >96 hrs	Conforms to RG 1.183, Main Section, RPs 4.2.2 and 4.2.3

Table 2.1-1 (continued)

Parameter	Input Value	Basis
CR Breathing Rate	Per RG 1.183 Section 4.2.6	Conforms to RG 1.183, Main Section, RP 4.2.6
CR Occupancy Factors	Per RG 1.183 Section 4.2.6	Conforms to RG 1.183, Main Section, RP 4.2.6
Dose Conversion Factors (DCF)	Federal Guidance Report 11 CEDE and Federal Guidance Report 12 EDE	Conforms to RG 1.183, Main Section, RPs 4.1.2 and 4.1.4
Control Room Ventilation System	Table 1.6.3-1	Conforms to RG 1.183, Main Section, RPs 4.2.4 and 5.1.2
Time of CR Isolation	10 seconds	Conservative value assumed; System design = 5 sec
CR Unfiltered Inleakage	82 cfm	Consistent with proposed new TS limit

ESF Leakage Input for LOCA Analyses

Parameter	Input Value	Basis
ESF Leakage Rate	782 cc/hr (4.603E-4 cfm)	Conforms to RG 1.183, Appendix A, RP 5.2
Fraction of Released Iodine in Sump Solution	1.0	Conforms to RG 1.183, Appendix A, RP 5.1
Iodine Species Distribution in Sump	0.97 elemental 0.03 organic	Conforms to RG 1.183, Appendix A, RP 5.6
Time to Recirculation	4257 sec (1.1825 hr)	Conforms to RG 1.183, Appendix A, RP 5.2
Iodine Partition Coefficient for ESF Leakage (Flashing Fraction)	Calculated – 4.58% Used in analysis – 10%	Conforms to RG 1.183, Appendix A, RPs 5.4 and 5.5
Release Filtration Assumed	None	Bounds RG 1.183, Appendix A, RP 5.6

Table 2.1-2

LOCA Dose Results Summary

	EAB (worst 2-hour)	**LPZ (30 days)**	**CR (82 cfm, 30 days)**
Containment Leakage	10.454	2.5263	3.5386
ECCS Leakage	3.3744 x 10-2	2.473 x 10-2	9.8989 x 10^{-2}
Cloud Shine	0	0	4.8078 x 10^{-2}
Containment Shine	0	0	8.01 x 10^{-2}
Total TEDE Dose	10.4877	2.551	3.7658

Table 2.2-1

FHA Input Parameters

Parameter	Input Value	Basis
Power level for analyses (102% of 2568 MWt)	2619.36 MWt	Conforms to RG 1.183, Main Section, RP 3.1; Licensed power = 2568 MWt
Core Average Fuel Burnup	41,045 MWD/MTU	Conforms to RG 1.183, Main Section, RP 3.1
Maximum Average Fuel Enrichment	4.1 w/o	Conforms to RG 1.183, Main Section, RP 3.1
Margin Added to ORIGEN Source Term Results	4%	Conservative addition to accommodate potential future variations in fuel enrichment
Peaking Factor	1.8	Consistent with CLB
Number of Fuel Assemblies in Core	177	Consistent with CLB
Number of Damaged Rods	82 (six rows)	Consistent with CLB
Fuel Rod Pressure Limit	1500 psig	Consistent with NRC SER on WCAP-16072-P (Ref. 1)
Water Level Above Damaged Fuel	23 feet minimum	Conforms to RG 1.183, Appendix B, RP 2
Delay Before Fuel Movement	72 hrs	Conservative with respect to TS Bases for TSs 3.7.13, 3.9.3, and 3.9.6
Core Fission Product Inventory	Table 1.7.5-1	Conforms to RG 1.183, Main Section, RPs 3.1 and 3.4
Gap Fractions Released Kr-85 I-131 (modified per NUREG/CR-5009) Other isotopes	 0.30 0.12 0.10	Bounds RG 1.183, Main Section, RP 3.2, Table 3, except alkali metals which are retained in pool water; Values conform to RG 1.25, but increased 20% for I-131 to accommodate higher burnup fuel
Iodine Form in Pool Elemental Organic	 99.85% 0.15%	Conforms to RG 1.183, Appendix B, RP 1.3
Iodine Form Above Pool Elemental Organic	 70% 30%	Bounds RG 1.183, Appendix B, RP 2
Pool Decontamination Factors Elemental Iodine Organic Iodine and Noble Gases	 286 (limited to provide overall DF = 200) 1	 Bounds RG 1.183, Appendix B, RP 2 Conforms to RG 1.183, Appendix B, RP 2

Table 2.2-1 (continued)

Parameter	Input Value	Basis
Offsite and CR Breathing Rate (duration of event)	3.5×10^{-4} m^3/s	Conforms to RG 1.183, Main Section, RPs 4.1.3 and 4.2.6
Offsite X/Q (duration of event)	6.8×10^{-4} s/m^3 EAB 1.1×10^{-4} s/m^3 LPZ	Consistent with CLB for LOCA; Conservative with respect to CLB for FHA
Control Room X/Q (containment more limiting than fuel handling area ventilation) (duration of event)	3.55×10^{-3} s/m^3	Conforms to RG 1.183, Main Section, RPs 4.2.2 and 4.2.3
Dose Conversion Factors (DCF)	Federal Guidance Report 11 CEDE and Federal Guidance Report 12 EDE	Conforms to RG 1.183, Main Section, RPs 4.1.2 and 4.1.4
Control Room Ventilation System	Table 1.6.3-1	Conforms to RG 1.183, Main Section, RPs 4.2.4 and 5.1.2
Time of CR Isolation	36 seconds	Conservative value assumed; System design = 5 sec
CR Unfiltered Inleakage	85 cfm	Bounds proposed new TS limit

Table 2.3-1

MSLB Input Parameters

Parameter	Input Value	Basis
Power level for analyses (102% of 2568 MWt)	2619.36 MWt	Conforms to RG 1.183, Main Section, RP 3.1; Licensed power = 2568 MWt
Initial Primary Coolant Activity	1.0 μCi/gm DE I-131 and 72/E-bar gross activity (Table 1.7.2-1)	Conforms to RG 1.183, Appendix E, RP 2; DE-131 value consistent with proposed new TS limit
Activity with Pre-existing Iodine Spike	60 μCi/g I-131	Conforms to RG 1.183, Appendix E, RP 2.1
Initial Secondary Coolant Activity	0.1 μCi/g I-131 (Table 1.7.3-1)	No guidance in RG 1.183; Value consistent with proposed new TS limit
Accident-Initiated Iodine Spike Factor	500	Conforms to RG 1.183, Appendix E, RP 2.2
Accident-Initiated Iodine Spike Duration	8 hrs	Conforms to RG 1.183, Appendix E, RP 2.2
Primary-to-Secondary Leak Rate	0.5 gpm/SG	Bounds RG 1.183, Appendix E, RP 5.1; TS 3.4.13 allows 150 gpd/SG
Time to Begin Cooldown (operator action)	30 min	No guidance in RG 1.183; Conservative value assumed
Time to Isolation of Unaffected SG (initiation of DHR)	237.8 hrs	Conforms to RG 1.183, Appendix E, RP 5.3; Conservative with respect to internal calculation
Time to Reach 212 F/Terminate Steam Release	251.8 hrs	Conforms to RG 1.183, Appendix E, RP 5.3; Conservative with respect to internal calculation
Faulted SG Mass	6.00E+4 lbm	Conservative internal calculation
Flashing Fraction in Unaffected SG	0.2	Conservative internal calculation based on flashing model; OTSG tubes never totally submerged
Partition Coefficient (faulted SG and intact SG via flashing and vaporization)	1.0	Conforms to RG 1.183, Appendix E, RP 5.5.1
Partition Coefficients (intact SG via steaming)	0.01 iodines 0.001 alkali metals	Conforms to RG 1.183, Appendix E, RP 5.5.4

Table 2.3-1 (continued)

Parameter	Input Value	Basis
RCS Mass	Maximum 2.38×10^8 gm Minimum 2.33×10^8 gm Maximum to produce largest equilibrium appearance rate; minimum to maximize activity concentration	Conservative internal calculation
SG Secondary Mass	Maximum 2.72×10^7 gm Minimum 1.71×10^7 gm Maximum in faulted SG to maximize release; minimum in intact SG to maximize activity concentration	Conservative internal calculation
Iodine Form of Secondary Release Particulate Elemental Organic	 0% 97% 3%	Conforms to RG 1.183, Appendix E, RP 4
Offsite X/Q	Table 1.8.2-1	Consistent with CLB for LOCA; Conservative with respect to CLB for MSLB
Offsite Breathing Rates	Per RG 1.183 Sections 4.1.3	Conforms to RG 1.183, Main Section, RP 4.1.3
CR X/Q (faulted SG – used worst-case main steam line)	3.15E-3 s/m3 0-2 hrs 2.16E-3 s/m3 2-8 hrs 8.90E-4 s/m3 8-24 hrs 6.61E-4 s/m3 24-96 hrs 5.01E-4 s/m3 > 96 hrs	Conforms to RG 1.183, Main Section, RPs 4.2.2 and 4.2.3
CR X/Q (intact SG – used worst-case MSSV for first 30 min, then used worst-case ADV for each time step)	1.90E-2 s/m3 0-0.5 hrs 4.10E-3 s/m3 0.5-2 hrs 2.59E-3 s/m3 2-8 hrs 1.12E-3 s/m3 8-24 hrs 8.32E-4 s/m3 24-96 hrs 5.91E-4 s/m3 > 96 hrs	Conforms to RG 1.183, Main Section, RPs 4.2.2 and 4.2.3

Table 2.3-1 (continued)

Parameter	Input Value	Basis
CR Breathing Rate	Per RG 1.183 Section 4.2.6	Conforms to RG 1.183, Main Section, RP 4.2.6
CR Occupancy Factors	Per RG 1.183 Section 4.2.6	Conforms to RG 1.183, Main Section, RP 4.2.6
Dose Conversion Factors (DCF)	Federal Guidance Report 11 CEDE and Federal Guidance Report 12 EDE	Conforms to RG 1.183, Main Section, RPs 4.1.2 and 4.1.4
Control Room Ventilation System	Table 1.6.3-1	Conforms to RG 1.183, Main Section, RPs 4.2.4 and 5.1.2
Time of CR Isolation	10 seconds	Conservative value assumed; System design = 5 sec
CR Unfiltered Inleakage	85 cfm	Bounds proposed new TS limit

Table 2.4-1

SGTR Input Parameters

Parameter	Input Value	Basis
Power level for analyses (102% of 2568 MWt)	2619.36 MWt	Conforms to RG 1.183, Main Section, RP 3.1; Licensed power = 2568 MWt
Initial Primary Coolant Activity	1.0 μCi/gm DE I-131 and 72/E-bar gross activity (Table 1.7.2-1)	Conforms to RG 1.183, Appendix F, RP 2; DE-131 value consistent with proposed new TS limit
Activity with Pre-existing Iodine Spike	60 μCi/g I-131	Conforms to RG 1.183, Appendix F, RP 2.1
Initial Secondary Coolant Activity	0.1 μCi/g I-131 (Table 1.7.3-1)	No guidance in RG 1.183; Value consistent with proposed new TS limit
Accident-Initiated Iodine Spike Factor	335	Conforms to RG 1.183, Appendix F, RP 2.2
Accident-Initiated Iodine Spike Duration	8 hrs	Conforms to RG 1.183, Appendix F, RP 2.2
Initial Ruptured SG Tube Leak Rate	435 gpm	Conservative internal calculation, consistent with CLB
Primary-to-Secondary Leak Rate	150 gpd per steam generator	Conforms to RG 1.183, Appendix F, RP 5.1
Time to Reactor Trip (full steaming until trip)	11 min	Bounds RG 1.183, Appendix F, RP 5.4; Conservative internal calculation, consistent with CLB; LOOP occurs at time of trip
Time to Isolation of Faulted SG	34 min	Conforms to RG 1.183, Appendix F, RP 5.3; Conservative internal calculation, consistent with CLB
Time to Isolation of Intact SG (initiation of DHR)	237.8 hrs	Conforms to RG 1.183, Appendix F, RP 5.3; Conservative with respect to internal calculation
Flashing Fraction in Faulted SG	0.15	Conservative internal calculation based on flashing model; OTSG tubes never totally submerged
Partition Coefficients prior to Reactor Trip (release via condenser)	0.0001 iodines and alkali metals	Value consistent with CLB; No specific guidance in RG 1.183, but value also consistent with 0.01 in SG (per RP 5.5.4 of RG Appendix E) x 0.01 in condenser

Table 2.4-1 (continued)

Parameter	Input Value	Basis
Partition Coefficient after Reactor Trip (flashing and vaporization via MSSV or ADV)	1.0	Conforms to RG 1.183, Appendix F, RP 5.6
Partition Coefficients after Reactor Trip (SG steaming via MSSV or ADV)	0.01 iodines 0.001 alkali metals	Conforms to RG 1.183, Appendix F, RP 5.6
RCS Mass	Maximum 2.38 x 10^8 gm Minimum 2.33 x 10^8 gm Maximum to produce largest equilibrium appearance rate; minimum to maximize activity concentration	Conservative internal calculation
SG Secondary Mass	1.71 x 10^7 gm Minimum used to maximize activity concentration	Conservative internal calculation
Offsite X/Q	Table 1.8.2-1	Consistent with CLB for LOCA; Conservative with respect to CLB for SGTR
Offsite Breathing Rates	Per RG 1.183 Sections 4.1.3	Conforms to RG 1.183, Main Section, RP 4.1.3
CR X/Q (used worst-case ADV for each time step, except from trip at 11 min to SG isolation at 34 min, used worst-case MSSV)	4.10E-3 s/m^3 0-11 min (reactor trip) 1.90E-2 s/m^3 11-34 min (faulted SG isolated) 4.10E-3 s/m^3 0.5667-2 hrs 2.59E-3 s/m^3 2-8 hrs 1.12E-3 s/m^3 8-24 hrs 8.32E-4 s/m^3 24-96 hrs 5.91E-4 s/m^3 > 96 hrs	Conforms to RG 1.183, Main Section, RPs 4.2.2 and 4.2.3
CR Breathing Rate	Per RG 1.183 Section 4.2.6	Conforms to RG 1.183, Main Section, RP 4.2.6
CR Occupancy Factors	Per RG 1.183 Section 4.2.6	Conforms to RG 1.183, Main Section, RP 4.2.6
Dose Conversion Factors (DCF)	Federal Guidance Report 11 CEDE and Federal Guidance Report 12 EDE	Conforms to RG 1.183, Main Section, RPs 4.1.2 and 4.1.4

Table 2.4-1 (continued)

Parameter	Input Value	Basis
Control Room Ventilation System	Table 1.6.3-1	Conforms to RG 1.183, Main Section, RPs 4.2.4 and 5.1.2
Time of CR Isolation	11 minutes	Conservative value assumed
CR Unfiltered Inleakage	85 cfm	Bounds proposed new TS limit

Table 2.4-2

Iodine Equilibrium Appearance Rate Without Spike

(Removal Term Based on Conservative Calculation)

Isotope	RCS Activity (Ci)	Removal (hr^{-1})		Appearance Rate (Ci/hr)	Appearance Rate (Ci/min)
I-130	6.82E+02	0.2002	=	1.37E+02	2.28E+00
I-131	7.33E+01	0.1479	=	1.08E+01	1.80E-01
I-132	1.00E+03	0.4457	=	4.46E+02	7.43E+00
I-133	6.91E+02	0.1776	=	1.23E+02	2.05E+00
I-134	1.48E+03	0.9356	=	1.39E+03	2.32E+01
I-135	1.22E+03	0.2492	=	3.04E+02	5.07E+00

Table 2.4-3

SGTR Activity for Accident-Initiated Iodine Spike

(Conforms to RG 1.183, Appendix F, RP 2.2)

Isotope	Equilibrium Iodine Appearance Rates (Ci/min)	Fuel Activity Modeled in *.nif file (Ci)
I-130	2.28E+00	2.28E+10
I-131	1.80E-01	1.80E+09
I-132	7.43E+00	7.43E+10
I-133	2.05E+00	2.05E+10
I-134	2.32E+01	2.32E+11
I-135	5.07E+00	5.07E+10

Table 2.6-1

CREA Input Parameters

Parameter	Input Value	Basis
Power level for analyses (102% of 2568 MWt)	2619.36 MWt	Conforms to RG 1.183, Main Section, RP 3.1; Licensed power = 2568 MWt
Core Average Fuel Burnup	41,045 MWD/MTU	Conforms to RG 1.183, Main Section, RP 3.1
Maximum Fuel Enrichment	4.1 w/o	Conforms to RG 1.183, Main Section, RP 3.1
Margin Added to ORIGEN Source Term Results	4%	Conservative addition to accommodate potential future variations in fuel enrichment
Core Fission Product Inventory	Table 1.7.4-1	Conforms to RG 1.183, Main Section, RPs 3.1 and 3.4
Fuel Failure (rods in DNB)	14%	Consistent with CLB
Peaking Factor	1.8	Consistent with CLB
Fission Product Gap Fractions (RG 1.183, Appendix H, Section 1)	0.10 noble gases and iodines 0.12 alkali metals	Conforms to RG 1.183, Appendix H, RP 1
Containment Release Iodine Species Distribution	95% particulate 4.85% elemental 0.15% organic	Conforms to RG 1.183, Appendix H, RP 4
Secondary Release Iodine Species Distribution	0% particulate 97% elemental 3% organic	Conforms to RG 1.183, Appendix H, RP 5
Primary-to-Secondary (P-S) Leak Rate (secondary release model)	300 gpd	Conforms to RG 1.183, Appendix H, RP 7.1
Duration of Secondary Release Event (switch to DHR system)	38.25 hrs	Conforms to RG 1.183, Appendix H, RP 7.1; Conservative with respect to internal calculation
Flashing and Vaporizing Fraction of P-S Leakage during Cooldown (no partitioning)	0.15	Conservative internal calculation based on flashing model; OTSG tubes never totally submerged
Containment Net Free Volume	1.81×10^6 ft^3	Conservative internal calculation
Containment Leak Rates	0.2%/day = 24 hrs 0.1%/day > 24 hrs	Conforms to RG 1.183, Appendix H, RP 6.2
Sedimentation Coefficient (Particulates only)	0.1/hr until DF = 1000, then 0	Conforms to RG 1.183, Appendix H, RP 6.1; Value consistent with Reference 2

Table 2.6-1 (continued)

Parameter	Input Value	Basis
Containment Spray	No credit taken	Bounds RG 1.183, Appendix H, RP 6.1; Credit allowed
Penetration Room Ventilation System	No credit taken	Bounds RG 1.183, Appendix H, RP 6.1; Credit allowed
Partition Coefficients of P-S Leakage Mixed with Secondary Liquid Inventory	0.01 iodines 0.001 alkali metals	Conforms to RG 1.183, Appendix H, RP 7.4
Steam Release Rates from Secondary	2.5815E+6 g/min 0-2 hrs 5.6977E+5 g/min 2-38.25 hrs	Conservative internal calculation based on steam release model
RCS Mass	2.332×10^8 gm Minimum used to maximize activity concentration	Conservative internal calculation
SG Secondary Mass	3.411×10^7 gm Minimum for 2 SGs used to maximize activity concentration	Conservative internal calculation
Offsite X/Q	Table 1.8.2-1	Consistent with CLB
Offsite Breathing Rates	Per RG 1.183 Sections 4.1.3	Conforms to RG 1.183, Main Section, RP 4.1.3
CR X/Q (containment release)	3.55E-3 s/m^3 0-2 hrs 2.49E-3 s/m^3 2-8 hrs 9.85E-4 s/m^3 8-24 hrs 8.30E-4 s/m^3 24-96 hrs 6.31E-4 s/m^3 > 96 hrs	Conforms to RG 1.183, Main Section, RPs 4.2.2 and 4.2.3
CR X/Q (secondary release) (used worst-case MSSV for first 30 min, then used worst-case ADV for each time step)	1.90E-2 s/m^3 0-0.5 hrs 4.10E-3 s/m^3 0.5-2 hrs 2.59E-3 s/m^3 2-8 hrs 1.12E-3 s/m^3 8-24 hrs 8.32E-4 s/m^3 24-96 hrs 5.91E-4 s/m^3 > 96 hrs	Conforms to RG 1.183, Main Section, RPs 4.2.2 and 4.2.3
CR Breathing Rate	Per RG 1.183 Section 4.2.6	Conforms to RG 1.183, Main Section, RP 4.2.6
CR Occupancy Factors	Per RG 1.183 Section 4.2.6	Conforms to RG 1.183, Main Section, RP 4.2.6
Dose Conversion Factors (DCF)	Federal Guidance Report 11 CEDE and Federal Guidance Report 12 EDE	Conforms to RG 1.183, Main Section, RPs 4.1.2 and 4.1.4
Control Room Ventilation System	Table 1.6.3-1	Conforms to RG 1.183, Main Section, RPs 4.2.4 and 5.1.2

Table 2.6-1 (continued)

Parameter	Input Value	Basis
Time of CR Isolation	10 seconds	Conservative value assumed; System design = 5 sec
CR Unfiltered Inleakage	82 cfm	Consistent with proposed new TS limit

Table 2.6-2

CREA Steam, Iodine and Alkali Metal Release Rates

(Conforms to RG 1.183, Appendix H, RP 7.4)

Time (hours)	Steam Release Rate (gm/min)	Iodine Release Rate(1) (gm/min)	Alkali Metal Release Rate(2) (gm/min)
0-2	2.5815×10^{6}	2.5815×10^{4}	2.5815×10^{3}
2-38.25	5.6977×10^{5}	5.6977×10^{3}	5.6977×10^{2}

(1) Assumes partition factor of 100 for iodines

(2) Assumes moisture carryover of <0.1% for alkali metals

Table 3-1

Arkansas Nuclear One, Unit No. 1
Summary of Alternative Source Term Analysis Results

CASE	Assumed Unfiltered CR Inleakage (cfm)	EAB Dose(1) (rem TEDE)	LPZ Dose(2) (rem TEDE)	CR Dose(2) (rem TEDE)
LOCA	82	10.49	2.56	3.77
SGTR Pre-existing Iodine Spike	85	2.20	0.37	2.33
MSLB Pre-existing Iodine Spike	85	0.45	0.19	1.84
Acceptance Criteria		**25.0**(3)	**25.0**(3)	**5.0**(4)
SGTR Accident-initiated Iodine Spike	85	1.26	0.23	1.00
MSLB Accident-initiated Iodine Spike	85	2.07	1.05	3.72
Acceptance Criteria		**2.5**(3)	**2.5**(3)	**5.0**(4)
FHA 72-hr decay	85	1.40	0.25	1.00
CREA Containment Release	82	4.73	2.28	3.40
CREA Secondary Release	82	3.03	1.64	4.95
Acceptance Criteria		**6.3**(3)	**6.3**(3)	**5.0**(4)

(1) Worst 2-hour dose

(2) Integrated 30-day dose

(3) RG 1.183, Table 6

(4) 10 CFR 50.67

Attachment 3

To

1CAN080801

Updated Markup of ANO-1 Safety Analysis Report

1.2 DESIGN SUMMARY

1.2.1 SITE CHARACTERISTICS

The site consists of approximately 1,100 acres providing for a 0.65 mile exclusion radius. The site is characterized by remoteness from population centers; freedom from flooding; sound, hard rock for structure foundations; a reliable network for emergency power; and favorable conditions of hydrology, geology, seismology, and meteorology.

1.2.2 POWER LEVEL

The design and license power level for the reactor core will be 2,568 MWt, and all physics and core thermal hydraulics information in this report is based on that power level. An additional 21 MWt is available to the cycle from the contribution of the reactor coolant pumps resulting in a design gross electrical output of 911.5 MWe.

1.2.3 PEAK SPECIFIC POWER LEVEL

For cycle one, the peak specific power level in the fuel for operation at 2,568 MWt results in a maximum thermal output of 17.63 kW/ft of fuel rod. This value is comparable with other reactors of this size presently under construction, and with reactors in the 400-500 MWe class such as San Onofre, Ginna, and Connecticut Yankee, and therefore did not represent an extrapolation of technology.

1.2.4 CONTAINMENT SYSTEM

The reactor building is a fully continuous reinforced concrete structure in the shape of a cylinder on a flat foundation slab with a shallow domed roof. The cylindrical portion is prestressed by a post-tensioning system consisting of horizontal and vertical tendons. The dome has a 3-way post-tensioning system. Hoop tendons are placed in 3-240 degree systems using three buttresses that run the full height of the cylinder as anchorages. The foundation slab is conventionally reinforced with high-strength reinforcing steel. A continuous access gallery is provided beneath the base slab for installation and inspection of vertical tendons. A welded steel liner is attached to the inside face of the concrete shell to ensure a high degree of leak tightness. The base liner has been installed on top of the structural slab and was covered with concrete. The structure provides shielding for both normal and accident conditions.

The reactor building will completely enclose the entire reactor and the Reactor Coolant System and ensure that an acceptable upper limit for leakage of radioactive materials to the environment would not be exceeded even if gross failure of the Reactor Coolant System were to occur. The building encloses the Pressurized Water Reactor, steam generators, reactor coolant loops and portions of the auxiliary systems and engineered safeguard systems.

1.2.5 ENGINEERED SAFEGUARDS

The Engineered Safeguards (ES) have sufficient redundancy of component and power sources such that under the conditions of the worst postulated Loss of Coolant Accident (LOCA) the system can maintain the integrity of the containment and keep the exposure of the public below the limits of 10 CFR 50.67~~100~~.

1.4.14 CRITERION 18 - INSPECTION AND TESTING OF ELECTRICAL POWER SYSTEMS

Electrical power systems important to safety shall be designed to permit periodic inspection and testing of important areas and features, such as wiring, insulation, connectors, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power buses, the offsite power system, and the onsite power system.

Discussion

All important passive components of the emergency power system such as wiring, insulation, connections, and switchboards are designed to permit appropriate periodic inspection and testing to assess their continuity and condition.

System design provides for the following periodic Emergency Diesel Generator electrical tests:

A. Each diesel generator is manually started each month and demonstrated to be ready for loading within 15 seconds. On this manual start, the signal initiating the start of the diesel is varied from one test to another to verify all starting circuits are operable. The generator is synchronized from the control room and loaded.

B. A test is conducted at least once each 18 months to demonstrate the overall automatic operation of the emergency power system. The test is initiated by a simulated simultaneous loss of normal and standby power sources and a simulated ES signal. Proper operations are verified by bus load shedding and automatic starting of selected motors and equipment to establish that restoration with emergency power has been accomplished within a limited time interval, approximately 70 seconds.

1.4.15 CRITERION 19 - CONTROL ROOM

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposure in excess of 5 TEDE~~rem whole body or its equivalent to any part of the body~~, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Discussion

The control room is designed to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions. As discussed in Sections 11.2.4 and 14.2.2.6, adequate radiation protection has been provided to insure that radiation exposures to personnel occupying the control room ~~during the 30-day period~~ following an ~~maximum hypothetical~~ accident will not exceed 5 rem TEDE~~whole body or its equivalent to any part of the body~~, for the duration of the accident.

The control room is designed so that one man can operate the unit during normal steady state conditions. During other operating conditions, other operators will be available to assist the control operator. In the event that the control room must be evacuated, equipment at appropriate locations outside the control room is provided with a design capability for prompt hot standby, > 525 degrees F of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot standby, > 525 degrees F. In addition, the potential capability for subsequent cold shutdown will be provided for use in the event that the control room has to be evacuated and is not accessible for a long period of time under the conditions where no accident has taken place and the control room is still intact.

The following design features are provided to insure continuous control room access:

A. Adequate shielding to maintain tolerable radiation levels in the control room during a Design Basis Accident (DBA).

B. Three points of entry from outside the control room.

C. Nonflammable construction.

D. Cables and switchboard wiring pass flame test per IPCEA publication S-61-402 and NEMA WC5-1961.

E. Combustibles such as furniture have been evaluated.

F. Smoke protection and detection equipment is provided.

The Reactor Protection System is designed to be essentially fail-safe without operator control. Thus, safe shutdown can be achieved without operator action.

If the reactor is tripped, and the control room evacuated, reactor decay heat is removed by the steam generators, with steam exhausting through the main turbine bypass valve and/or atmospheric dump valve. Either the main or an emergency feedwater pump will continue to supply feedwater to the steam generators. Additionally, a motor driven auxiliary feedwater pump, normally used for startup and connected in parallel with the main feedwater pumps, could be used to supply feedwater to the steam generators. Under these conditions, a balance will be maintained between heat removal and decay heat generation, the RCS will be maintained at normal hot standby, > 525 degrees F temperature, and no significant makeup will be required for several hours. Any makeup can be supplied by operating the makeup pump, taking suction from the Borated Water Storage Tank and discharging through the normal makeup system lineup. These makeup operations can be conducted locally and the controls and instrumentation are adequate to maintain the plant in a safe hot standby, > 525 degrees F condition during the period of control room inaccessibility.

Liquid and solid wastes are normally processed in batches for offsite disposal. Gaseous waste released to the environment is monitored and discharged to assure tolerable activity levels on the site and at the exclusion distance.

The Gaseous Waste System can store accumulated gas generated during operation. The contents of the decay tanks are periodically sampled, and a release rate is established consistent with the prevailing environmental conditions. In-line monitoring provides a continuous check on the release of activity.

Permanently installed area detectors and the plant vent detectors monitor the discharge levels. In addition, portable monitors are available on site for supplemental surveys if necessary.

Radioactive liquid leakage into the cooling water systems is detected by monitors. These monitors are used for normal operational protection as well as for accident conditions. Detectors monitor the gaseous activity prior to discharge.

1.4.52 CRITERION 61 - FUEL STORAGE AND HANDLING AND RADIOACTIVITY CONTROL

The Fuel Storage and Handling, Radioactive Waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability of a reliability and testability that reflects the safety importance of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

<u>Discussion</u>

The Fuel Storage and Handling, Radioactive Waste, and other systems which may contain radioactivity are designed to assure adequate safety under normal and postulated accident conditions. The systems have the capability for periodic inspection and testing of components important to safety. The shielding design considerations are discussed in Section 11.2.3.

Damage to a fuel assembly in the spent fuel pool releasing radioactive gases to the auxiliary building was evaluated. ~~With filtration and E~~exhaust of these gases through the plant vent without filtration results in~~, the~~ offsite dose~~s is well~~ below the 10 CFR 50.67 acceptance criteria~~100 guidelines~~.

Accidents assuming rupture of a waste gas tank have been evaluated and the consequences of the release were~~are~~ shown to be well below the guideline values of 10 CFR 100.

Radioactive liquid effluent which might accidentally leak into the Intermediate Cooling Water System will be detected by a radiation monitor. Any accidental leakage from liquid waste storage tanks will be collected in the auxiliary building sump and transferred to other tanks to prevent release to the environment.

A small purification loop removes fission products and other contaminants in the spent fuel storage pool water. The basis for the design of the Spent Fuel Cooling System reflects the importance of this system to safety. The capability for appropriate testing has been provided.

The shutdown condition assumes that the reactor core has been operating at 2,568 MWt for at least 1,000 hours. At 1,000 hours the fission product inventory approaches the infinite operation case.

11.2.4 SHIELDING DESIGN DESCRIPTIONS

The different areas of radiation protection are listed by specific location or building for convenience.

11.2.4.1 Main Control Room

The original DBA defining the protection required for the plant main control room was~~is~~ the Maximum Hypothetical Accident (MHA). ANO-1 has since completed alternate source term dose analyses using the guidance of NRC Regulatory Guide 1.183. These analyses show the dose to the control room operators following any postulated event will be below the 5 rem TEDE acceptance criteria of 10 CFR 50.67. Except as noted, the following discussion continues to describe the original MHA analysis and is retained for historical purposes.~~The accident condition is described in Chapter 14 of this report.~~ The main control room design was based on the airborne fission product inventory in the reactor building following an MHA. ~~The activity inventory of noble gases and halogens is shown in Chapter 14.~~ Containment and control room shielding have been designed such that the doses to operating personnel for the duration of the MHA are less than 5 rem whole body or its equivalent to any part of the body.

The dose to personnel occupying the control room continuously for 30 days from the onset of the accident derives primarily from two sources:

A. Direct radiation from radioactive fission products inside the containment.

B. Radiation from the cloud formed by the leakage of fission products out of the containment.

The dose computed for direct containment shine to the control room personnel makes several assumptions:

A. The reactor has been operating for a long time such that the fission product inventory is the saturation inventory given by $q_s = (0.865 \times 10^6)\ (P_0)\ (V_0)$ Curies

where:

q_s = the saturation inventory of isotope i

P_0 = reactor power (Mwt) = 2,568 Mwt

V_0 = fission yield for isotope i

B. One hundred percent of the noble gases, 50 percent of the halogens and one percent of the solid fission products are released to the containment.

C. The isotopes in "B" above are uniformly distributed in the containment, taken to be a cylinder with free volume of 1.9×10^6 ft^3. No credit is taken for shielding by the internal structures in the containment. Credit is taken for the 3-foot, 9-inch containment wall.

D. No credit is taken for containment leakage, plateout of iodines or effectiveness of the containment spray system in removing fission products from the containment atmosphere. The only decrease in source strength is decay.

The 30-day integrated dose from containment shine is 210 mrem. The 30-day integrated containment shine dose calculated using alternate source terms is 80 mrem following a LOCA.

In determining the dose from cloud shine all of the above assumptions apply except that the containment leakage is conservatively assumed to be 0.2%/day for the duration of the accident.

The leakage is taken to be at the control room roof level and passed directly over the control room, continuously for 30 days. The dose to personnel from cloud shine is 950 mrem. The 30-day integrated cloud shine dose calculated using alternate source terms is 48 mrem following a LOCA.

The total 30-day dose from containment shine and cloud shine is 685 mrem.

The Emergency Air Conditioning and Filtration Systems provided for the control room are described in Section 9.7.2.~~ The evaluation of control room operator doses given in Section 14.2.2.6 shows that the dose received during the 30 days following a postulated LOCA is less than the limits of 10CFR50 General Design Criterion 19.~~

11.2.4.2 Reactor Building Shell

The reactor building shell is a reinforced prestressed concrete structure which serves two main shielding purposes:

A. During normal operation, it shields the surrounding plant structures and yard areas from radiation originating at the reactor vessel and the primary loop components. Together with additional shielding inside the containment, the concrete shell will reduce radiation levels outside the shell to below 1.0 mrem/hr in those areas which are occupied by personnel either on a permanent or routine basis.

B. In the event of an accident~~ MHA~~, the shell shielding will reduce plant and offsite radiation intensities emitted directly from released fission products below levels as defined by: (1) onsite occupancy limits of 5 rem TEDE~~whole body dose and 30 rem thyroid~~ and, (2) exclusion distance acceptance criteria of 10 CFR 50.67~~limits of 10 CFR 100~~. The concrete roof of the reactor building has been specifically designed to reduce radiation contributions from sky-shine. Activities inside the reactor building following an accident~~during an MHA~~ and the off-site doses associated with the accident~~MHA~~ are given in Chapter 14.

11.2.4.3 Reactor Building Interior

During reactor operation, access to most areas inside the containment will be prohibited due to high radiation levels.

Large sections over the steam generator compartments and the refueling canal are open and unshielded. These openings cause a high dose rate at the refueling floor during reactor operation. Neutrons streaming out of these areas increase the containment internal dose rate. The reactor vessel which is the major radiation source is surrounded by a concrete shield.

14.1.2.8 Loss of Electric Power

14.1.2.8.1 Identification of Cause

The unit is designed to withstand the effects of loss of electric load or electric power. Emergency power systems are described in Chapter 8. Two types of power losses are considered:

A. A loss of load condition caused by separation of the unit from the transmission system.

B. A hypothetical condition resulting in a complete loss of all system and unit power except the unit batteries.

14.1.2.8.2 Reactor Protection Criteria

The criteria for reactor protection for this accident are:

A. Fuel damage must not occur.

B. RCS pressure shall not exceed code pressure limits.

C. ~~(1) Resultant doses for loss of all AC power shall not exceed 10 CFR 100 limits.~~

~~(2)~~ Resultant doses for loss of load shall not exceed 10 CFR 20 limits.

14.1.2.8.3 Results of Loss-of-Load Conditions Analysis

The unit has been designed to accommodate a loss-of-load condition without a reactor or turbine trip. Under circumstances where the external system deteriorates, as indicated by system frequency deviation, the unit will automatically disconnect from the transmission system. When this occurs, a runback signal causes an automatic power reduction to 15 percent reactor power. The runback may not be successful if the reactor high pressure setpoint is reached, at which time a reactor trip would occur. If successful, other actions that occur include:

A. All vital electrical loads, including the Reactor Coolant Pumps, condenser circulating water pumps, condensate pumps, and other auxiliary equipment, will continue to obtain power from the unit generator. Feedwater is supplied to the steam generators by the steam-driven feedwater pumps.

B. As the electric load is dropped, the electro-hydraulic system closes the governor valves. The unit frequency will change momentarily, but the governor will rapidly restore the set frequency.

C. During closure of the turbine governor valves, steam pressure increases to the turbine bypass valve setpoint and may increase to the steam system safety valve setpoint. Steam is relieved to the condenser and to the atmosphere. Steam venting to the atmosphere occurs for a brief period following loss of load from 100 percent initial power until the turbine bypass can handle all excess steam generated. The amount of steam relieved to the atmosphere is shown in Table 14-16. Steam relief permits energy removal from the RCS to prevent a high pressure reactor trip. The initial power runback is to 15 percent reactor power, which is a higher power level than needed for the unit auxiliary load. This allows sufficient steam flow for regulating turbine speed control. Excess steam above unit auxiliary load requirements is rejected to the condenser by the turbine bypass valves.

D. During the short interval while the turbine speed is high, the vital electrical loads connected to the unit generator will undergo speed increases in proportion to the generator's frequency increase. All motors and electrical gear so connected will withstand the increased frequency.

E. After the turbine generator has been stabilized at auxiliary load and set frequency, the station operator may reduce reactor power to the auxiliary load as desired.

The loss-of-load accident does not result in fuel damage or excessive pressures on the RCS. There is no resultant radiological hazard to station operating personnel or to the public from this accident, since only secondary system steam is discharged to the atmosphere.

Unit operation with one percent defective fuel and a 1 gpm primary-to-secondary tube leak has also been evaluated for this transient. The steam relief accompanying a loss-of-load accident would not change the whole body dose. The whole body dose is primarily due to the release of xenon and krypton and is considered to be negligible. Release of these gases is not increased by the steam relief because, even without relief, all of these gases are assumed to be released to the atmosphere through the condenser vacuum pumps. The rate of release of iodine during relief would increase because the iodine is released in steam vented directly to the atmosphere rather than through the condenser and unit vent. The iodine contained in the 1 gpm primary coolant leakage is assumed to be carried off in the secondary steam flow of 5.56×10^6 lbs/hour at the rate at which it enters the secondary system. Table 14-16 gives the quantity of steam released, the activity of the iodine contained in the steam, and the resulting site boundary thyroid dose. The relative concentration factor from Section 2.3 is based on mixing of the discharge in the wake of the reactor building and a wind speed averaged over the height of the reactor building.

14.1.2.8.4 <u>Results of Complete Loss of All Unit AC Power</u>

The second power loss considered is the hypothetical case where all unit power except the unit batteries is lost. Loss of all AC power is regarded as an incredible occurrence and was not a basis for the original plant design since in addition to the normal AC power supplies, redundant fast starting emergency diesels are provided. Addition of the Alternate AC Power Source per 10 CFR 50.63 makes a loss of all AC power even more incredible. However, analysis of this hypothetical event demonstrates that even in the absence of all sources of AC power, decay heat can be removed. The sequence of events and the evaluation of consequences for this accident are:

A. A loss of power results in gravity insertion of the control rods and trip of the turbine valves.

B. After the turbine stop valves trip, excessive temperatures and pressures in the RCS are prevented by excess steam relief through the main steam line safety valves and the atmospheric dump valves (turbine bypass valve steam relief is lost due to loss of power to the condenser cooling water circulating pumps). Excess steam is relieved until the RCS temperature is below the saturation temperature for the steam generator corresponding to the pressure setpoint of the atmospheric dump valves. Thereafter, the atmospheric dump valves are used to remove decay heat.

C. The RCS flow decays without the occurrence of fuel damage. Decay heat removal after coastdown of the Reactor Coolant Pumps is provided by the natural circulation characteristics of the system. This capability is discussed in the loss-of-coolant-flow evaluation (Section 14.1.2.6).

D. The Condensate Storage Tank provides emergency feedwater to the steam generators with the EFIC system raising the water level in the steam generators at a controlled rate until the 26 foot natural circulation setpoint is reached. The Condensate Storage Tank minimum inventory is 107,000 gallons, when only Unit 1 is aligned, or 267,000 gallons when both units are aligned.

E. The turbine-driven emergency feed pump normally takes suction from the Condensate Storage Tank and is driven by steam from either or both steam generators. The Emergency Feedwater System is discussed in Section 10.4.8. All required valves in the Emergency Feedwater System can be operated automatically since they are DC powered.

The following is a description of the necessary loads which are connected to the station batteries and which would be operable following a loss of AC power:

A. Four inverters supplying the necessary Nuclear and Non-Nuclear Instrumentation, Reactor Protection System, Engineered Safeguards Actuation System, and Emergency Feedwater Initiation and Control system.

B. Emergency lighting panels.

C. DC distribution panels.

D. DC Motor Control Centers

The above loads provide sufficient power for indication and control to maintain the reactor in a safe shutdown condition for a minimum period of two hours with an expected period of four hours. In the event a longer battery life is required, certain redundant loads can be disconnected.

In view of the foregoing sequence, the loss of all unit power does not result in fuel damage or excessive pressure in the RCS.~~ There is no resultant radiological hazard to plant operating personnel or to the public from this accident, since only secondary system steam is discharged to the atmosphere.~~

~~This transient has been evaluated further under conditions where the plant is assumed to have been operating with both one percent failed fuel and a 1 gpm tube leakage in one steam generator. This operation continues until decay heat can be removed by the steam generator with no tube leakage, and the atmospheric dump valve associated with the leaking generator is closed. This results in the following sequence of events:~~

~~A. It is assumed that the operator further opens the atmospheric dump valves 10 minutes after the loss of power.~~

~~B. Cooling down at the maximum available rate requires an additional 45 minutes to reach a temperature below the saturation temperature corresponding to the setpoint pressure for the steam safety valve having the lowest setting.~~

~~C. The steam generator with tube leakage is then completely isolated by closing its atmospheric dump valve, and the other steam generator is used to remove decay heat.~~

~~As in the loss of load transient evaluation, the whole body dose does not change because of steam relief. The total integrated thyroid dose is shown in Table 14-17. The activity of the iodine contained in the steam was calculated by the same method used in the loss-of-load accident above.~~

14.1.2.9 Turbine Overspeed

14.1.2.9.1 Background

There is a very low probability that the turbines used at ANO-1 will experience a major structural failure of a rotating part resulting in missile-like pieces leaving the turbine casing (see SAR References 1 through 13).

This is based upon:

A. Present manufacturing techniques - factory inspection and test procedures ensure sound discs with mechanical properties equal to or exceeding the specified levels.

B. Redundant control system - the main speed governing system will normally hold the turbine speed within set limits. An overspeed trip device backed by a redundant overspeed trip device provides three lines of protection in all.

C. Routine testing - testing of the main steam valves and the overspeed trip devices while the unit is carrying load.

D. Turbine Disc Inspection consisting of: (1CAN098109)

   1. Inspection of new discs at the first refueling outage or before any postulated crack would grow to more than ½ the critical depth;

   2. Discs previously inspected to be free of cracks or that have been repaired to eliminate all indications should be per Item 1 above, calculating crack growth from the time of the last inspection; and

   3. Discs operating with known and measured cracks should be reinspected before ½ the time calculated for any crack to grow to ½ the critical crack depth.

   NOTE: Inspection schedules may be varied to coincide with scheduled outages.

E. Use of fully integrated LP rotor - During the 1R8 Refueling Outage, a fully integrated type rotor was installed in the "1" section of the low pressure turbine. This rotor differs from the original "2" section of the low pressure turbine in that the original "2" rotor was a built up design with shrunk on "discs" and the fully integrated design is a one piece forging design (see section F). Westinghouse Electric Corporation's position is that the missile generation criteria for the shrunk on wheels does not apply to the fully integrated design because a failure for the fully integrated rotor would assume a situation where the rotor reaches a high enough overspeed to cause the centrifugal stresses to exceed the material strength. Calculations performed by Westinghouse show that the required overspeed cannot be reached even if loss of load occurs at full load conditions. The amount of steam entering the turbine from the time the load is lost to the time the stop valves close is insufficient to drive the turbine to the required overspeed. If the valves didn't close, other turbine components would fail (such as last stage blades, generator wedges, bearings if high vibration occurs, etc.) at speeds below the rotor burst speed and again eliminate the potential for any turbine missiles.

14.2 STANDBY SAFEGUARDS ANALYSIS

14.2.1 SITUATIONS ANALYZED AND CAUSES

This section presents an analysis of accidents in which one or more of the protective barriers are not effective and standby safeguards are required. All accidents evaluated are based on the rated core power level of 2,568 MWt. Table 14-18 summarizes the potential accidents studied.

14.2.2 ACCIDENT ANALYSIS

14.2.2.1 Steam Line Failure

14.2.2.1.1 Identification of Cause

Analyses were performed to determine the effects and consequences of loss of secondary coolant due to a double-ended steam line rupture. The main steam header piping (24″ and 36″) between the main steam block valves and the high pressure turbine stop valves, and the B31.1.0 portion of the main steam piping system (8″) between the atmospheric dump valve isolation valves and the dump valves themselves, is designed and constructed to meet ANSI B31.1.0 requirements with 100 percent volumetric examination of welds. The portion that penetrates the reactor building out through the main steam block valves is designed and constructed to meet the requirements of ANSI B31.7, Class II, or later appropriate ASME Section III Code sections provided that they have been reconciled. In addition, the main steam line from the steam generator outlet to the turbine has been analyzed and found adequate to withstand seismic loadings. Consequently the probability of a break in these lines is considered very low.

14.2.2.1.2 Reactor Protection Criteria

The criteria for reactor protection for this accident are:

A. The core will remain intact for effective core cooling.

B. Loss of reactor coolant boundary integrity resulting from steam generator tube failure due to the loss of secondary side pressure and resultant temperature gradients will not occur.

C. Resultant doses will not exceed 10 CFR 50.67~~100~~ limits.

14.2.2.1.3 Analysis and Results

14.2.2.1.3.1 Accident Dynamics

The loss of secondary coolant due to a failure of a steam line between the steam generator and the turbine causes a decrease in steam pressure and thus places a demand on the control system for increased feedwater flow. Increased feedwater flow, accompanied by steam flow through the turbine stop valves and the break, lowers the average reactor coolant temperature. The Emergency Feedwater Instrumentation and Control (EFIC) system (see Sections 7.1.4 and 7.2.4 is designed to protect against the consequences of a simultaneous blowdown of both steam generators. Upon detection of a steam line break, the EFIC system automatically initiates action to isolate each affected steam generator by closing its main steam isolation valve (MSIV) and its main feedwater isolation valve (MFIV).

14.2.2.1.4 Resultant Doses

The resultant doses from this accident are calculated by assuming that:

A. The unit has been operating with a maximum of 1 gpm steam generator tube leakage.

B. The unit has been operating at the maximum primary and secondary activity limits allowed by Technical Specifications~~with 1-percent defective fuel rods~~.

C. The steam line break occurs between the reactor building and a turbine stop valve.

D. Reactor coolant leakage into the faulted steam generator continues for 251.8~~4.5~~ hours until the RCS can be cooled down and depressurized and the leakage terminated.

E. Either an accident-initiated iodine spike (GIS) occurs or a pre-existing iodine spike (PIS) exists~~The steam line break isolation and control portion of the EFIC system does not function at all~~.

The steam line failure is assumed to result in the release of the activity contained in the steam generator inventory, the activity contained in the feedwater, and the activity contained in the reactor coolant leakage. (See Table 14-21.) The iodine, primarily resulting from reactor coolant leakage in the cooldown period following the steam line break, is assumed to be released directly to the atmosphere. Atmospheric dilution is calculated using the relative concentration developed in Section 2.3. Using these assumptions, TEDE~~the total integrated~~ dose~~s to the thyroid~~ ha~~ve~~s been calculated~~.~~ (s~~S~~ee Table 14-21~~.~~). Th~~e~~ise dose~~s~~ ~~are~~is less than~~several orders of magnitude below~~ the acceptance criteria~~guideline value~~ of 10 CFR 50.67~~100~~.~~ These conclusions were also found to be bounding of plant operation with the replacement OTSGs (Reference 94).~~

14.2.2.1.5 Building Pressure

The resultant mass and energy release to containment are taken from the above analysis using TRAP2 results (see Figures 14-21F and 14-21G) and are summed in Table 14-19b. A detailed thermal-hydraulics analysis was performed using the blowdown data in the ANO-1 DBA COPATTA model described in Section 14.2.2.5.5. The results from this analysis are shown in Figures 14-21H and 14-21I, and summarized in Table 14-20. A peak reactor building pressure of 51.1 psig occurs at 74 seconds. The peak pressure is within the DBA pressure of 54.0 psig and the reactor building design pressure of 59 psig. The MSLB temperature profile, although it exceeds the DBA temperature profile for less than 3 minutes early in the transient, is short lived and is considered bounded by the DBA profile.

Parameters used in the MSLB containment analysis along with those which are different from that assumed in DBA analysis are given in Table 14-19d. A high-high containment pressure setpoint of 36.7 psig is assumed for reactor building spray actuation. The spray response time is assumed to be 105.868 seconds based on offsite power being available as was determined above to be the most limiting case (see Section 14.2.2.1.3.3). Reactor building coolers are assumed to start at 300 seconds with the performance curve given in Figure 14-110. Only one train of sprays and coolers are modeled. This is conservative due to the single failure of the MFIV already assumed in developing the blowdown data.

At the end of Cycle 19, the original OTSGs were replaced. In support of Cycle 20 operation, an evaluation of the containment pressure/temperature response with the replacement OTSGs for LOCAs and MSLBs was performed and is documented in Reference 94. It was concluded that the current post-LOCA response would remain bounding for the replacement OTSGs. For the steam line break, the containment pressure response with the replacement OTSGs was also bounded by the current analysis. The post-MSLB temperature response with the replacement OTSGs would be worse. EOI has adopted NUREG-0458 into the ANO-1 licensing basis which recognizes that the post-MSLB atmosphere may become superheated, but the temperature spike is of such short duration that the thermal lag of any SSC inside containment will not increase significantly. Consequently, the initial temperature peak does not define operating limits on any system, structure, or component (SSC) and the long-term containment temperature (which is essentially the saturation temperature) dominates the temperature response of SSCs. Therefore, as long as the peak MSLB pressure is less than the peak pressure following a LOC~~CO~~A, the temperature response of SSCs will still be defined by the LOCA.

14.2.2.1.6 Conclusions

The ANO-1 plant response to a double-ended steam line break with a failure of the main feedwater isolation valve on the affected side has been shown to be acceptable. The analysis has shown the acceptability of a hot zero power moderator temperature coefficient of -3.5×10^{-4} (Δ k/k)/°F. The predicted maximum return to power assuming a conservative core kinetics model is below that necessary to exceed fuel design limits. The maximum temperature differential that occurs in the steam generator does not produce excessive stresses, and the integrity of the steam generator is maintained. The resultant doses are within acceptable limits.

14.2.2.2 Steam Generator Tube Failure

14.2.2.2.1 Identification of Cause

The resultant doses associated with steam generator tube leakage and subsequent release to the environment are evaluated in the preceding sections. The complete severance of a steam generator tube has also been evaluated. For this occurrence, activity contained in the reactor coolant would be released to the secondary system. Some of the radioactive noble gases and iodine would be released to the atmosphere through the main steam safety valves, atmospheric dump valves, and the condenser air removal system.

14.2.2.2.2 Reactor Protection Criteria

The criteria for reactor protection for this accident are:

A. Resultant doses shall not exceed 10 CFR 50.67~~100~~ limits.

B. Additional loss of reactor coolant boundary integrity shall not occur due to resultant temperature gradients.

14.2.2.2.3 Analysis and Results

In analyzing the consequences of this failure, the following sequence of events is assumed to occur (input parameters are shown in Table 14-22 and results are summarized in Table 14-23):

A. A double-ended rupture of one steam generator tube occurs with unrestricted discharge from each end.

B. The initial leak rate exceeds the normal makeup to the RCS, and system pressure decreases. No initial operator action is assumed, and a low RCS pressure trip will occur.

C. Following reactor trip, the RCS pressure continues to decrease until HPI is actuated. The capacity of the HPI is sufficient to compensate for the leakage and maintains both pressure and volume control of the RCS. Thereafter, the reactor is assumed to be cooled down and depressurized at 100 °F per hour until isolation of the affected steam generator can be achieved.

D. Following reactor trip, offsite power is lost and~~the turbine stop valves will close. S~~since a reactor coolant-to-secondary system leak has occurred, steam line pressure will increase, opening the ~~steam bypass valves to the condenser and briefly opening the~~ main steam safety valves. ~~The bypass valves actuate at a lower pressure than do the steam safety valves. The majority of the reactor coolant that leaks as a result of the tube failure is condensed in the condenser.~~ The fission products escaping from the main steam safety valves and the condensate are released to the atmosphere.

E. After the RCS temperature has decreased to a value that corresponds to a saturation pressure which is below the main steam line safety valves setpoint, the affected steam generator can be isolated. Cooldown on~~f~~ the unaffected steam generator continues using an atmospheric dump valve until the temperature is reduced to less than 280 °F. Thereafter, cooldown to ambient conditions is continued using the Decay Heat Removal (DHR) system. The initial leak rate is conservatively assumed to continue during the entire depressurization time.

F. The operator will receive early notification that a primary to secondary leak has occurred by the radiation alarm on the reactor console which is initiated by the radiation monitor on the air ejector. The operator does not have to make a judgment quickly since the analyses assumes that no action is taken until 20 minutes after the tube rupture. Thus there is sufficient time available before starting cooldown and depressurization for the operator to obtain samples from the steam generator and to definitely determine from radioactivity and chemical analyses which generator contains the leak. For tube leaks smaller than the 435 gpm leak analyzed in the steam generator tube rupture accident, the operator has even more time to identify that a tube leak has occurred and to determine the affected steam generator.

G. Either an accident-initiated iodine spike (GIS) occurs or a pre-existing iodine spike (PIS) exists.

The radioactivity released during this accident is assumed to be discharged both through the main steam safety valves and atmospheric dump valves to the environment and through the turbine bypass to the condenser and then out the condensate vacuum pump exhaust. A gas-to-liquid partition factor of 10^4 is assumed for the iodine in the condenser (See References 28 and 33), but noble gases are assumed to be released directly to the atmosphere. TEDE dose results are~~The total dose to the body from all the xenon and krypton released is~~ given in Table 14-23. ~~The corresponding dose to the thyroid is also tabulated.~~ The atmospheric dilution is calculated using the 2-hour relative concentrations developed in Section 2.3.

14.2.2.3 Fuel Handling Accident

14.2.2.3.1 Identification of Cause

Spent fuel assemblies are handled entirely under water. Before refueling, the boron concentrations of the reactor coolant and the fuel transfer canal water above the reactor are increased so that, with all control rods removed, the k_{eff} of the core would not exceed 0.99. The fuel assemblies are stored under water in the spent fuel storage pool; the storage racks have a safe geometric spacing. Under these conditions, a criticality accident during refueling is not considered credible. Mechanical damage to the fuel assemblies during transfer operations is possible but improbable. A mechanical damage type of accident is considered the maximum potential source of activity release during refueling operations.

14.2.2.3.2 Reactor Protection Criterion

The criterion for reactor protection for this accident is that resultant doses shall not exceed the acceptance criteria of 10 CFR 50.67~~25 percent of 10 CFR 100 limits~~.

14.2.2.3.3 Methods of Analysis

The assumptions made for this analysis are shown in Table 14-24. The reactor is assumed to have been shut down for 72~~100~~ hours, since Technical Specifications prohibit fuel handling operations prior to this time. It is further assumed that the cladding of six rows of fuel rods in the assembly, 82 of 208, suffers mechanical damage.

Since the fuel pellets are cold, only the gap activity is released, and consists of ~~10 percent of the total noble gases other than Kr^{85},~~ 30 percent of the KrR^{85}, ~~and~~ 12 percent of the I-131~~total radioactive iodine~~ in the damaged rods, and 10 percent of all other isotopes. Radioactive decay of the fission product inventory during the interval since shutdown and commencement of fuel handling operations is considered.

14.2.2.3.4 Results of Analysis

The gases released from the fuel assembly pass upward through the spent fuel storage pool water before reaching the atmosphere of the fuel handling building. The gas is assumed to pass through 23 feet of water, and 99.5 percent of the iodine released from the fuel assembly is assumed to remain in the water. No retention of the noble gases is assumed. The radionuclides released during the fuel handling accident are assumed to enter the atmosphere directly without filtration.~~The fuel handling building is ventilated, and discharge is through charcoal filters to the unit vent.~~ The atmospheric dilution is calculated using the 2-hour relative concentration developed in Section 2.3. Dose conversion factors consistent with FGR 11 and FGR 12~~ICRP 30~~ were utilized.

~~An additional case was analyzed to determine the offsite dose consequences of a fuel handling accident in containment with the personnel airlock and/or equipment hatch open during refueling. For this case the radionuclides released during the fuel handling accident are assumed to enter the atmosphere directly without filtration. The radionuclide source terms, release mechanism and pool scrubbing credited for this case were identical to the filtered release case.~~

The parameters used to analyze the fuel handling accident are given in Table 14-24. Table 14-25 gives the TEDE~~total integrated~~ doses at the exclusion distance and low population zone~~for the whole body and the thyroid~~

14.2.2.4.6 <u>Conclusions</u>

The hypothetical rod ejection accident has been investigated in detail at two different initial reactor power levels: rated power and zero power. Both BOL and EOL conditions were considered. The results of the analysis prove that the reactivity transient resulting from this accident will be limited by the Doppler effect and terminated by the Reactor Protection System with no serious core damage or additional loss of the coolant system integrity. Furthermore, it has been shown that an ejected rod worth greater than 1.52 percent Δk/k would be required to cause a pressure pulse, due to prompt dispersal of fragmented fuel and zirconium-water reaction, of sufficient magnitude to cause rupture of the pressure vessel, whereas the maximum rod worth as shown in Table 14-26 is about a factor of two less.

As a result of the postulated pressure housing failure associated with the accident (see Section 14.2.2.4.1), reactor coolant is lost from the system. The rate of mass and energy input to the reactor building is considerably lower than that subsequently reported for the smallest rupture size considered in the loss of coolant analysis (see Section 14.2.2.5.5). The maximum hole size resulting from a rod ejection is approximately 2.76 inches. This lower rate of energy input results in a much lower reactor building pressure than those obtained for any rupture sizes considered in the Loss of Coolant Accident (LOCA). Reactor building leakage is conservatively assumed to occur at the rate associated with the peak calculated pressure for the design basis loss of coolant accident, 0.2% volume per day for the first 24 hours, and 0.1% per day thereafter.~~ It is estimated that approximately 50 percent of any leakage will be through the Penetration Room Ventilation System.~~

The resultant doses from this accident are calculated assuming that all fuel rods undergoing DNB release all of their gap activity to the reactor coolant. Subsequently, this gap activity ~~and the activity in the reactor coolant from operation with one percent defective fuel pins~~ is released to the reactor building or the steam generators via primary-secondary leakage. For the case of a BOL rod ejection of the maximum rod worth of 0.65 percent Δk/k at rated power, the fuel rods that experience DNB are assumed to fail, releasing ~~gaseous~~ activity ~~to the reactor building~~ as shown in Table 14-31.

Fission product activities released to the reactor building atmosphere for this accident are calculated using the methods described in NRC Regulatory Guide 1.183~~discussed in Chapter 11~~. The ~~thyroid and whole body (gamma + beta)~~ doses were also calculated per Regulatory Guide 1.1834. The TEDE~~total integrated 2-hour~~ doses at the exclusion distance ~~to the thyroid and to the whole body~~ can be seen in Table 14-31. Also shown in Table 14-31 are the TEDE doses at the Low Population Zone (LPZ) and in the control room~~for 30-day exposure~~. No iodine removal by the spray or plateout on reactor building surfaces was assumed. These doses are less than the acceptance criteria of 10 CFR 50.67~~well within the guideline values of 10 CFR 100~~.

C. Post-Blowdown Margins

Following the time of peak pressure for the DBA, the adequacy of the reactor building design can be demonstrated by an energy margin defined as the difference between the energy capability of the reactor building (see Table 14-48) and the calculated energy content at any given time. Figure 14-65 shows this margin as a function of time for the vapor region only and for the sump plus vapor region. At the time of peak pressure the margins are 21.6 x 10^6 Btu for the vapor region and 27.1 x 10^6 for the entire reactor building. At 1,100 seconds the margins are 67.7 x 10^6 Btu and 84.9 x 10^6 Btu, respectively.

The vapor region energy margin can be related to the potential energy release of a hypothetical zirconium-water reaction. Using a reaction energy of 2,800 Btu/lbm zirconium (see Reference 19), reaction of 100 percent of the core zirconium would generate 140 x 10^6 Btu. If all the hydrogen liberated by this 100 percent metal-water reaction were burned and generated heat at a rate of 2,350 Btu/lbm zirconium, the total energy generated would be 258 x 10^6 Btu. Thus, the vapor region energy margin at the time of peak pressure could be associated with a 15 percent zirconium-water reaction or a 8 percent zirconium-water reaction with hydrogen combustion. At 1,100 seconds, the associated reactions would be 48 percent and 26 percent, respectively.

14.2.2.5.5.6 Conclusions

The pressure transient results indicate that, even with the conservative assumptions employed in the analyses, a margin of about 9.3 percent exists between the reactor building structure design pressure of 59 psig and the maximum calculated pressure of 54.0 psig. It may be concluded from the analyses of the LOCA that the reactor building design is adequate to withstand the postulated release of the reactor coolant and associated energy sources without exceeding the design pressure. Furthermore, the reactor building design has ample margin exceeding the energy releases considered.

Reactor building equipment environmental qualifications have been acceptably demonstrated for the long term conditions predicted by the DBA analysis.

14.2.2.5.6 Resultant Doses From a LOCA

The resultant doses from a LOCA are calculated by assuming that the activity associated with the gap of all fuel rods is released to the reactor building atmosphere. The timing of releases is modeled as specified in NRC Regulatory Guide 1.183~~While perforation of fuel cladding will require some time, it is conservatively assumed that all the fuel rods release their gap activity to the reactor building~~. The activity in the coolant was ~~also~~ evaluated ~~and was found~~ to be less than one percent of the gap activity and was~~can~~ therefore ~~be~~ neglected.

The activity released to the reactor building from the gaps of all fuel rods is tabulated in Table 14-49.

~~Half of the iodine released is assumed to plate out on exposed surfaces in the reactor building. The other half is assumed to remain in the reactor building atmosphere where it is available for leakage. No~~ ESF leakage equal to twice that described in Section 14.2.2.5.7 as outside the sealed rooms is assumed to occur in this analysis.~~ The sodium hydroxide in the reactor building spray reduces the airborne iodine as described in Section 14.2.2.6. Of the iodine available for leakage, two percent has been assumed to be unavailable for removal by the spray. The iodine removal constants used are described in Section 14.2.2.6.~~

~~The resultant doses due to the maximum break size LOCA are given in Table 14-49.~~

~~14.2.2.6 Maximum Hypothetical Accident~~

~~14.2.2.6.1 Description of the Accident~~

~~In order to demonstrate that the operation of Arkansas Nuclear One - Unit 1 does not produce undue risk to the public under any accident conditions, the dose that would be received at the exclusion distance and the low population zone from a release of radioactivity larger than any which could actually occur is calculated. The calculations assume a maximum hypothetical fission product release as described in TID-14844 (Reference 49). All of the noble gases, half of the iodine, and one percent of the solid fission products in the core are assumed to be released to the reactor building. Half of the released iodine is assumed immediately to plate out on surfaces within the reactor building, however, so that only one quarter of the core inventory of iodine remains in the reactor building atmosphere. This is consistent with NRC Regulatory Guide 1.4 guidance.~~

The maximum core power of Arkansas Nuclear One, Unit 1 is 2568 MWt. Fission product activities were calculated based on a ~~slightly higher (~1%)~~ power level of 2619.36~~2596~~ MWt to account for a 2% power uncertainty.~~ Table 14-50 shows the isotopic information used to obtain release activities.~~ A reactor building leak rate of 0.2 percent volume for the first day and 0.1 percent volume per day thereafter was assumed for containment leakage.

The TEDE~~thyroid, whole body, and beta skin~~ offsite dose calculations were calculated using the RADTRAD~~Bechtel~~ code~~ LOCADOSE~~. Dose conversion factors ~~for whole body and beta skin~~ were obtained ~~primarily~~ from FGR 11 and FGR 12~~RG 1.109 (See Calc. #89-E-0164-06, Attachment 4, "Correspondence from Robert G. Omen, Chief Nuclear Engineer, Bechtel to Richard Harris, Entergy", dated November 9, 1992)~~.

~~14.2.2.6.2 Iodine Removal~~

The sodium hydroxide in the reactor building spray reduces the airborne iodine. Table 14-51 lists the iodine removal constants used in this analysis. It is assumed that 0.154 percent of the iodine is organic, 95 percent is particulate, and 4.85~~91~~ percent is elemental. These numbers are consistent with Regulatory Guide 1.1834. Also, one spray header was assumed to be operating.

~~14.2.2.6.3 Offsite Thyroid Dose~~

~~The equation used to calculate the thyroid dose is that from TID-14844,~~

$$\cancel{D = B_T * X/Q_T * \sum_{i=1}^{5} (DCF)_i * Q_{iT}}$$

~~where:~~

~~D = thyroid dose, rem~~

~~$(DCF)_i$ = thyroid dose conversion factor for isotope i inhaled, rem/Ci~~

~~B_T = breathing rate: 3.47E-4 m^3/sec for the first eight hours, 1.75E-4 m^3/sec for eight to 24 hours and 2.32E-4 m^3/sec thereafter~~

~~X/Q_T = relative concentration factor with wind speed averaged over the 190-foot height of reactor building for time interval T, sec/m^3~~

~~Q_{iT} = Curies of isotope i released during time interval T.~~

The following relative concentration factors, X/Qs, are based on windspeeds averaged over the 190-foot height of the reactor building as discussed in Section 2.3.6.2.

0-2 hr, exclusion distance	6.8E-4 sec/m^3
0-8 hr, LPZ	1.1E-4 sec/m^3
8-24 hr, LPZ	1.1E-5 sec/m^3
1-4 day, LPZ	4.0E-6 sec/m^3
4-30 day, LPZ	1.3E-6 sec/m^3

The r~~R~~esultant doses due to a LOCA are shown in Table 14-49~~52. The 10 CFR 100 limit is 300 rem~~.

~~14.2.2.6.4 Offsite Whole Body Dose~~

~~The equation used to calculate the whole body dose from airborne radioisotopes in a semi-infinite cloud is:~~

$$\cancel{D = X/Q_T * \sum_{i=1}^{16} (DCF)_i * Q_{iT}}$$

~~where:~~

~~D = whole body dose, rem~~

~~(DCF)$_i$ = whole body dose conversion factor for isotope i, rem-m^3/Ci-sec~~

~~X/Q = Same as defined above~~

~~Q$_{iT}$ = Same as defined above~~

~~The whole body dose is shown in Table 14-52. The 10 CFR 100 limit is 25 rem.~~

~~14.2.2.6.5 Offsite Beta Skin Dose~~

~~The equation used to calculate the beta skin dose is:~~

$$\cancel{D = X/Q_T * \sum_{i=1}^{16} (DCF)_i * Q_{iT}}$$

~~where:~~

~~D = beta skin dose, rem~~

~~(DCF)$_i$ = beta skin dose per curie of isotope i, rem-m^3/Ci-sec~~

~~X/Q = Same as defined above~~

~~Q = Same as defined above~~

~~The beta skin dose is shown in Table 14-52~~

14.2.2.5.7~~6.6~~ Effects of Engineered Safeguards Systems Leakage during the Loss of Coolant~~Maximum Hypothetical~~ Accident

The Reactor Building Spray System pumps and LPI pumps are located in sealed rooms of the auxiliary building through which air does not circulate. Cooling is accomplished by a closed cycle ventilation system which blows room air over cooling water coils. Therefore iodine leaking from these pumps is not exhausted through the plant vent by the ventilation system. A flow path does exist from LPI and the Reactor Building Spray Pumps through the penetration rooms and into the Reactor Building. Leakage from portions of this flow path outside the sealed rooms has been evaluated to assess the dose impact. Offsite dose estimates from containment and ES leakage are included in the TEDE dose calculation results reported in Section 14.2.2.5.6~~shown in Table 14-52~~.

Iodine leaking from the HPI pumps and portions of the HPI System flow path is not contained in sealed rooms. This leakage has been evaluated to assess the impact upon the ~~MHA~~ doses even though recirculation through the HPI System in the piggyback mode is expected only for certain small break LOCAs. The additional dose from HPI System leakage, using source terms consistent with the minimal fuel damage expected during small break LOCAs, was determined to be less than 0.04 rem thyroid for both the 2 hour exclusion distance and 30 day low population zone dose. Therefore, no significant offsite doses result from these sources, and the radiation released is as low as practicable.

14.2.2.5.8~~6.7~~ Control Room Doses

The dose to the control room operator from reactor building and ES leakage has been assessed. The Emergency Air Conditioning and Filtration Systems provided for the Control Room are described in Section 9.7.2.1. Iodine efficiencies of 95% for the recirculation filters (99% for particulate) and 99% for the outside filtered air used for control room pressurization are assumed. Unfiltered inleakage is assumed to be 82~~10~~ cfm. The 30 day integrated TEDE dose to the ~~thyroid of a~~ control room operator due to a LOCA~~from reactor building and ES leakage~~ is 3.77~~18.93~~ Rem.

14.2.2.7 Waste Gas Tank Rupture

In this accident, it is assumed that a waste gas tank ruptures releasing the waste gas it contains into the auxiliary building. The radioactive waste gas is then assumed to be carried out the plant vent by the Auxiliary Building Ventilation System. In this analysis, it is assumed the plant vent remains open. In addition, no decay of radioisotopes is assumed after the waste gas tank rupture occurs.

The maximum inert gas activity which could accumulate in a single waste gas tank is given in Table 11-10. In addition the tank would also contain traces of radioiodine. The quantity of radioiodine was calculated using the maximum coolant activities in Table 11-5, a partition factor of 10^{-4} and hydrogen removal from the coolant of 55 cc/l. The maximum amount of iodine that could be found in a waste gas tank is listed in Table 14-53.

Table 14-1

ABNORMALITIES AFFECTING CORE AND COOLANT BOUNDARY

Event		Analysis Assumptions		Effect
Uncompensated Operating Reactivity Changes		Automatic control system inoperative or unused		Change in reactor system average temperature. Automatic reactor trip if uncompensated. No equipment damage or radiological hazard.
Startup Accident		Uncontrolled single-group and all-group rod withdrawal from subcriticality with the reactor at zero power. Only high flux and high pressure trips were used to terminate the accident.		Power rise terminated by negative Doppler effect, control rod inhibit on short period, high Reactor Coolant System pressure, or overpower. No equipment damage or radiological hazard.
Rod Withdrawal Accident at Rated Power Operation		Uncontrolled single-group and all-group rod withdrawal with the reactor at rated power. Only high flux and high pressure trips were used to terminate the accident.		Power rise terminated by overpower trip or high pressure trip. No equipment damage or radiological hazard.
Moderator Dilution Accident		Uncontrolled addition of unborated water to the Reactor Coolant System due to failure of equipment designed to limit flow rate and total water addition.		Slow change of power terminated by reactor trip on high temperature or pressure. During shutdown a decrease in shutdown margin occurs, but criticality does not occur. No radiological hazard.
Cold Water Accident		Two Reactor Coolant Pumps started with reactor at 60% of rated power and end-of-life conditions.		Power and pressure transient produced by increase in flow does not result in a reactor trip. No equipment damage or radiological hazard.
Loss of Coolant Flow		Reactor Coolant System flow decreases because of mechanical or electrical failure in one or more Reactor Coolant Pumps.		None. Reactor is protected by the flux-imbalance-flow and power-pump trip. No radiological hazard.
Stuck-out, Stuck-in, or Dropped-in Control Rod		Maximum worth control rod dropped into core with the reactor at rated power, end-of-life condition.		None. Subcriticality can be achieved if any one rod is stuck out. If stuck in or dropped in, continued operation is permitted if effect on power peaking is not.severe. No radiological hazard.
Loss of Electric Power		A blackout condition or a complete loss of all station power is considered. ~~One percent defective fuel and a 1 gpm steam generator~~		Possible power reduction or reactor trip, depending on condition. Redundancy provided for safe shutdown. ~~See~~

		~~tube leakage are assumed.~~		~~Table 14-16 for resultant doses.~~

Table 14-14

NATURAL CIRCULATION CAPABILITY

Time After Loss of Decay Heat Power, s	Decay Heat Core Power, %	Natural Circulation Core Flow Available, % full flow	Flow Required for Removal, % full flow
3.6×10^{1}	5	4.1	2.3
2.2×10^{2}	3	3.3	1.2
1.2×10^{4}	1	1.8	0.36
1.3×10^{5}	0.5	1.2	0.20

Table 14-15

DROPPED ROD ACCIDENT PARAMETERS

Moderator Coefficient, (Δk/k)°F	-4.0×10^{-4}
Doppler Coefficient, (Δk/k)°F	-1.3×10^{-5}
Control Rod Worth at Rated Power, % Δk/k	0.65
Control Rod Drop Time to Full Insertion, s (insertion rate = 0.325% Δk/k/s)	2.0

Table 14-16

LOSS-OF-LOAD ACCIDENT PARAMETERS AND RESULTS

Steam Relieved to the Atmosphere, lb	205,000
Steam Venting Time, min	3
Relative Concentration at Exclusion Distance, s/m^3	6.5×10^{-4}
Iodine Released During Relief (in Iodine-131 dose equivalent Curies)	3.8×10^{-2}
Total Integrated Thyroid Dose at Exclusion Distance, rem	1.2×10^{-2}

~~**Table 14-17**~~

~~**LOSS OF ALL AC POWER ACCIDENT PARAMETERS AND RESULTS**~~

~~Steam Relieved to Atmosphere, lb~~	~~203,900~~
~~Relative Concentration at Exclusion Distance, s/m^3~~	~~6.5×10^{-4}~~
~~Steam Generator Isolation Time, min~~	~~55~~
~~Iodine Released to Atmosphere (in Iodine-131 dose equivalent Curies)~~	~~7.1×10^{-1}~~
~~Total Integrated Thyroid Dose at Exclusion Distance, rem~~	~~2.4×10^{-1}~~

Table 14-18

SITUATIONS ANALYZED FOR STANDBY SAFEGUARDS ANALYSIS

Event	Analysis Assumptions	Effect
Steam Line	Reactor coolant leakage into the faulted steam generator continues for 251.8~~4.5~~ hours following reactor operation at Technical Specification activity limits~~with 1% defective fuel~~ and 1 gpm total steam generator tube leakage.	Reactor trips following a large rupture. See Table 14-21 for resultant doses.
Steam Generator Tube Failure	Reactor coolant leakage into the faulted steam generator continues for 34 minutes following reactor operation Technical Specification activity limits~~with 1% defective fuel~~.	Reactor automatically trips if leakage exceeds normal makeup capacity to Reactor Coolant System. See Table 14-23 for resultant doses.
Fuel Handling Accident	Gap activity is released from six rows of fuel rods in one assembly while in spent fuel storage pool or fuel transfer canal. No retention of noble gases and only 99.5% retention of iodine is considered.	See Table 14-25 for resultant doses.
Rod Ejection Accident	All fuel rods that experience DNB are assumed to release their total gap activity to the reactor coolant ~~(following operation with 1% defective fuel)~~.	Some fuel cladding failure. See Table14-31 for resultant doses.
Loss-of-Coolant Accident	The design of the ECCS is based on the double-ended rupture of the 36-in. diameter Reactor Coolant System pipe. The reactor building design is based on the 5.0 ft^2 rupture. Environmental effects are based on the release of all the gap activity.	Cladding temperature remains below 2,200 °F. See Table14-49 for resultant doses. See Table 4-43 for summary of reactor building pressure analysis.
~~Maximum Hypothetical Accident~~	~~Release of 100% noble gases, 50% iodine, a 1% solid fission products.~~	~~See Table 14-52 for resultant doses.~~
Waste Gas Tank Rupture	A tank is assumed to contain the gaseous activity evolved from degassing all of the reactor coolant following operation with 1% defective fuel.	See Section14.2.2.7 for resultant doses.

Table 14-20

SUMMARY OF STEAM LINE FAILURE ANALYSIS

Maximum Thermal Power During Transient, %	100
Maximum Return to Power After Trip, %	33
Minimum Subcritical Margin, % Δk/k	0.0084
Peak Reactor Building Pressure, psig (occurs at 74 s)	51.1
Peak Reactor Building Temperature, °F (occurs at 67 s)	386
Maximum Tube Stress, psi	7,350

Table 14-21

RESULTANT DOSES FROM A STEAM LINE FAILURE

~~Activity Released to the Atmosphere~~	
Source Terms~~Equivalent Ci ^{131}I~~	See Table 14-50~~4.73~~
Faulted~~Weight of Feedwater and~~ Steam Generator Mass~~Water~~, lbm	60~~2~~,0~~6~~00
Duration Primary-Secondary Leak Rate, gpm (per Steam Generator)	0.5
~~Reactor Coolant Leakage, gal~~	~~270~~
Relative Concentration at Exclusion Distance, s/m^3	6.8~~5~~ x 10^{-4}
TEDE~~hyroid~~ Doses~~ at Exclusion Distance~~, rem (GIS)	~~1.6~~
Exclusion Distance	2.07
Low Population Zone	1.05
Control Room	3.72
TEDE Doses, rem (PIS)	
Exclusion Distance	0.45
Low Population Zone	0.19
Control Room	1.84

Table 14-22

STEAM GENERATOR TUBE FAILURE INPUT PARAMETERS

Initial Leak Rate, gpm (faulted steam generator)	435
Duration Leak Rate, gpd (intact steam generator)	150
Normal Makeup Rate, gpm	70
High-Pressure Injection Setpoint, psig	1,500
~~Assumed Defective Fuel, %~~	~~1~~

Table 14-23

SUMMARY OF STEAM GENERATOR TUBE FAILURE ANALYSIS

Low-Pressure Trip Occurs at, min	11
Time to Isolation of Faulted Steam Generators~~Total Depressurization Time of Reactor Coolant System~~, min	34
Reactor Coolant Leakage through Faulted Steam Generator~~During Depressurization~~, ft^3	1,977
Time to Isolation of Intact Steam Generator, hr~~Activity Released to Atmosphere~~	237.8
Source Terms~~Noble Gases, equiv Ci ^{133}Xe~~	See Table 14-50~~25,605~~
~~Iodine, equiv Ci ^{133}I~~	~~13.91~~
TEDE~~Total Integrated~~ Doses, rem (GIS)~~ at Exclusion Distance~~	
Exclusion Distance~~Thyroid, rem~~	1.26~~4.64~~
Low Population Zone~~Whole Body, rem~~	0.23~~1.25 x 10^{-1}~~
Control Room	1.00
TEDE Doses, rem (PIS)	
Exclusion Distance	2.20
Low Population Zone	0.37
Control Room	2.33
Relative Concentration at Exclusion Distance, s/m^{-3}	6.8~~5~~ x 10^{-4}

Table 14-24

FUEL HANDLING ACCIDENT PARAMETERS

Fuel Batch Average Burnup for Peak Assembly, Mwd/ton	61,050~~60,000~~
Power Level During Operation, MW (including 2% uncertainty)	2619.36~~2855~~
Radial Peaking Factor	1.8
Decay Time, hrs	72~~100~~
~~Filter Efficiencies for Iodine Removal~~	
~~Organic, %~~	~~70~~
~~Inorganic, %~~	~~90~~
Relative Concentration at Exclusion Distance, sec/m^3	6.8~~5~~ x 10^{-4}
Pool Decontamination Factors	
Organic Iodine	1
Inorganic Iodine	286~~133~~
Noble Gases	1
Iodine GAP Composition	
Inorganic, %	99.8~~7~~5
Organic, %	0.1~~2~~5
Fraction of Assembly Activity in GAP	
Iodine-131, %	12
Krypton-85,~~-~~ %	30
Other Isotopes, %	10
~~Noble Gases Other than Krypton -85, %~~	~~10~~
Number of Damaged Pins	82

Table 14-25

FUEL HANDLING ACCIDENT DOSES

TEDE~~Total Integrated~~ Doses~~ at Exclusion Distance~~ for Fuel Handling Accident in Spent Fuel Pool or~~(Filtered Release)~~

~~Thyroid, Rem~~	~~10.4~~
~~Whole Body, Rem~~	~~0.3~~

~~Total Integrated Dose at Exclusion Distance for Fuel Handling Accident~~ in Reactor Containment, rem ~~Building~~ (Unfiltered Release)

Exclusion Distance~~Thyroid, Rem~~	1.40~~69.1~~
Low Population zone~~Whole Body, Rem~~	0.25~~0.3~~
Control Room	1.00

Table 14-26

ROD EJECTION ACCIDENT PARAMETERS

Worth of Ejected Rod	
Rated Power, No Xenon, % $\Delta k/k$	0.40
Rated Power, With Xenon, % $\Delta k/k$	0.40
Hot, Zero Power, Critical, % $\Delta k/k$	0.23
Rated Power, Maximum Worth, % $\Delta k/k$	0.65
Rod Ejection Time, s	0.15
Rated Power Level, MWt	2,568
Reactor Trip Delay Time	
High Flux Trip, s	0.3
High-Pressure Trip, s	0.5
Control Rod Drive Trip Time to 2/3 Insertion, s	1.4

Table 14-31

RESULTANT DOSES FROM ANALYSIS OF THE ROD EJECTION ACCIDENT

Source Terms~~Radioactivity released to Reactor Building~~ from fuel rods experiencing DNB

Isotope	Core Inventory~~Activity,~~ [Curies]
Kr-83m	8.77E+06
Kr-85	9.61~~4~~ x $10^{\cancel{4}5}$
Kr-85m	1.90~~6.87~~ x $10^{\cancel{3}7}$
Kr-87	3.73~~4~~ x $10^{\cancel{3}7}$
Kr-88	5.01~~1.23~~ x $10^{\cancel{4}7}$
Xe-131m	7.55~~1.13~~ x $10^{\cancel{4}5}$
Xe-133	1.4~~1~~8 x $10^{\cancel{6}8}$
Xe-133m	4.60~~1.32~~ x $10^{\cancel{4}6}$
Xe-135	3.51~~4.53~~ x $10^{\cancel{3}7}$
Xe-135m	3.09~~85~~ x $10^{\cancel{3}7}$
Xe-138	1.27 x 10^{8}
I-130	1.36 x 10^{6}
I-131	7.22~~1.83~~ x $10^{\cancel{5}7}$
I-132	1.05~~2.62~~ x $10^{\cancel{4}8}$
I-133	1.48~~3.95~~ x $10^{\cancel{4}8}$
I-134	1.67~~2.46~~ x $10^{\cancel{3}8}$
I-135	1.41~~25~~ x $10^{\cancel{4}8}$
Cs-136	2.98 x 10^{6}
Cs-137	9.88 x 10^{6}
Cs-138	1.38 x 10^{8}
Rb-86	1.29 x 10^{5}

Reactor Building Leak Rate

0.2%/day on the first day
0.1%/day thereafter
~~50% of containment leakage is processed by the penetration room ventilation system~~

Relative Concentrations, sec/m^3

0-2 hour, exclusion distance	6.8 x 10^{-4}
0-8 hour, low population zone	1.1 x 10^{-4}
8-24 hour, low population zone	1.1 x 10^{-5}
1-4 day, low population zone	4.0 x 10^{-6}
4-30 day, low population zone	1.3 x 10^{-6}

TEDE~~Two-Hour~~ Doses, rem (containment release path)~~ at Exclusion Distance:~~

2-hour, Exclusion Distance~~Thyroid, Rem~~	4.73~~6.266~~
30-day, Low Population Zone~~Whole Body, Rem~~	2.28~~0.012~~
30-day, Control Room	3.40

TEDE~~Thirty-Day~~ Doses, rem (primary-secondary release path)~~ at Low Population Zone:~~

2-hour, Exclusion Distance~~Thyroid, Rem~~	3.03~~5.025~~
30-day, Low Population Zone~~Whole Body, Rem~~	1.64~~0.009~~
30-day, Control Room	4.95

Table 14-49

RESULTANT DOSES FROM ~~MAXIMUM BREAK SIZE~~ LOCA

Core Power, MWt: 102% of 2,596

~~Total Fission Product Inventory in All Fuel Pin Gaps Released to Reactor Building Atmosphere (Based on lifetime-averaged thermal flux)~~

~~Isotope~~	~~Inventory Curies/MWt~~
~~$^{83}Kr^m$~~	~~3.5×10^0~~
~~$^{85}Kr^m$~~	~~1.91×10^1~~
~~^{85}Kr~~	~~2.68×10^2~~
~~^{87}Kr~~	~~1.04×10^1~~
~~^{88}Kr~~	~~3.42×10^1~~
~~$^{131}Xe^m$~~	~~3.14×10^1~~
~~$^{133}Xe^m$~~	~~3.67×10^1~~
~~^{33}Xe~~	~~3.29×10^3~~
~~$^{135}Xe^m$~~	~~1.07×10^1~~
~~^{135}Xe~~	~~1.26×10^1~~
~~^{138}Xe~~	~~1.0×10^1~~
~~^{131}I~~	~~5.1×10^2~~
~~^{132}I~~	~~7.3×10^1~~
~~^{133}I~~	~~1.1×10^2~~
~~^{134}I~~	~~6.85×10^0~~
~~^{135}I~~	~~3.48×10^1~~

Isotope	Core Inventory [Curies]	Isotope	Core Inventory [Curies]	Isotope	Core Inventory [Curies]
Kr-85	9.61E+05	**Sb-127**	6.56E+06	**Ce-143**	1.12E+08
Kr-85m	1.90E+07	**Sb-129**	2.01E+07	**Ce-144**	1.05E+08
Kr-87	3.73E+07	**Te-127**	6.52E+06	**Np-239**	1.39E+09

Kr-88	5.01E+07	**Te-127m**	1.16E+06	**Pu-238**	1.93E+05
Xe-131m	7.55E+05	**Te-129**	1.88E+07	**Pu-239**	2.51E+04
Xe-133	1.48E+08	**Te-129m**	3.66E+06	**Pu-240**	3.88E+04
Xe-133m	4.60E+06	**Te-131m**	1.40E+07	**Pu-241**	9.82E+06
Xe-135	3.51E+07	**Te-132**	1.02E+08	**Am-241**	1.02E+04
Xe-135m	3.09E+07	**Sr-89**	7.25E+07	**Cm-242**	2.71E+06
Xe-138	1.27E+08	**Sr-90**	7.47E+06	**Cm-244**	1.99E+05
I-130	1.36E+06	**Sr-91**	8.78E+07	**La-140**	1.32E+08
I-131	7.22E+07	**Sr-92**	9.40E+07	**La-142**	1.15E+08
I-132	1.05E+08	**Ba-139**	1.32E+08	**Nb-95**	1.34E+08
I-133	1.48E+08	**Ba-140**	1.28E+08	**Nd-147**	4.70E+07
I-134	1.67E+08	**Mo-99**	1.35E+08	**Pr-143**	1.11E+08
I-135	1.41E+08	**Rh-105**	7.25E+07	**Y-90**	7.75E+06
Cs-134	1.46E+07	**Ru-103**	1.14E+08	**Y-91**	9.53E+07
Cs-136	2.98E+06	**Ru-105**	7.64E+07	**Y-92**	9.51E+07
Cs-137	9.88E+06	**Ru-106**	4.19E+07	**Y-93**	1.07E+08
Cs-138	1.38E+08	**Tc-99m**	1.18E+08	**Zr-95**	1.29E+08
Rb-86	1.29E+05	**Ce-141**	1.23E+08	**Zr-97**	1.23E+08

Table 14-49 (continued)

Reactor Building Leak Rate

0.2%/day on the first day
0.1%/day thereafter

Relative Concentrations, sec/m^3

0 - 2 hour, exclusion distance	6.8×10^{-4}
0 - 8 hour, low population zone	1.1×10^{-4}
8 - 24 hour, low population zone	1.1×10^{-5}
1 - 4 day, low population zone	4.0×10^{-6}
4 - 30 day, low population zone	1.3×10^{-6}

Iodine removal constant (See Table 14-51)

~~Elemental (91%)~~	~~11.5/hr~~
~~Organic (4%)~~	~~0/hr~~
~~Particulate (5%)~~	~~2.6/hr~~

*TEDE Doses, Rem

~~Thyroid~~

2 hour, exclusion distance	10.49~~7.01~~
30 day, low population zone	2.56~~2.66~~
30 day, control room	3.77

~~Whole Body~~

~~2 hour, exclusion distance~~	~~1.65×10^{-2}~~
~~30 day, low population zone~~	~~1.06×10^{-2}~~

~~Skin~~

~~2 hour, exclusion distance~~	~~1.6×10^{-2}~~
~~30 day, low population zone~~	~~1.4×10^{-2}~~

* 50% of containment leakage is processed by the penetration ventilation system. Doses include the dose assessed from 782 cc/hr ESF leakage from components located outside sealed rooms.

Table 14-50

~~**TOTAL FISSION PRODUCT INVENTORY IN CORE**~~
~~(based on lifetime average thermal flux)~~

~~Isotope~~	~~Half-Life~~	~~Fission Product Inventory, Ci/MWt~~	~~Gamma Energy per Disintegration, MeV~~
~~^{131}I~~	~~8.05 d~~	~~2.51×10^4~~	~~0.374~~
~~^{132}I~~	~~77 hr[(a)]~~	~~3.81×10^4~~	~~2.390~~
~~^{133}I~~	~~20.3 hr~~	~~5.63×10^4~~	~~0.510~~
~~^{134}I~~	~~52.0 m~~	~~6.58×10^4~~	~~1.938~~
~~^{135}I~~	~~6.68 hr~~	~~5.10×10^4~~	~~2.443~~
~~$^{83}Kr^m$~~	~~1.86 hr~~	~~2.85×10^3~~	~~0.001~~
~~$^{85}Kr^m$~~	~~4.4 hr~~	~~8.41×10^3~~	~~0.151~~
~~^{85}Kr~~	~~10.76 y~~	~~3.08×10^2~~	~~0.002~~
~~^{87}Kr~~	~~76 m~~	~~1.54×10^4~~	~~1.375~~
~~^{88}Kr~~	~~2.80 hr~~	~~2.33×10^4~~	~~2.353~~
~~$^{131}Xe^m$~~	~~11.8 d~~	~~2.13×10^2~~	~~0.003~~
~~$^{133}Xe^m$~~	~~2.26 d~~	~~1.22×10^3~~	~~0.063~~
~~^{133}Xe~~	~~5.27 d~~	~~5.06×10^4~~	~~0.030~~
~~$^{135}Xe^m$~~	~~15.6 m~~	~~1.33×10^4~~	~~0.668~~
~~^{135}Xe~~	~~9.14 hr~~	~~1.02×10^4~~	~~0.146~~
~~^{138}Xe~~	~~17.5 m~~	~~5.10×10^4~~	~~4.940~~
~~Solids~~	~~2.72 hr (0 - 2 hr)~~ ~~$t(hr)^{-0.21}$ (> 2 hr)~~	~~1.44×10^6 Ci~~	~~0.7~~

~~[(a)] Half-life of precursor, ^{132}Te, used.~~

Source Terms for MSLB & SGTR

The RCS activity presented in the following table is based on an equilibrium 1 μCi/g dose equivalent I-131 and 72/E μCi/g total. The secondary activity is based on an equilibrium 0.1 μCi/g dose equivalent I-131.

Isotope	RCS Activity (Ci)	Secondary Activity (Ci)
Kr85	4.80E+02	0.00E+00
Kr85m	1.31E+03	0.00E+00
Kr87	2.09E+03	0.00E+00
Kr88	2.93E+03	0.00E+00
Xe131m	3.78E+02	0.00E+00
Xe133	3.02E+04	0.00E+00
Xe133m	7.64E+02	0.00E+00
Xe135	1.30E+04	0.00E+00
Xe135m	1.18E+03	0.00E+00
Xe-138	3.46E+03	0.00E+00
I130	6.82E+02	1.29E+01
I131	7.33E+01	1.39E+00
I132	1.00E+03	1.90E+01
I133	6.91E+02	1.31E+01
I134	1.48E+03	2.81E+01
I135	1.22E+03	2.31E+01
Cs-134	5.11E+02	9.70E+00
Cs-136	4.03E+01	7.66E-01
Cs-137	4.22E+02	8.01E+00
Cs-138	1.00E+04	1.90E+02
Rb-86	6.43E+01	1.22E+00

Table 14-51

REACTOR BUILDING SPRAY SYSTEM EFFECTIVENESS

Parameter	1 Spray Header Operates	2 Spray Headers Operate
Spray flow, gpm	1000	2000
Effective fall ht, ft	115	115
Rx Building Free volume, ft	1,81~~3~~0,000	1,81~~3~~0,000
Mass Median diameter, microns	786~~1163~~	786~~1163~~
~~Geometric standard deviation~~	~~1.5~~	~~1.5~~
Sprayed Volume Fraction	0.89~~87~~	
Unsprayed to Sprayed Volume Mixing Rate, cfm~~hr^{-1}~~	6270~~.176~~	
Average removal rate constant, hr^{-1}		
Elemental (4.85~~91~~%)	~~11.5~~20 (injection) 10 (recirculation	
Organic (0.154%)	0	
Partic. (95%)	2.6	
Recirculation Start Time (hr)	1.18~~0~~3	

~~Table 14-52~~

~~RESULTANT DOSES FROM A MAXIMUM HYPOTHETICAL ACCIDENT~~

~~TID-14844 Source at a Reactor Core Power of 2568 MWt~~

~~Reactor Building Leak Rate:~~

~~0.2%/day on the first day~~
~~0.1%/day thereafter~~

~~Relative Concentrations, s/m~~^{3}

~~0 - 2 hour, exclusion distance~~	~~6.8 x~~ 10^{-4}
~~0 - 8 hour, low population zone~~	~~1.1 x~~ 10^{-4}
~~8 - 24 hour, low population zone~~	~~1.1 x~~ 10^{-5}
~~1 - 4 day, low population zone~~	~~4.0 x~~ 10^{-6}
~~4 - 30 day, low population zone~~	~~1.3 x~~ 10^{-6}

~~Iodine removal constant~~

~~Elemental (91%)~~	~~11.5/hr~~
~~Organic (4%)~~	~~0/hr~~
~~Particulate (5%)~~	~~2.6/hr~~

~~*Doses, Rem~~

~~Thyroid~~	
~~2 hour, exclusion distance~~	~~148.68~~
~~30 day, low population zone~~	~~52.38~~
~~Whole Body~~	
~~2 hour, exclusion distance~~	~~4.66~~
~~30 day, low population zone~~	~~1.54~~
~~Skin Body~~	
~~2 hour, exclusion distance~~	~~2.16~~
~~30 day, low population zone~~	~~0.72~~

~~* 50% of containment leakage is processed by the penetration ventilation system. The thyroid dose at the exclusion distance and at the low population zone includes the dose assessed from 391 cc/hr ESF leakage from components located outside sealed rooms.~~