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SUSQUEHANNA STEAM ELECTRIC STATION
PROPOSED AMENDMENT NO. 281 TO LICENSE NPF-14 AND PROPOSED AMENDMENT NO. 251 TO LICENSE NPF-22: APPLICATION FOR LICENSE AMENDMENT AND RELATED TECHNICAL SPECIFICATION CHANGES TO IMPLEMENT FULL-SCOPE ALTERNATIVE SOURCE TERM IN ACCORDANCE WITH 10 CFR 50.67 **Docket Nos. 50-387**
PLA-5963 **and 50-388**

In accordance with the provisions of 10 CFR 50.90 and 10 CFR 50.67, PPL Susquehanna, LLC (PPL) is submitting a request for an amendment to the licensing basis for the Susquehanna Steam Electric Station (SSES) Units 1 and 2 that supports a full implementation application of an Alternative Source Term (AST) methodology with the following exceptions. The exceptions are that the current TID-14844 accident source term will remain the licensing basis for equipment qualification, NUREG-0737 evaluations other than Control Room habitability envelope (CRHE) doses, and FSAR accidents not included in Regulatory Guide 1.183. Associated proposed Technical Specification (TS) changes, which are supported by the AST analyses, are also included in this application for a license amendment.

10 CFR 50.67, "Accident Source Term," provides a mechanism for currently licensed nuclear power reactors to replace the traditional source term used in design basis accident analyses with an alternative source term. Under this provision, licensees who seek to revise the accident source term in design basis radiological consequence analyses must apply for a license amendment under 10 CFR 50.90.

Full implementation AST analyses were performed by PPL in accordance with the guidance in Regulatory Guide 1.183, and Section 15.0.1 of the Standard Review Plan. PPL performed AST analyses for the four BWR design basis accidents identified in Regulatory Guide 1.183 that could potentially result in significant Control Room and offsite doses. These include the loss of coolant accident, the main steam line break accident, the refueling accident, and the control rod drop accident. As discussed with the SSES NRC Project Manager, the Recirculation Pump Seizure event is not included in this license amendment request (LAR) and will be submitted as a separate document at a later date. The analyses demonstrate that using AST methodologies, post-accident Control Room and offsite doses remain within regulatory acceptance limits.

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PPL proposes implementation of this proposed change through a change to the SSES licensing basis, including the TS and associated Bases. Upon approval, conforming changes will be made to the SSES Final Safety Analysis Report (FSAR) and submitted to the NRC staff in accordance with 10 CFR 50.71 as part of the regular FSAR update process.

Proposed changes in the licensing basis for SSES resulting from application of the AST include the following:

- New offsite and Control Room atmospheric dispersion factors (χ/Qs) based on site specific meteorological data collected between 1999 and 2003, the new location of the CRHE air intake and Regulatory Guides 1.145 and 1.194 revised methodologies.
- Revised CRHE unfiltered inleakage from 10 cfm to 510 cfm.
- New AST analyses performed in accordance with the guidance in Regulatory Guide 1.183 for the four design basis accidents: loss of coolant accident, the main steam line break accident, the refueling accident, and the control rod drop accident.
- Revised TS Section 1.1 definition of Dose Equivalent I-131.
- Revised TS Section 3.1.7 to credit use of the Standby Liquid Control (SLC) System to buffer suppression pool pH to prevent iodine re-evolution following a postulated design basis loss of coolant accident (DBA LOCA).
- Revised TS Section 3.7.3 concerning Control Room Emergency Outside Air Supply System (CREOASS) to reflect Improved Standard Technical Specifications Change Traveler (TSTF-448).
- Add TS Section 5.5.13 concerning CREOASS to reflect Improved Standard Technical Specifications Change Traveler (TSTF-448) concerning Control Room Habitability Program.

Table 5-1 of Attachment 5 provides a description of each proposed TS and TS Bases change.

In addition to revising the SSES licensing basis to adopt the AST, licensing basis changes are proposed and justified to respond to NRC Generic Letter 2003-01, "Control Room Habitability," dated June 12, 2003 (Reference 12.1) and the Technical Specification Task Force Improved Standard Technical Specifications Change Traveler TSTF-448, Revision 2 (Reference 12.2). The proposed TS (Section 5.5.13), "Control Room Habitability Program", is provided in Attachment 6 of this LAR.

In PPL Letter PLA-5916, dated 06/28/2005, PPL identified that one commitment (provide dose consequence analysis using Regulatory Guide 1.183) will be provided with the PPL AST submittal. This submittal serves to close that commitment and as such, all actions PPL committed to take in response to Generic Letter 2003-01 are complete. No new Regulatory Commitments are made herein.

The current operating license allows SSES to operate at a maximum steady-state power level of 3489 megawatts thermal (MWt). PPL is also currently engaged in an Extended Power Uprate (EPU) project to increase the maximum licensed thermal power to 3952 MWt. Therefore, the AST analyses supporting this amendment request have been performed with the core isotopic values at EPU conditions and this application for license amendment is based on that bounding core isotopic inventory.

The use of an AST results in changes in the design basis accident radiological consequences; however, the AST methodology has no direct impact on the probability or initiation of the evaluated design basis accidents. Application of AST methodology and the other changes requested by this application for a license amendment do not increase the core damage frequency or the large early release frequency.

Therefore, this request for a revision to the SSES licensing basis is not being submitted as a "risk-informed licensing action" as defined by Regulatory Guide 1.174.

Several domestic boiling water reactors (Duane Arnold, Brunswick Units 1 and 2, Grand Gulf, Hope Creek, Clinton, and Perry) have previously provided justification for the use of AST methodology utilizing a similar approach. These applications of AST methodology have been approved by NRC.

Attachment 1 to this letter contains the overall description and summary of the proposed change. Attachment 2 provides the detailed AST Safety Assessment Report supporting the proposed AST license basis change. Attachments 3 and 4 are compliance tables addressing PPL's method of conforming to the regulatory guidance of Regulatory Guides 1.183 and 1.194 respectively. Attachment 5 contains the safety assessment for the proposed TS and Bases changes and their justification. Attachment 6 provides a mark-up of the current TS. Attachment 7 provides for information a mark-up of the current TS Bases. Attachment 8 contains a list of activities to be completed before AST implementation made in other portions of the request for the proposed license amendment. Attachment 9 includes the No Significant Hazards Consideration Determination and Environmental Consideration for the proposed changes. Attachment 10 contains non-proprietary calculations that support the Safety Assessment. Attachment 11 contains a CD with the updated meteorological data used to calculate the new Control Room and offsite χ/Qs. Attachment 12 is a list of applicable references.

These proposed changes to the current licensing basis in accordance with 10 CFR 50.92 have been reviewed by the Plant Operations Review Committee and approved by the Susquehanna Review Committee. PPL has concluded that the proposed change does not involve a significant hazards consideration. PPL has also determined that the proposed change satisfies the criteria for a categorical exclusion in accordance with 10 CFR 51.22(c)(9) and does not require an environmental review. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared for this change.

Implementation of the AST is scheduled for the spring 2007. To support this schedule, PPL requests approval of this proposed License Amendment by December, 2006, with the amendment conditioned to be effective upon startup from the U2-13RIO in spring 2007. Implementation of AST is required to support the extended power uprate (EPU) implementation for which the submittal is currently being prepared and is scheduled to be submitted to NRC in the spring of 2006.

In accordance with 10 CFR 50.91(b), PPL is providing the Commonwealth of Pennsylvania with a copy of this proposed License Amendment request.

If you have any questions regarding this submittal, please contact Mr. Michael H. Crowthers at (610) 774-7766.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 10-13-05

Respectfully,

B. T. McKinney

B. T. McKinney

Attachments:

Attachment 1 - Description for the Alternate Source Term License Amendment
Attachment 2 - AST Safety Assessment Report
Attachment 3 - Regulatory Guide 1.183 Compliance Table
Attachment 4 - Regulatory Guide 1.194 Compliance Table
Attachment 5 - Safety Assessment for the Proposed Technical Specification and Bases Changes Units 1 & 2
Attachment 6 - Proposed Technical Specification Changes Units 1 & 2, (Mark-ups)
Attachment 7 - For Information - Proposed Technical Specification Bases Changes Units 1 & 2, (Mark-ups)
Attachment 8 - Activities to be Completed Before AST Implementation
Attachment 9 - No Significant Hazards Consideration Determination and Environmental Consideration for the Proposed Changes
Attachment 10 - Non-Proprietary Versions of Supporting Calculations
Attachment 11 - CD of Meteorological Data Used to Determine New χ/Qs
Attachment 12 - References

cc: NRC Region I
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Attachment 1 to PLA-5963

Description for the Alternative Source Term License Amendment

DESCRIPTION AND SUMMARY OF PROPOSED CHANGES

1.0 INTRODUCTION

PPL Susquehanna, LLC (PPL) hereby proposes to amend the licensing basis of the Susquehanna Steam Electric Station (SSES) through the full implementation application of an Alternative Source Term (AST) methodology. The exceptions are that the current TID-14844 accident source term will remain the licensing basis for equipment qualification, NUREG-0737 evaluations other than CRHE doses, and FSAR accidents not included in Regulatory Guide 1.183. The Recirculation Pump Seizure event is not included in this LAR and will be submitted as a separate document at a later date. Applicable, proposed Technical Specification (TS) and TS Bases changes, which are justified by the AST analyses, are included in this application for a license amendment.

This full implementation of AST analyses will modify the SSES licensing bases by adopting the AST methodology which replaces the current accident source term with an alternative source term as prescribed in 10 CFR 50.67 and establishes the 10 CFR 50.67 total effective dose equivalent (TEDE) dose limits as a new acceptance criterion. The AST is characterized by the composition and magnitude of the radioactive material, the chemical and physical form of the radionuclides, and the timing of the releases of these radionuclides. The current TID-14844 accident source term will remain the licensing basis for equipment qualification, NUREG-0737 evaluations other than CRHE doses and radiological consequences for FSAR accidents not included in Regulatory Guide 1.183.

The use of an AST results in changes in the design basis accident radiological consequences; however, the AST methodology has no direct impact on the probability or initiation of the evaluated design basis accidents. Application of AST methodology and the other changes requested by this application for a license amendment do not increase the core damage frequency or the large early release frequency. Therefore, this request for a revision to the SSES licensing basis is not being submitted as a "risk-informed licensing action" as defined by Regulatory Guide 1.174.

Several domestic boiling water reactors (Duane Arnold, Brunswick Units 1 and 2, Grand Gulf, Hope Creek, Clinton, and Perry) have previously provided justification for the use of AST methodology utilizing a similar approach. These applications of AST methodology have been approved by NRC.

The current operating license allows SSES to operate at a maximum steady-state power level of 3489 megawatts thermal (MWt). PPL is currently engaged in an Extended Power Uprate (EPU) project to increase the maximum licensed thermal power to 3952 MWt. Therefore, the supporting AST analyses consider the core isotopic values at EPU conditions and this application for license amendment is based on this bounding core isotopic inventory.

Regulatory Guide 1.183 recommends that changes to the FSAR that reflect the revised analyses or the actual calculation documentation be submitted to the NRC staff.

Upon issuance of a license amendment, conforming FSAR changes will be completed as required by PPL procedures and submitted to the NRC staff in accordance with the regular FSAR update process as required by 10 CFR 50.71. In lieu of providing the NRC staff with proposed FSAR changes at this time, the supporting DBA calculations are being provided in Attachments 10 and 11.

The license amendment would revise the following SSES licensing bases:

- New offsite and Control Room atmospheric dispersion factors (χ/Qs) based on site specific meteorological data collected between 1999 and 2003, the new location of the CRHE air intake and Regulatory Guides 1.145 and 1.194 revised methodologies.
- Revised CRHE unfiltered inleakage from 10 cfm to 510 cfm.
- New AST analyses performed in accordance with the guidance in Regulatory Guide 1.183 for the four design basis accidents: loss of coolant accident, the main steam line break accident, the refueling accident, and the control rod drop accident.
- Revised TS Section 1.1 definition of Dose Equivalent I-131.
- Revised TS Section 3.1.7 to credit use of the Standby Liquid Control (SLC) System to buffer suppression pool pH to prevent iodine re-evolution following a postulated design basis loss of coolant accident (DBA LOCA).
- Revised TS Section 3.7.3 concerning Control Room Emergency Outside Air Supply System (CREOASS) to reflect Improved Standard Technical Specifications Change Traveler (TSTF-448).
- Add TS Section 5.5.13 concerning CREOASS to reflect Improved Standard Technical Specifications Change Traveler (TSTF-448) concerning Control Room Habitability Program.

Implementation of the AST is scheduled for the spring 2007. To support this schedule, PPL requests approval of this proposed License Amendment by December, 2006, with the amendment conditioned to be effective upon startup from the U2-13RIO in spring 2007. Implementation of AST is required to support the extended power uprate (EPU) implementation for which the submittal is currently being prepared and is scheduled to be submitted to NRC in the spring of 2006.

2.0 REGULATORY BACKGROUND

The current SSES licensing basis for design basis accident (DBA) analysis source terms is U.S. Atomic Energy Commission Technical Information Document TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 23, 1962. This is consistent with 10 CFR Part 100, Section 11 (10 CFR 100.11), "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," for reactor siting, which contains offsite dose limits in terms of whole body and thyroid dose and further makes reference to TID-14844.

In December 1999, the Nuclear Regulatory Commission (NRC) issued 10 CFR 50.67, "Accident Source Term," which provides a mechanism for licensed power reactors to replace the traditional accident source term used in their DBA analyses with an AST.

Regulatory guidance for the implementation of these ASTs is provided in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." 10 CFR 50.67 requires a licensee seeking to use an AST to apply for a license amendment and requires that the application contain an evaluation of the consequences of affected DBAs.

As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19, "Control Room," for a loss-of-coolant accident (LOCA), main steam line break (MSLB) accident, fuel handling accident (FHA), and control rod drop accident (CRDA). The Recirculation Pump Seizure event is not included in this LAR and will be submitted as a separate document at a later date.

The accident source term is intended to be representative of a major accident involving significant core damage and is typically postulated to occur in conjunction with a large break LOCA. As a result of significant core damage, fission products are available for release into the containment environment. The proposed AST is an accident source term that is different from the accident source term used in the original design and licensing of SSES Units 1 and 2. 10 CFR 50.67, as implemented in accordance with RG 1.183, identifies an AST that is acceptable to the NRC staff for use at operating reactors.

The following regulatory requirements and guidance are also considered within this proposed license amendment:

- GDC 19, "Control Room," of Appendix A to 10 CFR Part 50, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the Control Room under accident conditions without personnel receiving radiation exposures in excess of allowable values.

- NUREG-0800, SRP 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms, Revision 0," provides guidance for the safety review of the radiological consequences of DBAs associated with implementing an AST. SRP 15.0.1 supports the guidance outlined in RG 1.183.

- NRC Generic letter 2003-01, "Control Room Habitability," requests addressees to submit information that demonstrates that the Control Room at each of their respective facilities complies with the current licensing and design bases and applicable regulatory requirements, and that suitable design, maintenance and testing control measures are in place for maintaining this compliance.

- USNRC RG 1.145, "Atmospheric Diffusion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," provides an acceptable methodology for determining site-specific relative concentrations for a range of postulated accidental releases of radioactive material to the atmosphere.

- USNRC RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," provides guidance on determining atmospheric relative concentration (χ/Q) values in support of design basis Control Room radiological habitability assessments at nuclear power plants. This document describes methods acceptable to the NRC staff for determining χ/Q values that will be used in Control Room radiological habitability assessments performed in support of applications for licenses and license amendment requests. Many of the regulatory positions presented in this guide represent substantial changes from procedures previously used to determine atmospheric relative concentrations for assessing the potential Control Room radiological consequences for a range of postulated accidental releases of radioactive material to the atmosphere. These revised procedures are largely based on the NRC sponsored computer code, ARCON96.

- TSTF-448, Revision 2, BWOG-111, R0, "Technical Specification Task Force – Improved Standard Technical Specifications Change Traveler," developed proposed changes to Technical Specifications to replace the differential pressure surveillance with a tracer gas surveillance and to institute a Control Room Habitability Program that will ensure that Control Room habitability is maintained.

On August 15, 1995, the NRC staff issued amendment 121 to Facility Operating License No. NPF-22 and amendment 151 for Facility Operating License No. NPF-14, to increase the allowable main steam isolation valve (MSIV) leakage rate and to delete the MSIV Leakage Control Systems. These amendments permitted SSES Units 1 and 2 to take credit for the Isolated Condenser Treatment Method (ICTM) for reducing the radiological consequences of MSIV leakage for a DBA LOCA. The ICTM uses the main steam drain lines to direct any MSIV leakage to the main condenser, as an alternative method for MSIV leakage treatment and the removal of the MSIV leakage control system (MSIVLCS). This drain path takes advantage of the large volume of the main steam lines (MSLs) and condenser to provide holdup and plate-out of fission products that may leak through the closed MSIVs. PPL performed evaluations and seismic verification walk downs to demonstrate that the main steam system piping and components which comprise the ICTM system were seismically rugged and are able to perform the safety function of an MSIV leakage treatment system. The seismic ruggedness evaluation was performed to demonstrate the seismic adequacy of the Turbine Building which houses the ICTM system.

The structural integrity of the Turbine Building is an important consideration to the adequacy of the alternate MSIV leakage path because a non-seismically designed Turbine Building should be capable of withstanding the earthquake without degrading the capability of the ICTM system.

These amendments reference the General Electric Company (GE) Report, NEDC-31858P-A, "Boiling Water Reactor Owners Group (BWROG) Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," Revision 2, as a basis for the acceptability of crediting the MSL piping and condenser.

3.0 SAFETY ASSESSMENT

PPL has performed a full implementation of the AST as defined in RG 1.183 with the exception that the current TID-14844 accident source term will remain the licensing basis for equipment qualification, NUREG-0737 evaluations other than CRHE doses and radiological consequences for FSAR accidents not included in Regulatory Guide 1.183. A detailed description of the AST analyses, including a safety assessment, is provided in Attachment 2. Copies of AST accident dose calculations and all calculations required to support the licensing bases changes for AST are included in Attachment 10.

The detailed Safety Assessment Report associated with AST analyses is provided in Attachment 2. This report is supplemented with RG 1.183 and 1.194 Compliance Tables presented in Attachments 3 and 4.

The basis/safety assessment associated with the proposed changes to the SSES Technical Specifications and Bases is included in Table 5-1 of Attachment 5 and supported by the Safety Assessment in Attachment 2 and the calculations in Attachment 10.

A list of Regulatory Commitments is included in Attachment 8.

A No Significant Hazards Consideration Determination and Environmental Report for the proposed changes are included in Attachment 9.

4.0 CONCLUSION

In conclusion, based on the considerations discussed above and detailed in the remainder of this submittal, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will be conducted in compliance with the Commission's regulations; and (3) the issuance of the requested license amendment will not be inimical to the common defense and security or to the health and safety of the public.

A comparison of the calculated AST doses and the RG 1.183 dose limits for the Control Room operator, the exclusion area boundary and the low population zone for the DBA LOCA, MSLB, CRDA, and the FHA/EHA are provided in the table below. Note that the calculated AST doses provide a conservative estimate of the dose consequences for the postulated event. The conservative models and methodologies that were used to calculate the dose consequences are presented in Attachment 2.

Summary of Design Basis Accident Radiological Consequences for AST – Calculated Dose Versus RG 1.183 Dose Criteria			
		Calculated Dose (Rem TEDE)	RG 1.183 Dose Criteria (Rem TEDE)
LOCA	CR Operator Dose	4.78	5
	EAB Dose	7.81	25
	LPZ Dose	3.80	25
MSLB	Case 1: Maximum Reactor Coolant Equilibrium Activity Concentration for Full Power Operation (0.2 µCi/gm DE I-131)		
	CR Operator Dose	0.04	5.0
	EAB Dose	0.010	2.5
	LPZ Dose	0.006	2.5
	Case 2: Maximum Reactor Coolant Activity Concentration for Pre-Accident Iodine Spike (4.0 µCi/gm DE I-131)		
	CR Operator Dose	0.77	5.0
	EAB Dose	2.0	25
	LPZ Dose	0.12	25
CRDA	Case 1: Full Power Operation		
	CR Operator Dose	0.49	5.0
	EAB Dose	0.19	6.3
	LPZ Dose	0.05	6.3
	Case 2: Mechanical Vacuum Pump Operating		
	CR Operator Dose	1.80	5.0
	EAB Dose	2.30	6.3
	LPZ Dose	0.18	6.3
FHA/EHA	CR Operator Dose	0.13	5.0
	EAB Dose	1.74	6.3
	LPZ Dose	0.10	6.3

Attachment 2 to PLA-5963

AST Safety Assessment Report

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Acronyms and Abbreviations

μCi/gm	micro-curies per gram
χ/Q	atmospheric dispersion factor in sec/m^3
ARM	area radiation monitor
AST	alternative source term
BWR	boiling water reactor
BWROG	Boiling Water Reactor Owners Group
CAM	continuous air monitor
cfm	cubic feet per minute
CRD	control rod drive
CRDA	control rod drop accident
CRHE	Control Room habitability envelope
CREOASS	Control Room emergency outside air supply system
CsI	cesium iodine
DBA	design basis accident
DE	dose equivalent
DF	decontamination factor
DW	drywell
EAB	exclusion area boundary
ECCS	emergency core cooling system
EDE	effective dose equivalent
EHA	equipment handling accident
EOF	emergency operations facility
EPU	extended power uprate
ESF	engineered safeguard features
FHA	fuel handling accident
FSAR	Final Safety Analysis Report
ft	feet
GDC	general design criterion
GE	General Electric
gpm	gallons per minute
GWd	gigawatt days
HEPA	high efficiency particulate air
hrs	hours
in	inch
lbm	pounds-mass
LAR	license amendment request
LOCA	loss of coolant accident
LPZ	low population zone
m/s	meters per second
MSIV	main steam isolation valve
MSL	main steam line
MSLB	main steam line break
MSIVLCS	main steam isolation valve leakage control system
MWD/MTU	megawatt days/metric tons uranium
MWt	megawatt thermal

Acronyms and Abbreviations – continued

NRC	Nuclear Regulatory Commission
OSC	operations support center
pH	hydrogen ion concentration
PPL	PPL, Susquehanna, LLC
psid	pounds per square inch differential
psig	pounds per square inch gauge
RAI	requests for additional information
RB	Reactor Building also referred to as secondary containment
Rem	roentgen equivalent man
RG	USNRC regulatory guide
sccm	standard cubic centimeters per minute
scfh	standard cubic feet per hour
scfm	standard cubic feet per minute
SDV	scram discharge volume
secs	seconds
SER	safety evaluation report
SGTS	standby gas treatment system
SLC	standby liquid control
SR	surveillance requirement
SRP	standard review plan
SSE	safe shutdown earthquake
SSES	Susquehanna Steam and Electric Station
TEDE	total effective dose equivalent
TS	technical specification
TSC	technical support center

1.0 DESCRIPTION

In accordance with 10 CFR 50.67, "Accident Source Term," a licensee may voluntarily revise the accident source term used in design basis radiological consequence analyses. Paragraph 50.67(b) requires that applications under this section contain an evaluation of the consequences of applicable design basis accidents (DBAs) previously analyzed in the plant Final Safety Analysis Report (FSAR). Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference 2), provides guidance to licensees on performing evaluations, and reanalyses as required to adopt an alternative source term (AST).

SSES has performed radiological consequence analyses of the four applicable boiling water reactor (BWR) DBAs identified in RG 1.183. These DBAs are a Loss of Coolant Accident (LOCA), a Fuel Handling Accident (FHA), a Control Rod Drop Accident (CRDA) and a Main Steam Line Break (MSLB). These analyses were performed using the guidance of RG 1.183 and Standard Review Plan (SRP) Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms" (Reference 3). The analyses were prepared, reviewed, and approved in accordance with the PPL 10 CFR 50, Appendix B Quality Assurance Program. Comparison with the guidance contained in RG 1.183 and RG 1.194 is summarized in Attachments 3 and 4 respectively of this license amendment request (LAR).

The supporting analyses consisted of the following steps:

- Determination of the AST based on plant-specific analysis of the fission product inventory.
- Application of the release fractions for the four BWR DBAs.
- Application of the deposition and removal mechanisms.
- Evaluation of suppression pool pH to ensure that the particulate iodine deposited into the suppression pool during a DBA LOCA does not re-evolve and become airborne as elemental iodine.
- Evaluation of activity transport pathways to the environment.
- Analysis of the atmospheric dispersion for the radiological propagation pathways.
- Calculation of the offsite and Control Room personnel Total Effective Dose Equivalent (TEDE).
- Evaluation of other related design and licensing bases pertaining to NUREG-0737 (Reference 5) requirements and operation of the SLC System.

The radiological dose analyses have been performed assuming reactor operation at 4032 MWt (102% of the EPU rated power level of 3952 MWt, conservatively rounded high). This results in a conservative estimate of fission product releases for operation at current licensed power of 3489 MWt.

In addition to revising the SSES licensing basis to adopt the AST, licensing basis changes are proposed and justified to respond to NRC Generic Letter 2003-01, "Control Room Habitability", dated June 12, 2003.

The proposed TS (Section 5.5.13), "Control Room Habitability Program", is provided in Attachment 6 of this LAR.

2.0 PROPOSED CHANGES

The licensing and design basis changes included in this LAR are described below. The proposed Technical Specification (TS) and Bases changes are described in Attachment 5 and a mark-up of the affected TS and Bases pages is provided in Attachments 6 and 7 respectively.

3.0 BACKGROUND

On December 23, 1999, the NRC published 10 CFR 50.67, "Accident Source Term," in the Federal Register. This regulation provides a mechanism for licensed power reactors to replace the current accident source term used in design basis accident (DBA) analyses with an alternative source term. The direction provided in 10 CFR 50.67 is that licensees who seek to revise their current accident source term in design basis radiological consequence analyses must apply for a license amendment under 10 CFR 50.90.

Regulatory Guide (RG) 1.183 and Standard Review Plan Section 15.0.1 were used by PPL in preparing the AST analyses. These documents were prepared by the NRC staff to address the use of ASTs at current operating power reactors. The RG establishes the parameters of an acceptable AST and identifies the significant attributes of an AST acceptable to the NRC staff. In this regard, the RG provides guidance to licensees for operating power reactors on acceptable applications for an AST; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on risk; and acceptable radiological analysis assumptions. The SRP provides guidance to the staff on the review of AST submittals.

Acceptance criteria consistent with that required by 10 CFR 50.67 were used to replace PPL's SSES current design basis source term acceptance criteria. The AST analyses were performed for the four BWR DBAs identified in RG 1.183 that could potentially result in Control Room and offsite doses. These include the loss of coolant accident, the main steam line break accident, the refueling accident, and the control rod drop accident.

In addition, this LAR provides the bases for resolving the non-conformance issue with the current design and licensing basis for the CREOASS. TS 3.7.3 (Control Room Emergency Outside Air Supply System) was revised and TS 5.5.13 was added in response to TSTF-448, Revision 2, BWOG-111, R0, "Technical Specification Task Force – Improved Standard Technical Specifications Change Traveler" (Reference 36).

4.0 TECHNICAL ANALYSIS

4.1 Atmospheric Dispersion Factors – Offsite and CRHE

Table 4.1-1 provides a comparison of the design inputs utilized to determine the existing licensing basis χ/Qs and the proposed AST χ/Qs, which were determined per the guidance of RG 1.183, Section 5.3 (Meteorological Assumptions).

Table 4.1-1: Design Input Comparison – Current Licensing Basis vs. AST Design - χ/Q

Parameter	CLB Parameter	AST Parameter
CRHE Supply Air Intake Location	Centered on S wall of CRHE complex, between column lines L & M and 32.3, el. 797'	SE corner Unit 2 RB roof, column lines U and 36, el. ~874'
Meteorological Data – Data Collection Requirements	NRC RG 1.23	NRC RG 1.23
Meteorological Data – Collection Time Period	1973 – 1976	1999 – 2003
Atmospheric Dispersion Model Guidance - Offsite	RG 1.3, U.S. NRC BTP (Hydrometeorological Branch), "Diffusion Conditions for Design Basis Accident Evaluations," 1/77	RG 1.145
Atmospheric Dispersion Model Guidance - CRHE	Halitsky methodology	RG 1.194 & NUREG/CR-6331 (ARCON96)

4.1.1 Atmospheric Dispersion Factors – Offsite

A detailed discussion of the design input parameters, assumptions, methodology, analysis, and results supporting the new offsite χ/Qs are provided in Attachment 10 as calculation EC-ENVR-1057, "Offsite χ/Q Values for SSES Based on 1999 – 2003 Meteorological Data." The χ/Q estimates are based on the methods described in Regulatory Guide 1.145 (Reference 6). All estimates use vertical temperature difference to determine stability classification, based on the joint frequency distribution of temperature difference categories used to summarize 5 years of SSES data into Pasquill groups for use in computing σ_y and σ_z in the diffusion equations. Results are based on evaluation of meteorological data files consisting of five years of hourly data, covering the years from 1999 to 2003.

These five years of data were selected based on the quality of the data, the quantity (i.e., recovery rate) of the data, and the representation of long term meteorological conditions and seasonal trends and meet the requirement of NRC Safety Guide 23, "Onsite Meteorological Programs" (Reference 9). The data set is consistent with RG 1.194 (Reference 10) that states that five years of hourly observations are considered representative of long term trends at most sites and that one year including all four seasons is the acceptable minimum. The five year data set used by SSES includes all four seasons for the five consecutive years.

Calculations of the offsite χ/Qs were made using the ABS Consulting, Inc. WINDOW computer program and were validated with the NRC's PAVAN computer code (Reference 37). This analysis is given in calculation EC-ENVR-1057, "Offsite χ/Q Values for the SSES Based on 1999 – 2003 Meteorological Data" and is provided in Attachment 10. Attachment 11 of this LAR contains a CD with the updated meteorological data used to calculate the new offsite atmospheric diffusion coefficients. The following table provides a comparison between CLB and the new offsite atmospheric diffusion coefficients used for the AST:

Table 4.1-2: Current Licensing Basis vs. AST Design - Offsite χ/Qs (sec/m^3)		
Parameter	CLB Parameter	AST Parameter
χ/Qs (sec/m^3)	EAB: 9.60E-04, 0-2 hr LPZ: 2.18E-05, 0-8 hr 2.82E-06, 8-24 hr 1.43E-06, 1-4 day 1.08E-06, 4-30 day	EAB: 8.30E-04, 0-2 hr LPZ: 4.90E-05, 0-8 hr 3.50E-05, 8-24 hr 1.70E-05, 1-4 day 6.10E-06, 4-30 day

4.1.2 Atmospheric Dispersion Factors - CRHE

A detailed discussion of the design input parameters, assumptions, methodology, analysis, and results supporting the new CRHE χ/Qs are provided in Attachment 10 as Calculations EC-ENVR-1058, "CRHE Accident Dispersion Factors (χ/Q)" and EC-ENVR-1059, "CRHE Accident Dispersion Factors (χ/Q) RB U2 Intake."

The atmospheric dispersion factors (χ/Q) were calculated using plant specific meteorological data and the ARCON96 (Reference 8) computer code per RG 1.194. The meteorological data files used for the CRHE includes a record of each hourly data containing a location identifier, Julian day (1-366), hour (0 to 23), low-level direction, low-level speed, stability class (1=A to 7=G), upper level direction, and upper level speed. Wind speeds are entered in tenths of a reporting unit with no decimal. Wind directions are from 1 to 360 in degrees. The five yearly files were combined into one file for ease of code execution. This file contains all the data for the SSES site from 1999-2003, which satisfies the ARCON96 requirements for having 3 to 5 years of hourly data. The meteorological tower used for collecting the data is located on the plant site. Instrumentation is provided at the 10 and 60 meter level. Attachment 11 of this LAR contains a CD with the updated meteorological data used to calculate the new CRHE atmospheric diffusion coefficients.

4.1.2.1 Infiltration

Section 3.3.3, "Infiltration Pathways", of RG 1.194 states that a χ/Q should be determined for any infiltration pathway that could result in a significant intake of radioactive contaminated air into the Control Room Habitability Envelope (CRHE). The infiltration pathways actually applicable to a particular facility will be identified via inleakage testing or CRHE inspections and surveillances.

The current licensing basis for the infiltration rate is 10 scfm, attributed to leakage through improperly sealed doors or by personnel ingress and egress, per NUREG-0800, USNRC Standard Review Plan, Section 6.4, "Control Room Habitability Systems," Reference 7.

In December of 2004, tracer gas inleakage tests were performed on the CRHE by Lagus Applied Technology, Inc. and NCS Corporation. Air inleakage rate into the CRHE and the corresponding fresh air makeup flow rate were measured with the Control Room Emergency Outside Air Supply System (CREOASS) operating in two filtered emergency pressurization modes (A and B train). Measured air inleakage rates are summarized below.

Item	Value
A Train Inleakage	150 +/- 235 scfm
B Train Inleakage	129 +/- 298 scfm

As stated in the report provided by Lagus Applied Technology, Inc. and NCS Corporation, statistically, the inleakage value for both tests is zero, since the values lie within the 95% uncertainty limits. The 95% upper confidence limit values were conservatively assumed for CRHE inleakage rate.

The infiltration rate for the four DBAs analyzed in Sections 4.4 through 4.7 is conservatively assumed to be 500 scfm at the Control Room intake (PLA-5916, Reference 39), plus an additional 10 scfm in accordance with USNRC Standard Review Plan, Section 6.4.

Since the tracer gas inleakage tests statistically yielded a CRHE infiltration rate of zero, it was determined that the most likely entry point of inleakage is at the CRHE intake. Consequently, the inleakage and intake χ/Qs are the same.

4.1.2.2 Calculations

The ARCON96 computer code (Reference 8) was used to calculate the Control Room χ/Qs. The release points to the environment evaluated for possible modeling with the ARCON96 computer code are as follows:

1. Reactor Building Unit 1 exhaust vent.
2. Reactor Building Unit 2 exhaust vent.
3. Turbine Building Unit 1 exhaust vent.
4. Turbine Building Unit 2 exhaust vent.
5. Standby Gas Treatment System exhaust vent.
6. Reactor Building Unit 1 closest distance.
7. Reactor Building Unit 2 closest distances.
8. Turbine Building Unit 1 closest distance.
9. Turbine Building Unit 2 closest distances.
10. Reactor Building Unit 1 main steam tunnel blowout panel.
11. Reactor Building Unit 2 main steam tunnel blowout panel.
12. Turbine Building Unit 1 main steam tunnel blowout panel.
13. Turbine Building Unit 2 main steam tunnel blowout panel.

Based on the layout of the SSES plant structures, exhaust vents and components and the locations of the release points relative to the location of the Control Room Habitability Envelope intake and the potential to be a source of post LOCA activity release, release points 3, 4, 5, 11 and 13 were determined to be limiting with respect to atmospheric relative concentrations for DBA Control Room dose analysis. See Figures 1 and 2 for the relative locations of the 13 release points and the CRHE outside air intake. The specific locations at which χ/Qs were determined with ARCON96 in Attachment 10 are:

3. Turbine Building Unit 1 exhaust vent.
4. Turbine Building Unit 2 exhaust vent.
5. Standby Gas Treatment System exhaust vent.
11. Reactor Building Unit 2 main steam tunnel blowout panel.
13. Turbine Building Unit 2 main steam tunnel blowout panel.

The CRHE χ/Q's for release points 3, 4 and 5 are provided in Table 4.1-3.

The ARCON96 results for release points 11 and 13 were not used in any accident analyses and are given in Attachment 10 but are not provided in Table 4.1-3. The main steam tunnel blowout panels are only pertinent release points for postulated main steam line breaks. The χ/Q used for evaluating the CRHE for these breaks is based on "puff" methodology as described in Section 4.4.4.

Table 4.1-3: AST Design – χ/Q (sec / m^3) (1)					
Release Point	CRHE χ/Q's (sec/m^3) without Occupancy Correction Factors				
Time Period	0 to 2 hours	2 to 8 hours	8 to 24 hours	1 to 4 days	4 to 30 days
	RB Unit 2 CRHE Outside Air Intake Location				
TB Unit 1 Exhaust Vent	1.24E-03	9.55E-04	3.14E-04	1.99E-04	1.73E-04
TB Unit 2 Exhaust Vent (2)	1.36E-03	1.03E-03	3.36E-04	2.20E-04	1.85E-04
SGTS Exhaust Vent (3)	1.45E-03	1.12E-03	3.55E-04	2.29E-04	2.01E-04
	Outside CRHE (for unprotected CR operator dose determination)				
TB Unit 1 Exhaust Vent	5.09E-03	4.15E-03	1.20E-03	1.16E-03	1.01E-03
TB Unit 2 Exhaust Vent	6.00E-03	4.93E-03	1.44E-03	1.38E-03	1.21E-03
SGTS Exhaust Vent	5.15E-03	4.22E-03	1.23E-03	1.19E-03	1.04E-03

Note 1: For the LOCA, the primary containment and ESF leakages to the secondary containment are released to the environment via the SGTS vent. The primary containment bypass and MSIV leakages are released to the environment via the Turbine Building exhaust vent.
Note 2: For the CRDA, the release to the environment is via the TB Unit 2 vent.
Note 3: For the FHA, the release to the environment is via the SGTS exhaust vent.

The CLB CRHE post LOCA χ/Q's were determined using the Halitsky methodology and are independent of the release point. For the CLB only the LOCA was evaluated for Control Room habitability. The values used are:

<u>CLB CRHE χ/Qs</u>:
3.32E-04 sec/m^3, 0-8 hr
1.96E-04 sec/m^3, 8-24 hr
1.27E-04 sec/m^3, 1-4 day
5.48E-05 sec/m^3, 4-30 day

4.1.2.3 Miscellaneous

Attachment 4 provides a matrix which compares the RG 1.194 regulatory position with the methodologies utilized to calculate the new CRHE χ/Qs.

The χ/Qs calculated at the CRHE air intake are based on a new location, located on the southeast corner of the Unit 2 Reactor Building roof (at column lines U and 36). The desire to utilize the NRC recommended ARCON96 code necessitated the change in the air intake location. Application of the ARCON96 code at the present CRHE air intake location results in unacceptable χ/Qs. The new CRHE location minimized CRHE doses by taking into account the locations and distances from the CRHE air intake to identified release pathways for postulated accident conditions. The new CRHE air intake will be operable prior to AST implementation.

4.2 Accident Source Terms

The SAS2H/ORIGIN-S code system (References 11 and 12) was used to generate the source terms for the ATRIUM-10 fuel used in the SSES core. The SAS2H/ORIGIN-S code comprises an advanced version of ORIGIN, and is consistent with the source term code recommendations given by the USNRC RG 1.183 for generation of alternate source terms.

The models assume a core thermal power of 4032 MWt (102% of 3952 MWt), which corresponds to an average fuel assembly power of 5.28 MWt. The fission product source terms calculated are slightly conservative (~2%) because the uranium mass assumed is higher than the actual ATRIUM-10 assembly mass. Enrichment sensitivities were evaluated to bound the range of expected U-235 enrichments in the reload fuel.,

Core source terms (full core activity inventories) were generated utilizing a core average burnup of 39 GWd/MTU. See Attachment 10, Calculation EC-FUEL-1615, "AREVA Alternate Source Term (AST) Fission Product Inventory for Atrium-10 Fuel," Appendix B, Tables 4 and 5.

For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory was used. For DBA events that did not involve the entire core, the fission product inventory of each of the damaged fuel rods was determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors were applied in determining the inventory of the damaged rods.

No adjustment to the fission product inventory was made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For the fuel handling accident postulated to occur while the facility is shutdown, a radioactive decay from the time of shutdown was modeled.

Per RG 1.183, the release fractions associated with the BWR core inventory released into containment for the DBA LOCA and non-LOCA events have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup of 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWd/MTU. Per Siemens Power Corporation EMF-85-74(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model", February 1988, including Supplement, 1(P)(A) and Supplement 2(P)(A) (Reference 13), the NRC approved licensing limit for ATRIUM-10 fuel is 62,000 MWD/MTU for extended burnup design – peak fuel rod exposure and 54,000 MWD/MTU for extended burnup design – peak bundle exposure.

4.3 Loss of Coolant Accident

A detailed discussion of the design input parameters, assumptions, methodology, analysis, and results supporting the LOCA are provided in Attachment 10 as Calculations EC-RADN-1125, "CRHE and Offsite Post LOCA Doses – AST" and EC-RADN-1129, "DBA LOCA Total Control Room Dose." Attachment 3 provides a matrix which compares the RG 1.183 regulatory position with the parameters and methodologies utilized to calculate the LOCA CRHE and offsite doses.

Table 4.3-1 provides a comparison of the design inputs utilized to determine the existing licensing basis LOCA and the proposed AST LOCA doses.

Table 4.3-1: Design Input Comparison – Current Licensing Basis vs. AST Design – LOCA

Parameter	CLB Parameter	AST Parameter
Core Thermal Power Level	3616 MWt	4032 MWt
Activity Inventory in Core Ci/MWt	18 isotopes (I, Kr and Xe)	60 dose significant isotopes used in RADTRAD
Radioisotope Decay Properties	TACT 5	RADTRAD Table 1.4.3.2-3
Activity Release to Containment	Per R.G. 1.3	Per R.G. 1.183 Table 1 (Gap & Early In-Vessel Phases Only)
Release Timing	Instantaneous	Per R.G. 1.183 Table 4
Radioiodine Chemical Species	91% Elemental 5% Particulate 4% Organic	95% Aerosol (CsI) 4.85% Elemental 0.15% Organic
Primary Containment Volume	Drywell free volume = 239,600 ft^3 Wetwell free volume = 148,590 ft^3 Total free volume = 388,190 ft^3	Drywell free volume = 239,600 ft^3 Wetwell free volume = 148,590 ft^3 Total free volume = 388,190 ft^3 Note: For the MSIV and bypass leakage pathways, activity released from the core is conservatively assumed to be mixed in the drywell free air volume only for the first 2 hours post accident. After 2 hours, credit is taken for mixing in the drywell plus wetwell free air volumes.
Primary Containment Cleanup (Natural Deposition)	None	Aerosol removal via Natural Deposition (10th percentile Powers Model)
Primary Containment Cleanup (Drywell Sprays)	No credit taken	No credit taken
Primary Containment Design Leak Rate	1%/day for duration.	1%/day for first 24 hours; 0.5%/day thereafter.
Containment Bypass Leak Rate	9 scfh	9 scfh 0 – 24 hours 4.5 scfh thereafter
Dose Conversion Factors	ICRP 30	RADTRAD Table 1.4.3.3-2

Table 4.3-1: Design Input Comparison – Current Licensing Basis vs. AST Design – LOCA

Parameter	CLB Parameter	AST Parameter
Suppression Pool Scrubbing	DF = 7.96 (> 180 seconds) for elemental and particulate iodines only	Not Credited
Secondary Containment (RB) Free Air Volume	Zone I 1,488,600 ft^3 Zone II 1,598,600 ft^3 Zone III 2,668,000 ft^3 Total Volume = 5,755,200 ft^3	Zone I 1,488,600 ft^3 Zone II 1,598,600 ft^3 Zone III 2,668,000 ft^3 Volume used = 4,156,600 ft^3 (Zone I & III)
Secondary Containment Volume Mixing Fraction	50% mixing	50% mixing
Post-LOCA RB Drawdown Time	3 minutes (used in analysis)	10 minutes (used in analysis)
RB Leakage Till End of Drawdown	100 %/day or 4000 cfm for three Zone mixing	100 %/day or 2,885 cfm for Zone I & III mixing
RB Leakage After Drawdown	0 cfm	0 cfm
SGTS Flow Rate	0 cfm, (0-30s) 10,500 cfm, draw down (30s-3min) 4,000 cfm (> 3min)	11,110 cfm for the first 10 minutes and 2,885 cfm thereafter
SGTS Filter Bed Depth	8 in. charcoal	8 in. charcoal
SGTS Filter Bed Efficiency	99% for all iodine species	99% for all iodine species
MSIV Leak Rate	300 scfh (4 lines)	300 scfh (4 lines) modeled as: 100 scfh in one assumed faulted line 66.67 scfh each in the remaining lines
MSL and Drain Line Plateout	Isolated Condenser Treatment Method (ICTM) Per GE Report NEDC-31858P-A (Reference 4)	RG 1.183, Appendix A & AEB-98-03, Appendix A (Reference 17) & J.E. Cline Letter Report (Reference 18)
Effective Condenser Volume for Each Pathway	98,601 ft^3	98,601 ft^3
Effective Removal Efficiency in Condenser for Drain Line Pathway	99.6%, effective on aerosols and elemental iodine No organic iodine removal.	99.6%, effective on aerosols and elemental iodine No organic iodine removal.
Modeled MSIV Leakage – MSL Inlet, Initial Pressure and Temperature.	Drywell Pressure = 48 psia Time dependent temperature based on post LOCA heat transfer analysis of piping cool down	Drywell Peak Accident Values MSL Pressure = 50 psia MSL Temperature = 550 °F (conservative temperature assumed for duration)
MSIV Leakage Pressure and Temperature into Condenser	Pressure = 1 atmosphere Temperature = 100 °F	Pressure = 1 atmosphere, Temperature = 100 °F (conservative temperature assumed for duration)
MSIV Source Release Timing	Instantaneous	Instantaneous
Minimum Suppression Pool Volume Post LOCA	Total: 132,000 ft^3	Total: 132,000 ft^3

Table 4.3-1: Design Input Comparison – Current Licensing Basis vs. AST Design – LOCA

Parameter	CLB Parameter	AST Parameter
ESF System Leakage Source Term to Environment	Iodine only	Iodine only
ESF Leakage into RB ESF CRD	 5 gpm 15 gpm	 5 gpm 15 gpm
ESF Leakage Outside of the RB	None	None
ESF Leakage post-LOCA Time	Begins at 0 sec – ends at 30 days	Begins at 0 sec – ends at 30 days
SSES post-LOCA Suppression Pool Maximum Temperature	< 212 °F	< 212 °F
ESF Flash Fraction	10%	10%
SP Iodine Species	91% elemental, 5% particulate, & 4% organic	97% elemental, 3% organic
Iodine Re-evolution	None assumed	None assumed since pH >7
RB Sump Iodine Species	91% elemental, 5% particulate, & 4% organic	97% elemental, 3% organic
Control Structure Habitability Envelope Total Volume	518,000 ft^3	518,000 ft^3
Control Room Free Air Volume	110,000 ft^3	110,000 ft^3
Geometry Correction Factor	$GF = 1173/V^{0.338}$ V = CR volume = 110,000 ft^3	$GF = 1173/V^{0.338}$ V = CR volume = 110,000 ft^3
CR Isolation Time	0	0
Emergency Intake Air Flow, Total into Control Structure	5810 cfm	5229 to 6391 cfm
Unfiltered Air Inleakage Ingress/Egress	10 cfm	10 cfm
Other Unfiltered Air Inleakage	0 cfm	500 cfm
Emergency Filter Bed Depth	4 in. charcoal	4 in. charcoal
Emergency Filter Bed Removal Efficiency	99%	99%
CRHE Operator Breathing Rates Offsite Breathing Rates	3.5E-04 m^3/sec (0 – 30 days) 3.5E-04 m^3/sec (0 – 8 hours) 1.8E-04 m^3/sec (8 – 24 hours) 2.3E-04 m^3/sec (1 – 30 days)	3.5E-04 m^3/sec (0 – 30 days) 3.5E-04 m^3/sec (0 – 8 hours) 1.8E-04 m^3/sec (8 – 24 hours) 2.3E-04 m^3/sec (1 – 30 days)
CRHE Operator Occupancy Factors	1.0 0-24 hrs 0.6 1-4 days 0.4 4-30 days	1.0 0-24 hrs 0.6 1-4 days 0.4 4-30 days
Dose Calculation Program	TACT 5	RADTRAD

4.3.1 Introduction and Background

SSES is a BWR/4 with a Mark II containment. The planned extended uprated power is 3952 MWt. This value is increased by 2% to 4032 MWt. The core inventory used to develop the source term for the LOCA analysis is based on an SAS2H/ORIGIN-S code run. The SSES Mark II primary containment consists of two compartments. The two compartments are connected by a vent system that allows steam released from the reactor vessel (located in the drywell) to flow into the suppression pool. The primary containment leakage is limited by TS to 1.0% by weight of containment air per day at the calculated peak accident pressure. Because of post-accident containment depressurization, this leakage rate will decrease with time. A factor of two reduction in the leak rate after 24 hours is assumed in this analysis.

The post LOCA doses include contributions from the following activity release pathways:

1. Primary containment leakage to the secondary containment.
2. Primary containment bypass leakage directly to the environment.
3. ESF leakage to the secondary containment.
4. MSIV leakage to the environment via the condenser.

The following DBA LOCA dose contributors to the CRHE are included in this analysis:

1. Contamination of the Control Room atmosphere by the intake or the infiltration of the radioactive material contained in the radioactive plume released from the facility.
2. Radiation shine from the external radioactive plume released from the facility.
3. Contamination of the Control Room atmosphere by the intake or the infiltration of the radioactive material from areas and structures adjacent to the Control Room envelope.
4. Radioactive shine from radioactive materials in building adjacent to the control structure: including containment, Reactor Building, and Turbine Building.
5. Radioactive shine from radioactive materials in systems and components inside or external to the Control Room envelope, e.g., piping, components, and radioactive material buildup on HVAC filters.

Maintaining the suppression pool pH above 7.0 serves to improve its iodine retention capability and reduces the amount of radioactive iodine available for release in the design basis LOCA. Buffering of the suppression pool by the SLC system is credited to maintain the pH of the suppression pool above 7.0. The suppression pool pH calculation is provided in Attachment 10 as EC-059-1041, "Suppression Pool pH post LOCA." The initiation of the SLC system is a manual action. The SLC system is discussed in Section 4.7.1 of this Attachment and a supporting calculation EC-053-1012, "Assessment of SLC System for Suppression Pool pH Control" is included in Attachment 10.

The radiological dose to the Control Room operators during the postulated design basis LOCA is mitigated by the integrity of the Control Room habitability envelope (CRHE), operation of the Control Room emergency outside air supply system (CREOASS), and administrative controls. The doses calculated in this AST evaluation are based on the limiting combinations of unfiltered inleakage rates and filtered intake flows coupled with conservatively selected χ/Qs.

The NRC approved computer code RADTRAD, endorsed by RG 1.183, is used to calculate the dose to the Control Room operator as well as the doses at the EAB and LPZ.

4.3.2 Source Term

The source term used for the design basis LOCA analysis is defined by the quantity, type, and timing of the release of radioactivity from a damaged reactor core to the containment. The core inventory is provided in Attachment 10, Calculation EC-FUEL-1615, "AREVA Alternate Source Term (AST) Fission Product Inventory for ARTIUM – 10 Fuel", Appendix B, Table 4. The assumed core power of 4032 MWt is the proposed licensed extended power increased by 2%. The release fractions and timing are per RG 1.183, Sections 3.2 and 3.3 respectively. Radionuclide composition and chemical form are per RG 1.183, Sections 3.4 and 3.5 respectively.

4.3.3 Mitigation

The radiological consequences of the LOCA are actively mitigated by several safety-related systems and the use of administrative controls. The CREOASS is credited for the mitigation of the dose to the Control Room operator. The isolation of the CR and the initiation of the CREOASS are automatic in response to a LOCA accident signal.

The Control Room outside air intake is normally open. Per SSES Units 1 & 2 Technical Specifications 3.7.3.4 and 5.5.7a, the CREOASS filtered intake flow ranges from 5229 cfm to 6391 cfm with positive pressure guaranteed at ≤ 5810 cfm. Dose analyses were performed at 5229 and 6391 cfm with the 5229 cfm flow rate resulting in the worst case doses.

Filtered intake flow of 5229 cfm for Control Room pressurization is assumed with CREOASS filter efficiencies of 99% for all the iodine species. No manual actions are credited in the analysis relative to the CREOASS system. Control Room unfiltered inleakage is assumed to be 510 cfm, which bounds tracer gas inleakage test results.

The SGTS is credited for the mitigation of the radiological releases. SSES Units 1 & 2 proposed Technical Specifications B3.6.4.1.4 change state that one SGTS subsystem should draw down the Reactor Building including Zones I, II and III to greater than or equal to 0.25 inch of vacuum water gauge in less than or equal to 300 seconds (5 minutes). A 10 minute Reactor Building drawdown time is conservatively used in the LOCA analysis.

Releases into the Reactor Building during the first 10 minutes are assumed to leak directly to the environment as a ground level release with no filtration. After 10 minutes, these releases are filtered by the SGTS. Per RG 1.52 (Reference 23), a SGTS charcoal filter efficiency of 99% is assumed for all species of iodine, based on an 8 inch charcoal bed.

The SSES secondary containment is mixed by a recirculation system. Per RG 1.183, Appendix A.4.4, a mixing efficiency of 50% is assumed.

The manual injection of boron via the SLC system is credited for suppression pool pH control. The maintenance of a suppression pool pH level above 7.0 is important to prevent re-evolution of iodine from the suppression pool water. This use of SLC is consistent with several other BWR submittals using AST for LOCAs. The initiation of SLC is performed from the Control Room and is not a new manual action. New procedural guidance is required to address reliance on SLC for pH control. The appropriate procedural guidance will be established during the implementation of the LAR. (See Section 4.7.1 for additional information on the SLC system and the justification for the use of SLC in this application.)

The portion of the main steam lines from the primary containment isolation valve to the point of the drain piping "tee" is used for plateout.

The method for modeling the removal of aerosols, elemental and organic iodine in the main steam line piping is based on Appendix A to AEB-98-03 (Reference 17). Only the horizontal runs of piping are considered in determining the aerosol and elemental iodine plateout. Additionally, since aerosol plateout is a mechanistic settling process, only the bottom ½ of the inside surface area of the lines is applicable for plateout. In addition, since the bottom half of a circular pipe has sides which are essentially vertical or inclined, the area for aerosol plateout is modeled as the projected area of the diameter of the pipe [diameter X length] in lieu of the actual surface area [π X diameter X length]. Another potential issue related to the phenomenon of aerosol settling is the possibility for steam condensation in the piping to wash out and re-evolve some of the settled aerosols. While the actual process would be difficult to quantify, a factor of 2 reduction in the conservatively calculated projected area is used in RADTRAD to provide additional margin. Therefore, the aerosol settling area is defined as ½ the projected area or ½ X pipe diameter X length.

The key parameter in the aerosol removal equations is the settling velocity of the material of interest. AEB-98-03 provides values for aerosol settling velocities. Early AST submittals used an aerosol settling velocity equal to the median value of the settling velocity or 0.00117 m/sec (Page A-3 of AEB-98-03). In a meeting with the NRC staff on May 24, 2005, PPL was informed that use of the median settling velocity of 0.00117 m/sec (50^{th} percentile value) was no longer acceptable to the staff. The SSES LOCA analysis conservatively uses an aerosol settling velocity equal to ¼ of 10^{th} percentile value from AEB 98-03 or 5.25E-05 m/sec [1/4 * 2.1E-04].

The elemental removal is based on the same AEB-98-03 methodology as the aerosol removal except that the elemental iodine deposition velocity based on J. E. Cline, MSIV Leakage Iodine Transport Analysis, Letter Report, 3/26/1991 (Reference 18) is used in the analysis. Re-suspension of elemental iodine is conservatively included in the analysis.

The assumed leakage rate from the primary containment, secondary containment bypass leakage, and leakage via the MSIVs are reduced by a factor of two after 24 hours into the event. Credit for reduction of leakage is consistent with the guidance in RG 1.183.

4.3.4 Radiological Transport Modeling (see Figures 3 – 8)

A simplified radiological release model and individual pathway models developed to calculate LOCA doses utilizing RADTRAD are shown in Figures 3 through 8. Specific model details and the supporting RADTRAD runs are provided in Attachment 10 as Calculation EC-RADN-1125, "CRHE and Offsite Post LOCA Doses – AST."

Various flow paths model activity transport between volumes and the environment. These flow paths are associated with volumetric flows that determine the rate at which radioactivity is exchanged between the control volumes. In addition, removal processes such as deposition in pipes and filtration are modeled within and between the control volumes, as appropriate. No credit is conservatively taken in this analysis for fission product reduction due to the initiation of the drywell sprays.

The drywell and wetwell are connected by downcomers and vacuum breakers, which allows steam that has been released into the drywell to flow to the suppression pool. Non-condensables could then collect in the wetwell air space above the pool. When the drywell pressure is reduced, a portion of these non-condensables will return to the drywell. The suppression pool scrubbing of activity carried to the suppression pool by this process is not credited in this analysis.

Per SSES Technical Specifications, Section 3.6.1.1 and Primary Containment and Technical Specification Bases, B3.6.1.1, the leakage rate for the primary containment is defined as 1% by weight of containment air per 24 hours @ 45 psig. In accordance with RG 1.183, Section 3.7: "The primary containment should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50% of the Technical Specification leak rate". Based on the significant reduction of the calculated internal pressure of the primary containment at 24 hours into the LOCA, the 50% reduction in leak rate was utilized in the analysis (Figures 5, 7, and 8).

The Reactor Building was credited for holdup, based on a minimum free volume of two zones (Zones I and III). For modeling purposes, the SGTS flow rate for Reactor Building drawdown is assumed to be at the maximum system flow rate of 11,110 cfm from SSES Units 1 & 2 Technical Specifications 5.5.7a during the drawdown period (i.e., the first 10 minutes) and then 100% per day of the post LOCA secondary containment volume (2885 cfm). The SGTS filter efficiency for all forms of iodine and for particulates is 99% (Figures 5 & 6).

Primary Containment Leakage to the Secondary Containment (Figure 5)

Credit for natural deposition in the drywell is taken. The 10th percentile Power's Aerosol Decontamination Model is conservatively used in the analysis.

Leakage from primary containment to secondary containment is modeled based on the combined drywell and wetwell volumes. The assumed leakage rate from the primary containment is reduced by a factor of two after 24 hours into the event. Credit for this reduction of primary containment leakage is consistent with the guidance in RG 1.183. Releases into the secondary containment during drawdown (the first 10 minutes after startup of SGTS) are assumed to leak directly to the environment as a ground level release with no filtration. After 10 minutes, these releases are filtered by the SGTS.

Primary Containment Bypass Leakage Directly to the Environment (Figure 7)

SSES TS SR 3.6.1.3.11 specifies that the combined leakage rate for all miscellaneous secondary containment bypass leakage pathways be equal to or less than 9 scfh. The supporting LOCA analysis is based on a secondary containment bypass leakage rate of 9 scfh for the first 24 hours. Consistent with the treatment of MSIV leakage, this leakage value was reduced by a factor of two at 24 hours. This activity leakage path is from primary containment directly to the environment and is modeled based on the drywell volume for the first 2 hours and the combined drywell and wetwell volumes for the remainder of the accident.

MSIV Leakage to the Environment via the Condenser (Figure 8)

The MSIV leakage test limits are provided in SSES Technical Specification Surveillance Requirement 3.6.1.3.12 as ≤ 100 scfh from any one valve or ≤ 300 scfh total from the four valves. This analysis assumes one MSL is faulted and the faulted line has 100 scfh flow. The remaining leakage is evenly split between the three non-faulted lines. Credit for natural deposition within the main steam lines was taken and is discussed in Section 4.3.3 above. To accommodate a postulated single failure of an MSIV to close, credit for natural deposition was taken for only three of the four steam lines.

NEDC-31858P-A, BWROG Report for Increasing Main Steam Isolation Valve Leakage Rate Limits and Elimination of Leakage Control Systems, Appendix C, Section 7 (Reference 19) provides the BWROG evaluation of the MSIV leakage flow and removal in the BWR condenser. It was concluded that the BWROG condenser removal model was clearly conservative. The BWROG methodology was based on the TID-14844 iodine species fractions of 91% elemental, 4% organic and 5% particulate. The NRC addressed this issue in a Letter from Brenda Mozafari, Division of Licensing Project Management, Office of Nuclear Reactor Regulation to Mr. J. S. Keenan, Brunswick Steam Electric Plant, Carolina Power & Light Company, Southport, North Carolina, Brunswick Steam Electric Plant, Units 1 and 2 – Issuance of Amendment RE: Alternative Source Term (TAC Nos. MB2570 and MD2571) dated 3/30/02 (Reference 20), Section 3.2.1.4, Pages 14 and 15, wherein the NRC states that use of NEDC-31858P-A condenser iodine removal efficiency methodology is bounding for the AST with its 95% aerosol iodine species and 0.15% organic iodine species.

The condenser equivalent elemental and particulate (aerosol) DF of 250 (effective filter efficiency of 99.6%) was used in approval of MSIVLCS elimination and as an input to RADTRAD for the AST LOCA analysis.

ESF Leakage to the Secondary Containment (Figure 6)

Two sources of potential leakage are included in the release model. The first is ESF system leakage directly into secondary containment. Currently, there is no TS limit for ESF leakage outside primary containment. TS 5.5.2, "Primary Sources Outside Containment" discusses a program that provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable.

The systems included in the ESF transport pathway are core spray, high pressure coolant injection, residual heat removal, reactor core isolation cooling, reactor water cleanup, standby gas treatment, scram discharge, post accident sampling (until such time as a modification eliminates the post accident sampling system penetration as a potential leak path), and containment air monitoring systems. The program includes: (1) preventive maintenance and periodic visual inspection requirements, and (2) integrated leak test requirements for each system at least once per 24 months. The provisions of SR 3.0.2 are applicable. Per station procedure, the maximum allowable limit is 5 gpm. This meets FSAR commitments 15.6.5.5.1.2 and 18.1.69. This procedure will be revised to reflect a maximum limit of 2.5 gpm (see Attachment 8 of this LAR). Consistent with RG 1.183, Appendix A, Item 5.2, the ESF leakage rate was increased by a factor of two. Leakage is assumed to start at t = 0 minutes post-accident and continues for the duration of the accident.

The second source of potential leakage (non-ESF) is through the control rod drives (CRDs) and scram discharge (SDV). The combined contribution of the CRDs and SDV is assumed to be 15 gpm and was not increased by a factor of two for this analysis per RG 1.183, Appendix A, Item 5.2. Although this source is not an ESF source, it was included for conservative reasons.

For all LOCA cases evaluated, the maximum bulk suppression pool water temperature does not exceed 212°F per Section A20 of Reference 14. Therefore, per RG 1.183, Appendix A, Section 5.5, "If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid." A flash fraction of 10% was used in the LOCA analysis.

Releases into the secondary containment during drawdown (the first 10 minutes after startup of SGTS) are assumed to leak directly to the environment as a ground level release with no filtration. After 10 minutes, these releases are filtered by the SGTS.

Control Room (Figure 4)

The activity flow path model for the Control Room models the intake and the infiltration of radioactive material contained in the radioactive plume released to the environment from the facility for CRHE doses. The Control Room air intake flow can vary between 5229 cfm and 6391 cfm. The air intake flow that results in the highest CHRE dose is used. An infiltration flow rate of 510 cfm is assumed and includes 10 cfm to account for CRHE ingress/egress. The Control Room exhaust flow rate is the sum of filtered and unfiltered incoming flow rates. The assumed filter efficiencies are 99% for all the iodine species.

The total Control Room habitability envelope free volume is used in the RADTRAD model. In order to take credit for the radiation shielding effects of the control structure floors, the EDE portion of the Control Room TEDE is based on the actual free volume of Control Room floor elevation 729'-1" and is evaluated by adjusting the RADTRAD dose by the ratio of the geometry factor for 518,000 ft^3 to the geometry factor for the Control Room free volume of 110,000 ft^3.

Summary of Release Model

The general assumptions are:

- The 10^{th} percentile Power's Aerosol Decontamination Model.
- 1.0% weight per day primary containment leakage. This leakage rate is reduced by a factor of two at 24 hours.
- 0.0134% volume per day primary containment leakage bypassing the secondary containment. This leakage rate is reduced by a factor of two at 24 hours.

- MSIV leakage based on the TS leakage limit. Credit is taken for aerosol and iodine deposition in the three intact steam lines. MSIV leakage is reduced by a factor of two at 24 hours.
- The condenser provides an equivalent elemental and particulate DF of 250.
- 20 gpm of ESF and CRD/scram discharge volume combined leakage into secondary containment (2 X 2.5 + 15.)
- A flash fraction of 10% is used for ESF leakage.
- Secondary containment drawdown time of 10 minutes.
- SGTS flow of 11,110 cfm initially, and 2,885 cfm (100%/day) after secondary containment drawdown.
- SGTS filters: 99% efficient for all species except noble gases.
- No credit for drywell sprays.
- Credit for holdup in the secondary containment, based on mixing by the recirculation system (a mixing efficiency of 50% is assumed).
- CRHE unfiltered inleakage is 510 cfm.
- Control Room air intake filters: 99% efficient for gaseous iodine, 99% for particulates.
- Suppression pool pH maintained above 7.0 due to manual injection of boron via the SLC system.
- Automatic initiation of the CREOASS and SGTS.
- No credit for suppression pool scrubbing.

4.3.5 Results – Control Room Operator Dose

The DBA-LOCA Control Room operator dose is shown in Table 4.3-2. Consistent with RG 1.183 Item 4.2.1, the calculation was performed to assess the Control Room operator dose from all sources of radiation that will cause exposure. The Control Room operator dose is the sum of the dose contributions from the nine sources presented in Section 4.3.1 and determined in Calculations EC-RADN-1125 and EC-RADN-1129 provided in Attachment 10.

Table 4.3-2
LOCA Control Room Operator Dose (rem)

	Calculation TEDE	RG 1.183 TEDE
CR Operator Dose	4.78	5

As a result of the analysis performed in calculation EC-RADN-1129, "DBA LOCA Total Control Room Dose", limited access controls are required in the CRHE to maintain the Control Room operator dose as calculated and to meet the dose acceptance requirements of RG 1.183 and 10 CFR 50.67. Access control requirements in the CRHE result from core spray pipe shine located in the Reactor Building.

Elevations 698' (computer room), 729'-1" (Control Room), and 741'-1" (TSC) are the only areas of the CRHE that require personnel to meet the occupancy requirements of RG 1.183, Section 4.2.6 during a DBA LOCA. Other areas of the CRHE require significantly less occupancy. Core spray piping is located in the adjacent RB near the northeast and southeast corners of the CRHE. A section of an Office (Room C-401) and section of the Operational Support Center (Room C-402) located on elevation 729'-1" and the Electrical Room (Room C-413) and a section of the NRC Conference Room (Room C-414) located on elevation 741'-1", must have access controls to limit personnel exposure to ≤ 0.738 rem TEDE from the direct shine from the core spray piping for the duration of the accident. This is accomplished by designating an area 5' from the CRHE east wall as a limited entry zone on elevations 729'-1" and 741'-1". Due to the location of the computer room on elevation 698', access controls are not required. Emergency planning station procedures shall be revised to incorporate access controls prior to AST implementation.

4.3.6 Results – Offsite Doses

The RADTRAD computer code was used to determine the offsite dose. Table 4.3-3 shows the proposed licensing basis dose limit compared to the regulatory limit.

Table 4.3-3
LOCA Offsite Doses (rem)

	Calculation TEDE	RG 1.183 TEDE
EAB Dose(1)	7.81	25
LPZ Dose	3.80	25

(1) The EAB dose represents the maximum 2-hour TEDE over the accident period.

4.3.7 Conclusion

The LOCA Control Room operator dose is below the 5 rem TEDE regulatory limit and the offsite doses are well below the 25 rem TEDE regulatory limit.

4.3.8 Summary of Calculation Conservatisms

- The analysis uses the effective iodine removal efficiency values based on 550°F for the duration of the accident for the main steam piping (the effective removal efficiency increases with decreasing temperature of the piping).

- Unfiltered inleakage is assumed to be 500 cfm (not including 10 cfm from ingress/egress). This amount bounds the leakage and error band from the December 2004 CRHE tracer gas inleakage tests.

- Primary containment cleanup by natural deposition assumes the 10th percentile Power's Aerosol Decontamination Model.

- No credit is conservatively taken for fission product reduction due to the initiation of the drywell sprays.

- Post LOCA secondary containment drawdown time from SSES Units 1 & 2 proposed Technical Specification B3.6.4.1.4 changes state that one SGTS subsystem should draw down the Reactor Building including Zones I, II and III or Zones I and III to greater than or equal to 0.25 inch of vacuum water gauge in less 5 minutes. A 10 minute Reactor Building drawdown time is conservatively used in this analysis.

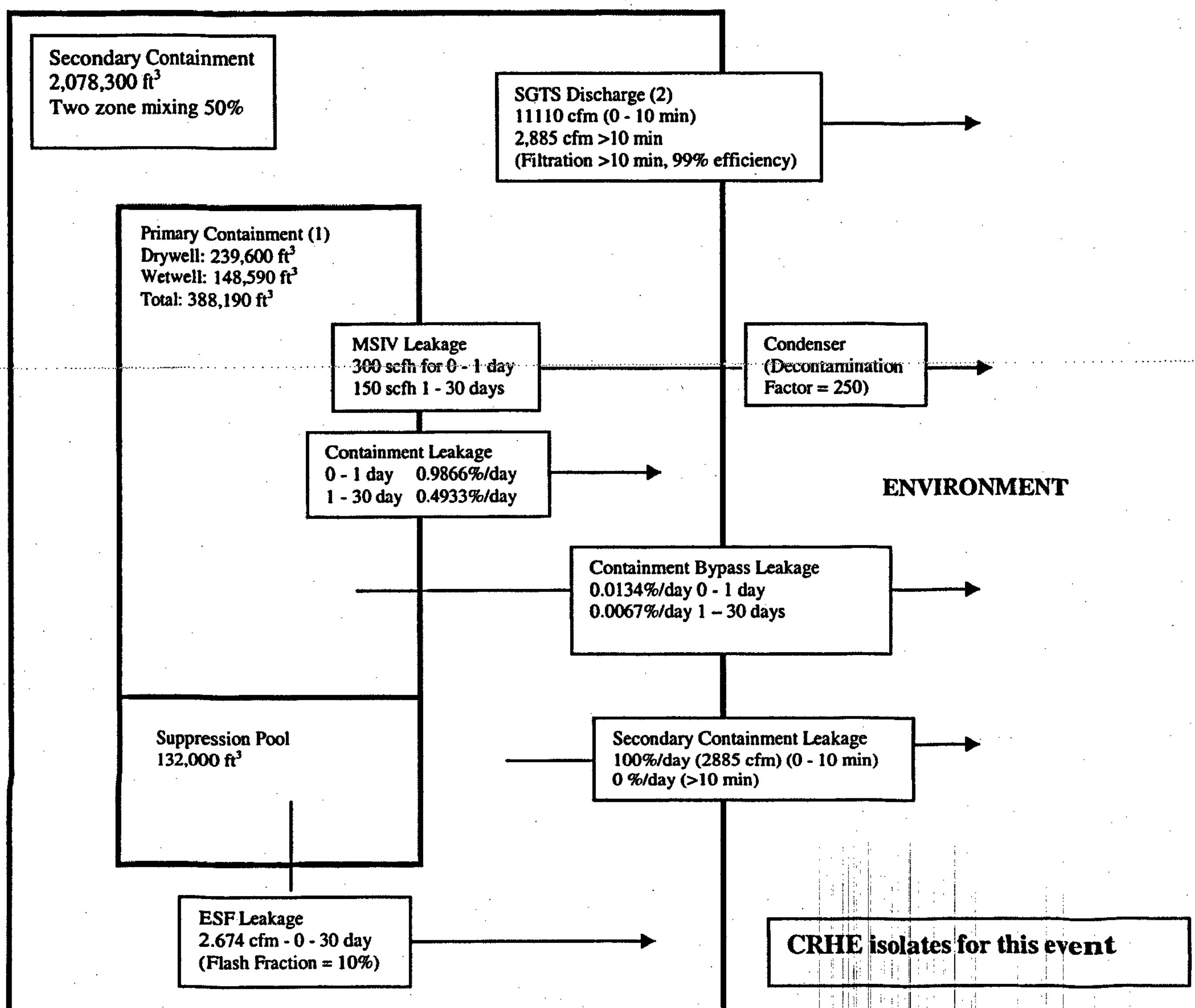
- The model for the removal of iodine in the main steam lines only credits horizontal runs of piping in determining aerosol plateout.

- The aerosol settling velocity is conservatively set equal to ¼ of the 10th percentile value from AEB 98-03.

- Only the portion of the main steam lines from the primary containment isolation valve to the point where the drain piping takes off is used for plateout. No credit is taken in the faulted line, which is assumed to be the line of greatest length.

- The activity available for release from the MSIV and containment bypass leakage paths are conservatively taken to be mixed in the drywell free volume for the first two hours. After two hours (to allow for mixing resulting from thermal – hydraulic activity), the activity is assumed to be further diluted in the drywell plus wetwell free volume.

Figure 3: A Simplified Activity Flow Path Composite Model Developed to Calculate LOCA Doses Utilizing RADTRAD

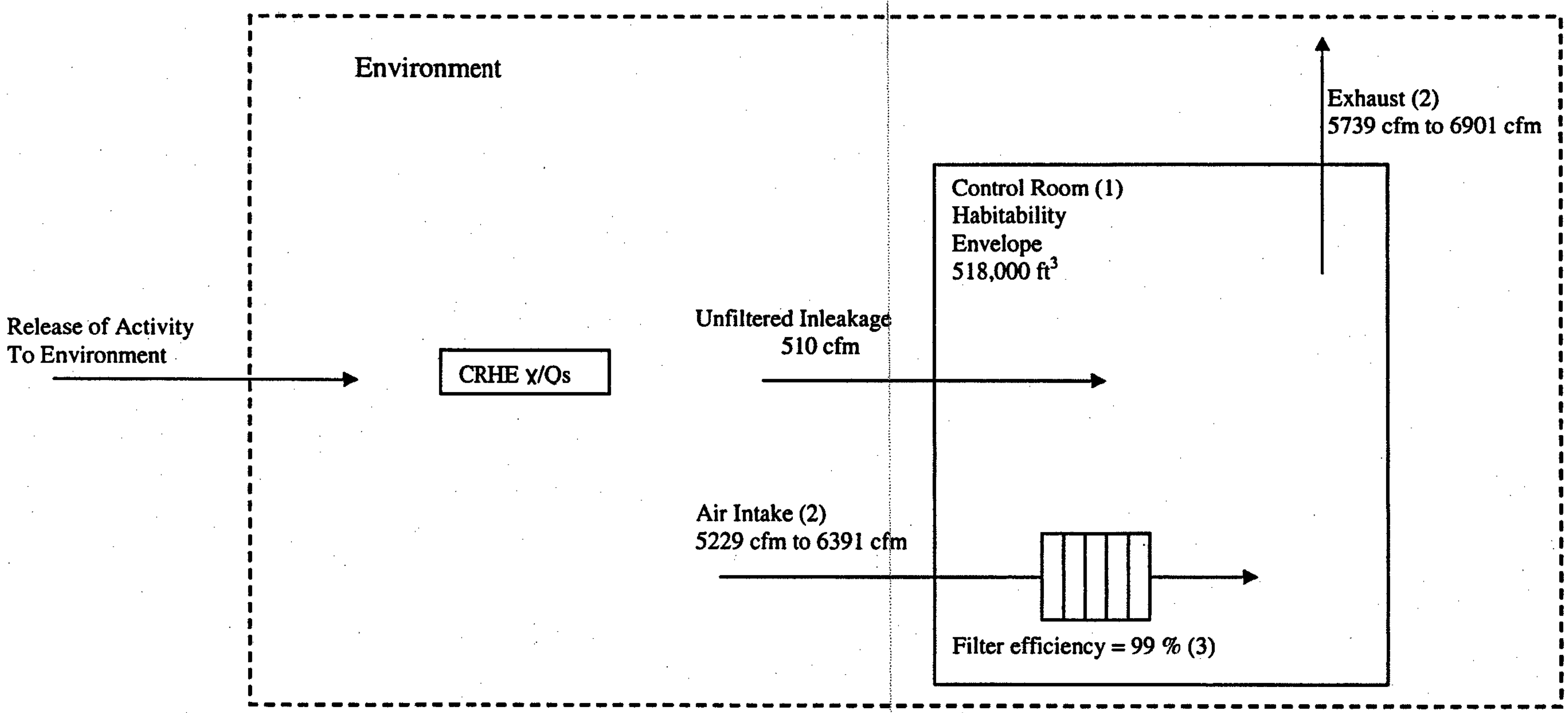


Notes:

(1) For the MSIV and bypass leakage paths, activity released from the core is conservatively assumed to be mixed in the drywell free air volume only for the first two hours post-accident. After 2 hours, credit is taken for mixing in the drywell plus wetwell free air volumes.

(2) No credit for air filtration of the SGTS discharge is taken during secondary containment drawdown (0 – 10 min).

Figure 4: A Simplified Activity Flow Path Model for the Control Room Habitability Envelope Developed To Calculate Accident Doses Utilizing RADTRAD



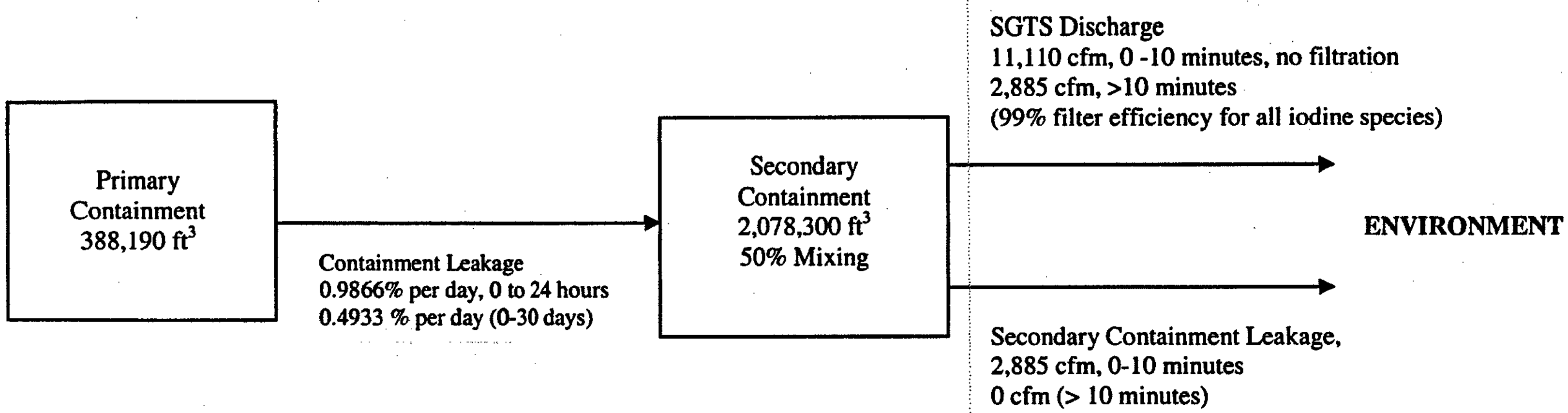
Notes:

1. The total Control Room habitability envelope free volume of 518,000 ft^3 is modeled into the RADTRAD code. In order to take credit for the radiation shielding effects of the control structure floors, the EDE portion of the Control Room TEDE as calculated by RADTRAD is adjusted by the ratio of the geometry factor GF for 518,000 ft^3 to the GF for 110,000 ft^3 (free of volume of Control Room elevation 729'-1"). The geometry factor is defines as: $GF = 1173 / (V^{0.338})$.

2. Air intake flow can vary between 5229 cfm and 6391 cfm. The CRHE doses are based on the air intake and corresponding CRHE exhaust flow rates that result in the highest dose.

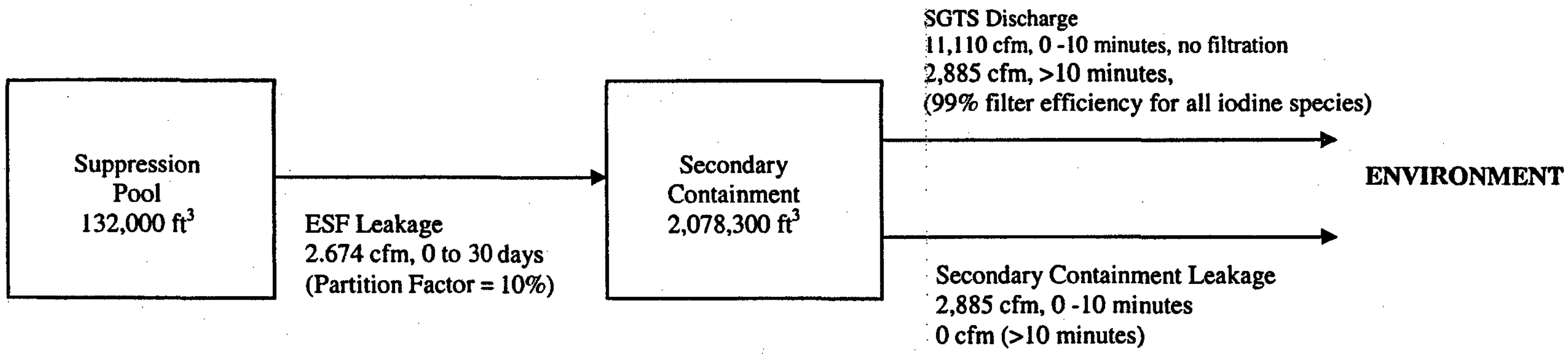
3. If the Control Room Emergency Outside Air System (CREOAS) is not activated for the accident, the air intake activity flow path is unfiltered.

Figure 5: A Simplified Radiological Release Model Developed to Calculate LOCA Doses from Primary Containment Leakage Path into Reactor Building Utilizing RADTRAD



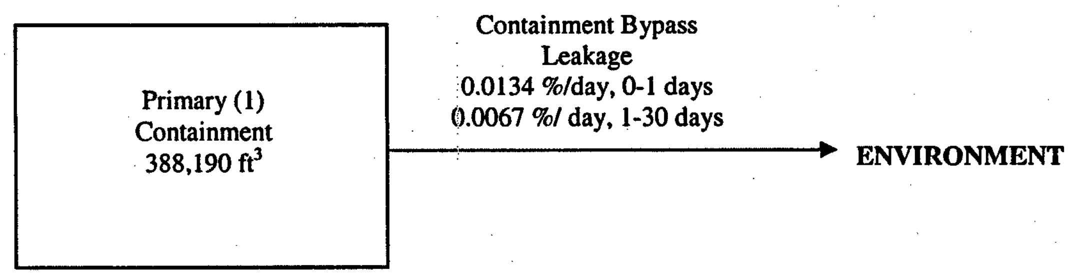
CRHE isolates for this event

Figure 6: A Simplified Radiological Release Model Developed to Calculate LOCA Doses from ESF Leakage Path Utilizing RADTRAD



CRHE isolates for this event

Figure 7: A Simplified Radiological Release Model Developed to Calculate LOCA Doses from Primary Containment Bypass Leakage Path Utilizing RADTRAD

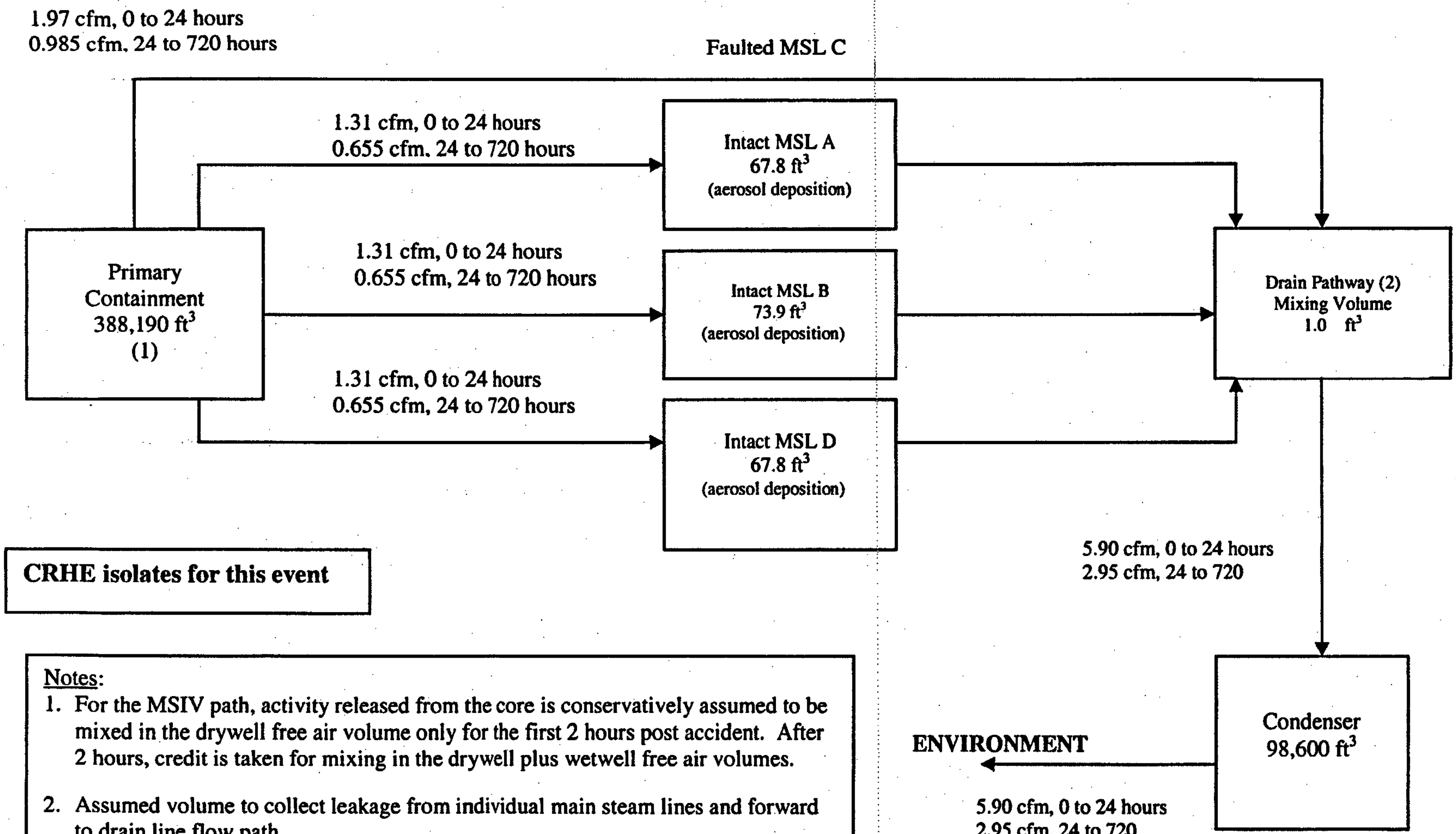


CRHE isolates for this event

Note:

(1) For the bypass leakage path, activity released from the core is conservatively assumed to be mixed in the drywell free air volume (239,600 ft^3) only for the first 2 hours post accident. After 2 hours, credit is taken for mixing in the drywell plus wetwell free air volumes.

Figure 8: A Simplified Radiological Release Model Developed to Calculate LOCA Doses from MSIV Leakage Path Utilizing RADTRAD



Notes:

1. For the MSIV path, activity released from the core is conservatively assumed to be mixed in the drywell free air volume only for the first 2 hours post accident. After 2 hours, credit is taken for mixing in the drywell plus wetwell free air volumes.

2. Assumed volume to collect leakage from individual main steam lines and forward to drain line flow path.

4.4 Main Steam Line Break

A detailed discussion of the design input parameters, assumptions, methodology, analysis, and results supporting the MSLB accident are provided in Attachment 10 as Calculation EC-RADN-1128, "Steam Line Break Accident CRHE and Offsite Doses – AST." Attachment 3 provides a matrix which compares the RG 1.183 regulatory position with the parameters and methodologies utilized to calculate the MSLB accident CRHE and offsite doses.

Table 4.4-1 provides a comparison of the design inputs utilized to determine the existing licensing basis MSLB and the proposed AST MSLB accident doses.

Table 4.4-1: Design Input Comparison – Current Licensing Basis vs. AST Design - MSLB

Parameter	CLB Parameter	AST Parameter
Core Thermal Power Level	3616 MWt	4032 MWt
Reactor Coolant Iodine Concentrations	0.2 μCi/gm DE I-131 ICRP 30 4 μCi/gm DE I-131 ICRP 30	0.2 μCi/gm DE I-131 ICRP 30 4 μCi/gm DE I-131 ICRP 30
Radioiodine Chemical Species	NA	95% Aerosol 4.85% Elemental 0.15% Organic
Pre-accident Iodine Spike	4 μCi/gm DE I-131 ICRP 30	4.0 μCi/gm DE I-131 TEDE
Maximum Equilibrium Iodine Spike	0.2 μCi/gm DE I-131 ICRP 30	0.2 μCi/gm DE I-131 TEDE
Iodine Carryover Fraction	2%	2% for reactor coolant 8% for steam
Design Offgas Release Rate (Noble Gases)	362,000 μCi/sec at 30 minutes decay	403,000 μCi/sec after 30 minutes decay
Failed Fuel Resulting from MSLB	0	0
MSIV Isolation Time	5.0 seconds	5.0 seconds
Total Release Duration	5.5 seconds	5.5 seconds
Holdup in TB	None	None
Liquid Release	84,840 lbm	101,808 lbm
Steam Release from the Steam Dome	6,650 lbm	7,980 lbm
Steam Release from Flashed Liquid	6,468 lbm	7,776 lbm
Dose Conversion Factors	ICRP 30	RADTRAD Table 1.4.3.3-2
Control Structure Habitability Envelope Total Volume	Note 1	518,000 ft^3
CR Isolation Time	Note 1	None
Intake Air Flow, Total into Control Structure	Note 1	6391 cfm
Emergency Filtered Recirculation Air Flow	Note 1	0 cfm
Unfiltered Air Inleakage Ingress/Ingress	Note 1	10 cfm
Other Unfiltered Air Inleakage	Note 1	500 cfm

Table 4.4-1: Design Input Comparison – Current Licensing Basis vs. AST Design - MSLB		
Parameter	CLB Parameter	AST Parameter
Emergency Filter Bed Depth/ Removal Efficiency	Note 1	Filters not credited during puff passage or in clean up after passage.
Operator Breathing Rates	Note 1	3.5E-04 m^3/sec
Operator Occupancy Factors	Note 1	1.0, 0-24 hrs 0.6, 1-4 days 0.4, 4-30 days
Dose Calculation Program	EXCEL	RADTRAD (Note 2)

Notes: 1. CLB for the MSLB did not include a dose evaluation for CRHE. The DBA LOCA dose was assumed bounding.

2. Releases can occur from the steam tunnel blowout panels in the turbine and Reactor Building. The RB Unit 2 blowout panel Control Room χ/Q values (worst case) are used in RADTRAD.

4.4.1 Introduction and Background

The postulated MSLB accident assumes a double-ended break of one main steam line outside the primary containment. The assumed displacement of the pipe ends permits a maximum blowdown rate. The mass of coolant released is the amount in the steam line and connecting lines at the time of the break plus the amount passing through the MSIVs prior to closure (5.5 seconds). For the design basis accident, the reactor is assumed to be in hot standby prior to the break. This condition maximizes the liquid mass release and hence activity releases. The mass releases for the design basis cases are:

Liquid release	84,840 lbm
Steam release from flashed liquid	6,480 lbm
Steam from steam dome	6,650 lbm

The above mass releases were increased by 20% to provide additional margin. Evaluations of steam line break masses for other extended power uprate plants determined that the increases in mass releases were small compared to the pre-uprate main steam line break masses while at power.

The release of steam to the environment resulting from the MSLB is assumed to be an instantaneous ground level puff release from the RB Unit 2 blowout panel. The methodology used to establish the puff transit time and the normalized concentration as a function of distance traveled is consistent with RG 1.194.

The key parameters used in the MSLB analyses are discussed in Attachments 3 and 10.

The NRC approved computer code RADTRAD, endorsed by RG 1.183, is used to calculate the dose to the Control Room operator as well as the doses at the EAB and LPZ.

4.4.2 Source Term

The fission product inventory available for release is based on (1) the maximum equilibrium reactor coolant DE I-131 concentration of 0.2 uCi/gm and (2) the pre-accident iodine spike DE I-131 concentration of 4.0 uCi/gm specified in TS 3.4.7. To account for iodine spiking, the equilibrium level of DE I-131 was increased by a factor of 20 to achieve a spiking concentration of 4 uCi/gm. No fuel damage is predicted to occur for the MSLB.

The FSAR design basis noble gas source term is 100,000 μCi/s after 30 minutes decay. Short lived isotopes (half lives less than one minute) were excluded due to radiological decay. The concentrations shown have been determined to be conservative for extended power uprate with and without hydrogen water chemistry (Reference 32).

A total off gas release rate of 403,000 μCi/sec after 30 minutes delay was used for the noble gas source with a core power of 4032 MWt based on TS 3.7.5. The individual noble gas concentrations were scaled from the 100,000 μCi/sec by applying a factor of 4.03.

The scaling of the 100,000 μCi/sec is similar to the approach in Regulatory Guide 1.98 (Reference 33) for analyzing postulated failures in off gas systems. The same approach is used for setting SSES's main condenser offgas rate in Technical Specification 3.5.7 (Reference 34).

The methodology and assumptions used to calculate the total number of curies in the source term is consistent with RG 1.183 and the current licensing basis. The activity (in the terms of DE I-131 and noble gases) in the mass of the initial liquid blowdown was assumed to be released to the atmosphere instantaneously, as a ground level release, and no credit was taken for plateout, holdup, or dilution within facility buildings.

4.4.3 Mitigation

The only mitigative action credited for the MSLB event was the termination of the release upon the automatic closure of the MSIVs. The MSIV isolation actuates on a high flow signal. Per RG 1.183, Appendix D, Section 4.1, "The main steam line isolation valves (MSIV) should be assumed to close in the maximum time allowed by Technical Specification." The MSIV closure time is 5.5 seconds. The 5.5 second closure time is consistent with the current licensing basis and is supported by the Technical Specification 3.6.1.3. This surveillance requires the performance of periodic stroke time tests with an acceptance criterion of greater than or equal to 3 seconds and less than or equal to 5 seconds. An additional 0.5 seconds was conservatively added to account for the high flow signal to actuate MSIV isolation.

No credit is taken for filtration of radioiodine from the steam line break. The Control Room emergency outside air supply system (CREOASS) does not automatically actuate for this accident. Manual operation is not credited because the accident occurs rapidly with the release moving away from the Control Room within minutes.

4.4.4 Radiological Transport Modeling (see Figures 4 & 9)

The release of steam resulting from the MSLB (through a blowout panel in the Reactor Building steam tunnel) was assumed to be an instantaneous ground level puff.

The Control Room outside air intake is normally open and CREOASS filters bypassed. Per SSES Units 1 & 2 Technical Specifications 3.7.3.4 and 5.5.7a, the outside air intake flow ranges from 5229 cfm to 6391 cfm with positive pressure guaranteed at ≤ 5810 cfm. Dose analyses were performed at 5229 and 6391 cfm with the 6391 cfm flow rate resulting in the worst case doses. Additional unfiltered inleakage of 500 cfm is assumed in the analysis. This amount bounds the leakages and error bands from the December 2004 CRHE tracer gas tests. Unfiltered ingress/egress leakage through doors of 10 cfm is assumed based on NUREG-0800, Chapter 6.4.

The RG 1.194 methodology was used to establish the puff transit time, normalized concentration as a function of distance traveled in the downwind or "x" direction, and the time-integrated normalized centerline concentration. Equation 10 of RG 1.194 was used to calculate the Control Room χ/Q. The puff from the steam release (including the flashed steam) was assumed to be released at ground level with an initial volume corresponding to standard atmospheric conditions. No buoyancy was considered. All the activity in the liquid was assumed to be released into the puff. The time required for the plume to transit to the new local Control Room air intake was based on the plume moving with a horizontal velocity of 1 m/s per RG 1.194. The puff centerline is assumed to pass directly over the local Control Room air intake.

4.4.5 Results – Control Room Operator Dose

The RADTRAD computer code was used to determine the Control Room operator dose and is shown in Table 4.4-2.

Table 4.4-2
MSLB Control Room Operator Doses (rem)

Source Term	Calculation TEDE	RG 1.183 TEDE
Dose with maximum equilibrium iodine	0.04	5.0
Dose with pre-accident iodine spiking	0.77	5.0

4.4.6 Results – Offsite Doses

The offsite dose calculation assumes a direct unfiltered release to the environment; but because of the greater distances to the EAB and LPZ boundary, the dispersed release is assumed to be a continuous plume, modeled with the equations provided in RG 1.145. Plume dilution due to buoyancy is not credited.

The RADTRAD computer code was used to determine the offsite doses, which are provided in Table 4.4-3.

Table 4.4-3
MSLB Offsite Doses (rem)

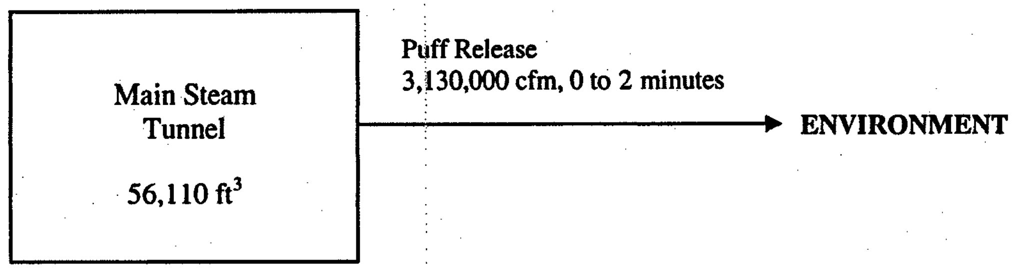
	Calculation TEDE	RG 1.183 TEDE
Doses with maximum equilibrium iodine		
EAB Dose	0.010	2.5
LPZ Dose	0.006	2.5
Dose with pre-accident iodine spiking		
EAB Dose	2.0	25
LPZ Dose	0.12	25

4.4.7 Conclusions

The MSLB Control Room operator dose for the maximum equilibrium case is a small fraction of the 5 rem TEDE regulatory limit. The dose for the pre-accident iodine spike case is also well below the 5 rem TEDE regulatory limit.

The MSLB offsite doses for the maximum equilibrium case are a small fraction of the 2.5 rem TEDE regulatory limit. The dose for the pre-accident iodine spike case is a small fraction of the 25 rem TEDE regulatory limit.

Figure 9: A Simplified Radiological Release Model Developed to Calculate MSLB Doses Utilizing RADTRAD



No CRHE isolation for this event

Notes:

1. 3,130,000 cfm is an arbitrary flow rate used in the RADTRAD model to ensure that at least 99% of the activity in MSLB blowdown mass is released by the time the MSIVs are fully closed at 5.5 seconds and virtually all activity is released in two minutes.

2. The puff release passes beyond the CRHE air intake in approximately 3 minutes. The CRHE flows are modeled to 30 days in order to calculate the dose from the accident and the residual in the CRHE after the accident.

4.4.8 Summary of Calculation Conservatisms

- The iodine water concentrations are maximized by the use of a 2% steam carryover (corresponding to normal water chemistry operation) while the steam concentrations are maximized using an 8% steam carryover (corresponding to operation with hydrogen water chemistry). This approach bounds normal and hydrogen water chemistry operation.

- The mass releases are increased by 20% to provide additional margin. Evaluations of steam line break masses for other extended power uprate plants determined that the increases in mass releases were small compared to the pre-uprate main steam line break masses while at power.

- The CRHE's nominal inlet air flow rate is 5810 cfm. This analysis uses 6391 cfm, the maximum value for which the system is in compliance with Technical Specification 5.5.7a.

- Unfiltered inleakage is 500 cfm (not including 10 cfm from ingress/egress). This amount bounds the leakage and error band from the December 2004 CRHE tests.

- Additional noble gases not included in the standard list of the RADTRAD 60 isotopes were included in the analysis.

4.5 Control Rod Drop Accident

A detailed discussion of the design input parameters, assumptions, methodology, analysis, and results supporting the CRDA are provided in Attachment 10 as Calculation EC-RADN-1127, "Control Rod Drop Accident CRHE and Offsite Doses – AST." Attachment 3 provides a matrix which compares the RG 1.183 regulatory position with the parameters and methodologies utilized to calculate the CRDA onsite and offsite TEDEs.

Table 4.5-1 provides a comparison of the design inputs utilized to determine the existing licensing basis CRDA and the proposed AST CRDA doses.

Table 4.5-1: Design Input Comparison – Current Licensing Basis vs. AST Design - CRDA

Parameter -	CLB Parameter	AST Parameter
Core Thermal Power Level	3616 MWt	4032 MWt
Core Radial Peaking Factor	1.5	1.6
Number of Fuel Bundles in Core	764	764
Number of Fuel Rods per Bundle Atrium 10	87.8	87.8
Number of Fuel Rods Damaged for CRDA	1000	2000
CRDA Damaged Rods – Gap Activity	10% noble gases and iodines	10% noble gases and iodines 12% Cs and Rb
Failed (Melted) Fuel Fraction for CRDA	0	0.77%
CRDA Release Timing (Gap + Melted Fuel)	0	0
Pre-incident Iodine Spike	NA	NA
Condenser Free Air Volume	195,000 ft^3	195,000 ft^3
Fraction of Fuel Activity Released to Reactor Coolant from Melted Regions	NA	100% noble gases 50% iodines
Fraction of Activity Release in Reactor Coolant Reaching Condenser	100% noble gases 10% iodines	100% noble gases 10% iodines 1% others
Fraction of Activity Reaching Condenser Available for Release to Environment	100% noble gases 10% iodines	100% noble gases 10% iodines 1% others
Release Rate from Condenser to Turbine Building	1% per day for a duration of 24 hrs	1% per day for a duration of 24 hrs
Removal Rate from Condenser to Environment with MVP Running	NA	1212%/day
Number of Rods Damaged by CRDA Needed to Cause MVP Trip and Isolation	NA	30
Turbine Building Holdup	None	None

Table 4.5-1: Design Input Comparison – Current Licensing Basis vs. AST Design - CRDA		
Parameter -	CLB Parameter	AST Parameter
Radioiodine Chemical Species Released from Condenser	NA	97% Elemental 3% Organic
Dose Conversion Factors	ICRP 30	RADTRAD Table 1.4.3.3-2
Control Structure Habitability Envelope Total Volume	Note 1	518,000 ft^3
CRHE Isolation Time	Note 1	None assumed.
Intake Air Flow, Total into Control Structure	Note 1	6391 cfm
Unfiltered Air Inleakage Ingress/Egress	Note 1	10 cfm
Other Unfiltered Air Inleakage	Note 1	500 cfm
Emergency Filter Bed Depth	Note 1	Filters not credited.
Emergency Filter Bed Removal Efficiency	Note 1	Filters not credited.
Operator Breathing Rates	Note 1	3.5E-04 m^3/sec
Operator Occupancy Factors	Note 1	1.0, 0-24 hrs 0.6, 1-4 days 0.4, 4-30 days
Dose Calculation Program	TACT 5	RADTRAD (Note 2)

Notes: 1. CLB for the CRDA did not include a dose evaluation for CRHE. The DBA LOCA dose was assumed bounding.

2. The condenser leakage is released to the environment via the Turbine Building exhaust vent. The TB Unit 2 exhaust vent Control Room χ/Q is used in RADTRAD for the CRDA and is the worst case value.

4.5.1 Introduction and Background

The postulated CRDA involves the rapid removal of a highest worth control rod resulting in a reactivity excursion. Core performance analyses show the energy deposition that results from this event is below the threshold postulated to melt the fuel pellets (Reference 21). Even though no fuel melting is postulated for the SSES CRDA, 0.77% of the fuel within a failed fuel rod is conservatively assumed to melt. This fuel melt assumption is intended to ensure compatibility with the same assumption made in GE's Topical Report NEDO-31400A, which evaluated the elimination of certain main steam radiation monitor (MSLRMs) safety functions.

Currently, 1000 rods are assumed to fail in the CRDA design basis accident for the pre-uprate core. Compliance check calculations are performed for each unit and cycle to verify that the number of rods is less than 1000. For these compliance checks, rods with 170 cal/gm or more energy deposited in the fuel are assumed to fail. It is expected that the number of rods will not substantially change based on favorable results from compliance checking analyses and on scoping studies performed for extended power uprate.

In order to establish a conservative bound for assessing accident doses, it is assumed that 2000 fuel rods fail. A core average radial peaking factor of 1.6 is assumed in the analysis.

The CRDA is terminated at 24 hours per RG 1.183, Appendix C, Section 3.4. The activity released from the damaged fuel that reaches the turbine and condenser is released from the Turbine Building at ground level at a rate of 1% condenser volume per day for a period of 24 hours. No credit is taken for Turbine Building holdup or dilution. The analysis assumptions for the transport, reduction, and release of the radioactive material from the fuel and the reactor coolant are consistent with RG 1.183.

A second postulated accident was evaluated at low power operation with the mechanical vacuum pump (MVP) running. At low power with fewer than 30 rods failing, main steam line dose rates may be too low to be reliably sensed by the MSLRMs to generate a trip signal for the MVP. Failure to trip the MVP could result in an unfiltered release of fission products to the environment from the Turbine Building vent stack, which results in larger CRHE, EAB, and LPZ doses.

The key parameters used in the CRDA analysis are discussed in Attachments 3 and 10.

The NRC approved computer code RADTRAD, endorsed by RG 1.183, is used to calculate the dose to the Control Room operator as well as the doses at the EAB and LPZ.

4.5.2 Source Term

The source term used for the CRDA analysis was composed of releases from melted fuel and the gap activity from the fuel pins postulated to be damaged. This initial amount of activity was released into the reactor coolant at time zero. Activity in the reactor coolant available for release to the environment was calculated by applying transport fractions.

The fraction of the core inventory available for release to the environment was calculated as follows:

- [Percent of failed rods in core x gap release fraction + percent of failed rods in core x percent of failed rods melted x melted fuel release fraction] x fraction that reaches the condenser x fraction that is available for release to environment.

The iodine species released to the reactor coolant are assumed to be 95% aerosol, 4.85% elemental, and 0.15% organic. The iodine species released from the condenser to the environment are 97% elemental iodine and 3% organic iodine. To properly model this release speciation in RADTRAD, the proportions of 97% elemental and 3% organic were used at the source.

4.5.3 Mitigation

The CRDA is terminated at 24 hours per RG 1.183, Appendix C, Section 3.4. The activity released from the fuel from either the gap or from fuel pellets was assumed to be instantaneously mixed in the reactor coolant within the pressure vessel. No partitioning of the initial activity in the pressure vessel or by steam separator removal was credited. No credit was assumed for holdup or dilution within the Turbine Building; however, radioactive decay during the holdup in the turbine and condenser was credited. For the scenario that evaluates a release through the TB vent stack, no filtration is assumed. No other mitigation of the radiological release was credited.

No credit is given for isolation or filtration of the Control Room habitability envelope's (CRHE's) air supply. The Control Room emergency outside air supply system (CREOASS) does not automatically start-up for this accident. Manual actuation by the Control Room operator is possible, but is not credited. Control Room ventilation normal intake flow was unfiltered.

4.5.4 Radiological Transport Modeling (see Figures 4 & 10)

The radiological release model for the CRDA was developed consistent with RG 1.183. A ground level release was modeled from the Turbine Building's normal ventilation system at a rate of 1% condenser volume per day over a period of 24 hours. For the scenario with the MVP running, the removal from the condenser is at a rate of 1212% per day.

The CRHE's nominal inlet air flow rate is 5810 cfm unfiltered. This analysis uses 6391 cfm, the maximum value for which the system is in compliance with Technical Specification 5.5.7a. The use of the maximum flow rate results in the greatest uptake of activity during the accident. Additional unfiltered inleakage of 500 cfm is assumed in the analysis. This amount bounds the leakage and error band from the December 2004 CRHE tracer gas tests. Unfiltered ingress/egress leakage through doors of 10 cfm is assumed based on NUREG-0800, SRP Section 6.4.

4.5.5 Results – Control Room Operator Dose

The RADTRAD computer code was used to determine the Control Room operator dose. Table 4.5-2 shows the proposed licensing basis dose limit compared to the regulatory limit.

Table 4.5-2
CRDA Control Room Operator Dose (rem)

	Calculation TEDE	RG 1.183 TEDE
2000 Rods – MVP Not Operational		
CR Operator Dose	0.49	5.0
30 Rods – MVP Operational		
CR Operator Dose	1.80	5.0

4.5.6 Results – Offsite Doses

The RADTRAD computer code was used to determine the offsite dose. Table 4.5-3 shows the proposed licensing basis dose limit compared to the regulatory limit.

Table 4.5-3
CRDA Offsite Doses (rem)

	Calculation TEDE	RG 1.183 TEDE
2000 Rods – MVP Not Operational		
EAB Dose	0.19	6.3
LPZ Dose	0.05	6.3
30 Rods – MVP Operational		
EAB Dose	2.30	6.3
LPZ Dose	0.18	6.3

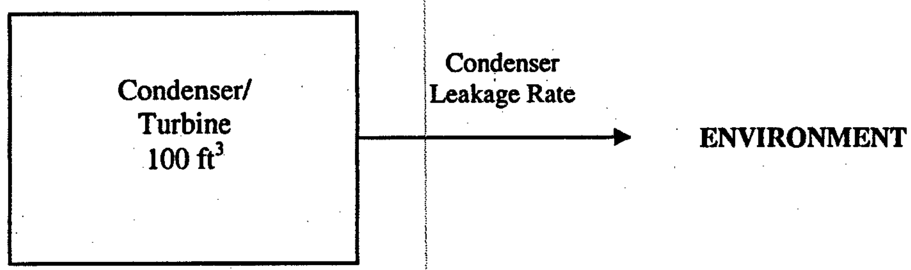
4.5.7 Conclusions

The CRDA Control Room operator dose is well below the 5 rem TEDE regulatory limit and each offsite dose is a small fraction of the 6.3 rem TEDE regulatory limit.

4.5.8 Summary of Calculation Conservatisms

- Assume 2000 rods experience clad damage instead of 1000 rods per current reload calculations.
- A failed fuel fraction of 0.77% was assumed, while no fuel melt is anticipated at SSES.
- Inclusion of the solids released from the melted fuel.
- The CRHE's nominal inlet air flow rate is 5810 cfm unfiltered. This analysis uses 6391 cfm, the maximum value for which the system is in compliance with Technical Specification 5.5.7a.
- Unfiltered inleakage is 500 cfm (not including 10 cfm from ingress/egress). This amount bounds the leakage and error band from the December 2004 CRHE tests.
- Additional noble gases not included in the standard list of the RADTRAD 60 isotopes were included in the analysis.
- No partitioning of the initial activity in the pressure vessel or by the steam separators was credited.

Figure 10: A Simplified Radiological Release Model Developed to Calculate CRDA Doses Utilizing RADTRAD



No CRHE isolation for this event

Condenser Leakage Rate = Leakage of 1 % per day, 0 to 24 hours, Full Power Operation
= Leakage of 1212 % per day, 0 to 24 hours, Mechanical Vacuum Pump Operating

Notes:

1. The condenser volume used in the RADTRAD model is arbitrary. The release is modeled as a rate process, e.g., %/day. For a rate process, the only restriction is that the condenser volume be non-zero.

2. The accident duration is one day.

4.6 Fuel Handling Accident/Equipment Handling Accident

A detailed discussion of the design input parameters, assumptions, methodology, analysis, and results supporting the FHA are provided in Attachment 10 as Calculation EC-RADN-1126, "CRHE and Offsite FHA/EHA Doses – AST." Attachment 3 provides a matrix which compares the RG 1.183 regulatory position with the parameters and methodologies utilized to calculate the FHA onsite and offsite TEDEs.

Table 4.6-1 provides a comparison of the design inputs utilized to determine the existing licensing basis FHA/EHA and the proposed AST FHA/EHA doses.

Table 4.6-1: Design Input Comparison – Current Licensing Basis vs. AST Design – FHA/EHA

Parameter	CLB Parameter	AST Parameter
Core Thermal Power Level	3616 MWt	4032 MWt
Earliest Fuel Handling Time	24 Hours	24 Hours
Number of Fuel Rods in Fuel Assembly/Number of Assemblies in Core	87.8/764	87.8/764
EHA/FHA – Number of Damaged Rods	Case 1 (FHA): 156/254.8 Case 2 (EHA): 366/460.8	Case 1 (FHA): 156/254.8 Case 2 (EHA): 366/460.8
Core Radial Peaking Factor	1.6	1.6
Damaged Rods – Gap Activity	10% - I 30% - Kr-85 10% - Other nobles gases	8% - I-131 10% - Kr-85 5% - Other nobles gases & halogens 12% - Alkali metals
Core Burnup (MWD/MTU)	60,000	39,000
Maximum Fuel Rod Pressurization	<1200 psig	<1200 psig
EHA Release Timing (Gap)	Instantaneously released & mixed into pool water	Instantaneously released & mixed into pool water
Minimum Pool Water Depth	21 feet	21 feet
Iodine Species Released to Pool	99.75%Elemental 0.25% Organic	99.85%Elemental 0.15% Organic
Pool DF	Noble gases - 1.0 Iodine - 100	Noble gases - 1.0 Aerosols - infinite Iodine - 138 (corrected for 21 feet)
EHA Release Duration	2 hours	2 hours
Activity Transport from Pool to Environment	SGTS actuation prior to activity release to environment	SGTS actuation prior to activity release to environment
SGTS Filter Bed Depth	8 in. Charcoal	8 in. Charcoal
SGTS Filter Bed Efficiency	99% for all iodine species	99% for all iodine species
Offsite Breathing Rates(m^3/sec)	3.5E-04, 0-8 hrs 1.8E-04, 8-24 hrs 2.3E-04, 1-30 d	3.5E-04, 0-8 hrs 1.8E-04, 8-24 hrs 2.3E-04, 1-30 d
Dose Conversion Factors	ICRP 30	RADTRAD Table 1.4.3.3-2

Table 4.6-1: Design Input Comparison – Current Licensing Basis vs. AST Design – FHA/EHA		
Parameter	CLB Parameter	AST Parameter
Control Structure Habitability Envelope Total Volume	Note 1	518,000 ft^3
CRHE Isolation Time	Note 1	Time = 0
Emergency Intake Air Flow, Total into Control Structure	Note 1	5229 cfm to 6391 cfm
Unfiltered Air Inleakage Ingress/Egress	Note 1	10 cfm
Other Unfiltered Air Inleakage	Note 1	500 cfm
Emergency Filter Bed Depth	Note 1	4 inches charcoal
Emergency Filter Bed Removal Efficiency	Note 1	99%
Operator Breathing Rates	Note 1	3.5E-04 m^3/sec
Operator Occupancy Factors	Note 1	1.0, 0-24 hrs 0.6, 1-4 days 0.4, 4-30 days
Dose Calculation Program	EXCEL	RADTRAD (Note 2)

Notes: 1. CLB for the FHA/EHA did not include a dose evaluation for CRHE. The DBA LOCA dose was assumed bounding.

2. The SGTS exhaust vent Control Room χ/Q is used in RADTRAD for this event.

4.6.1 Introduction and Background

Two postulated events are evaluated. Case 1 (fuel handing accident – FHA) considers a dropped fuel assembly unit (fuel assembly, channel, grapple and mast) weighing 1500 lbs. falling a distance of 32.95 feet onto the core. The number of failed rods is given as 156 rods for the Atrium 10 fuel assemblies. To conservatively address the issue of lead fuel assemblies (whether in the reactor or in the spent fuel pool) radiological dose results are included which assume that another Atrium 10 assembly representing a lead use assembly (LUA) completely fails resulting in a total of 254.8 failed rods. Case 2 (equipment handling accident – EHA) assumes an object weighing 1100 lbs. is dropped 150 feet onto the core. The dropped assembly and the core are assumed to be ATRIUM-10 fuel assemblies. The number of failed rods is given as 366 rods considering the Atrium 10 fuel assemblies and 460.8 rods considering the case for Atrium 10 + 1 LUA. The extent of damage was calculated based on the free fall distance and the resulting kinetic energy of the dropped assembly.

The key parameters used in the FHA analysis are discussed in Attachments 3 and 10.

The NRC approved computer code RADTRAD, endorsed by RG 1.183, is used to calculate the dose to the Control Room operator as well as the doses at the EAB and LPZ.

4.6.2 Source Term

The fission product inventory that constitutes the source term for this event is the gap activity in the fuel pins assumed to be damaged as a result of the postulated design basis FHA and EHA. A radial peaking factor of 1.6 and a 24-hour decay time after plant shutdown is assumed in the analysis.

Of this activity, all of the noble gases and only a fraction of the iodine were available for release (i.e., for the purpose of calculating radiological dose consequences) based on the scrubbing effect (i.e., DF) of the water above the dropped fuel assembly.

Consistent with RG 1.183, an overall DF of 138 was calculated utilizing Technical Paper, "Evaluation of Fission Product Release and Transport", G. Burley, 1971 (Reference 22) for the released iodines, based on a minimum fuel pool water depth of 21' (TS B3.7.7). An infinite DF was credited for the remaining particulate forms of the radionuclides contained in the gap activity. Consequently, the remaining particulates are not considered in the analysis. No DF credit was taken for the noble gas constituents of the gap activity.

The activity in the fuel rod gap available for release from the damaged rods is defined in RG 1.183, Section 3, Table 3 as:

8%	I-131
10%	Kr-85
5%	Other Nobles Gases & Halogens
12%	Alkali Metals

These release fractions have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU. The SSES extended burnup design - peak bundle exposure is 54,000 MWD/MTU and the maximum fuel rod pressurization is < 1200 psig.

4.6.3 Mitigation

The analysis assumes a ground level release over a 2-hour period through the SGTS (filtered). Under accident conditions, habitability for the CRHE is provided by the Control Room Emergency Outside Air Supply System (CREOASS). This system provides habitability zone isolation and a positive pressure for the CRHE. For this event, the CRHE automatically isolates and enters the emergency mode in sequence with the SGTS prior to commencement of the release of activity to environment.

Per SSES Units 1 & 2 Technical Specifications 3.7.3.4 and 5.5.7a, the Control Room Emergency Outside Air System (CREOASS) filtered intake flow ranges from 5229 cfm to 6391 cfm with positive pressure guaranteed at ≤ 5810 cfm. RADTRAD runs were made at 5229 cfm and 6391 cfm and it was determined that the 6391 cfm was limiting.

An unidentified unfiltered inleakage rate of 500 cfm is conservatively assumed in the RADTRAD model. This inleakage bounds the tracer gas test results (including error band) from the December 2004 SSES CRHE inleakage test.

In addition, 10 cfm of unfiltered inleakage is added to the CRHE to account for ingress/egress of personnel.

4.6.4 Radiological Transport Modeling (see Figures 4 & 11)

The radiological release modeled in this analysis is consistent with RG 1.183.

The activity was released from the Reactor Building via the SGTS via the environment over a 2 hour period.

The 0 – 2 hour CRHE χ/Q from the SGTS exhaust vent to the Control Room outside air intake of 1.45E-03 sec/m^3 was used in the analysis.

4.6.5 Results – Control Room Operator Dose

The RADTRAD computer code was used to determine the Control Room operator dose for the FHA/EHA events. Table 4.6-2 shows the proposed licensing basis dose limit for the FHA/EHA compared to the regulatory limit.

Table 4.6-2
FHA/EHA Control Room Operator Dose (rem)

	Calculation TEDE	RG 1.183 TEDE
CR Operator Dose (FHA)	0.10	5.0
CR Operator Dose (EHA)	0.13	5.0

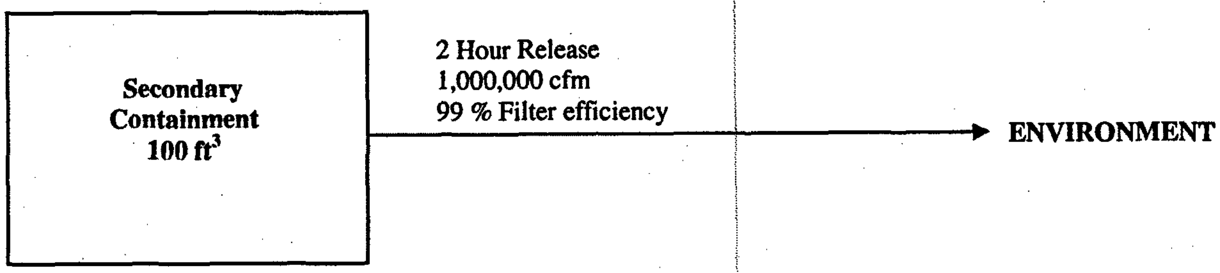
4.6.6 Results – Offsite Doses

The RADTRAD computer code was used to determine the offsite doses. Table 4.6-3 shows the proposed licensing basis dose limit for the FHA/EHA compared to the regulatory limit.

Table 4.6-3
FHA/EHA Offsite Doses (rem)

	Calculation TEDE	RG 1.183 TEDE
	FHA	
EAB Dose	1.38	6.3
LPZ Dose	0.08	6.3
	EHA	
EAB Dose	1.74	6.3
LPZ Dose	0.10	6.3

Figure 11: A Simplified Radiological Release Model Developed to Calculate FHA/EHA Doses Utilizing RADTRAD



CRHE isolates for this event

Note:

1. The Reactor Building volume and the 1,000,000 cfm flow rate used in the RADTRAD model are arbitrary values used to assure that 100% of the activity from the pool is released to the environment over the two hour period.

4.6.7 Conclusions

The FHA/EHA Control Room operator dose is below the 5 rem TEDE regulatory limit and each offsite dose is well below the 6.3 rem TEDE regulatory limit.

4.6.8 Summary of Calculation Conservatisms

- 500 cfm (not including 10 cfm from ingress/egress) of unidentified unfiltered inleakage is conservatively assumed in the RADTRAD model. This inleakage bounds the tracer gas test results (including error band) from the December 2004 SSES CRHE inleakage test.
- Additional noble gases not included in the standard list of the RADTRAD 60 isotopes were included in the analysis.
- Additional fuel failures associated with one lead use assembly.

4.7 <u>Miscellaneous Issues</u>

4.7.1 Use of Standby Liquid Control (SLC) System

This section provides the basis for crediting boron injection from the SLC system for suppression pool pH control. The maintenance of a suppression pool pH level above 7.0 is important to prevent re-evolution of iodine from the suppression pool water. The use of SLC is consistent with several other BWR submittals using AST methods.

The SLC system consists of two 100% capacity positive displacement triplex type (3-piston) injection pumps which when operating together are capable of delivering sodium pentaborate to the reactor vessel to meet 10 CFR 50.62 requirements, two 100% capacity explosive actuated injection valves, one storage tank, one test tank, and the piping, valves, instrumentation and controls necessary to inject the solution into the reactor and to test the SLC system. All equipment in the SLC system which comes in contact with the neutron absorbing solution is stainless steel for corrosion protection.

The SLC system is an independent and diverse backup system to the Control Rod Drive (CRD) System. The SLC system shuts down the reactor by injecting a neutron absorbing solution into the reactor coolant, which is circulated through the core. The neutron absorber used in the SLC system is an aqueous solution of sodium pentaborate decahydrate, $Na_2B_{10}O_{16}*10H_2O$. Sufficient solution is injected to bring the reactor from maximum rated power conditions to cold subcritical over the entire reactor temperature range, from maximum operating to cold shutdown conditions. The SLC system is not required to scram the reactor, or to serve as a backup scram system.

The SLC System is listed as a Nuclear Steam Supply System. Based on the PPL SSES Final Safety Analysis Report (FSAR) (Reference 24), Section 9.3.5, the SLC system is identified as a safe shutdown system having a safety-related classification. Safety-related systems provide the actions necessary to assure safe shutdown of the reactor, to protect the integrity of radioactive material barriers, and/or to prevent the release of radioactive material in excess of allowable dose limits. Safe shutdown of the reactor is classified as a nuclear safety function, and thus, the SLC system is classified as having a safety-related function.

However, the system fails to meet all the requirements of a safety-related system in that the SLC system is not designed for the single active component failure criteria. Therefore, a failure of a critical component would prevent the system from controlling suppression pool liquid pH by using the buffering action of boron injected to the suppression pool from the SLC system (via the reactor vessel). The success of the SLC system is necessary in order to take credit for this pH control function and prevention of iodine re-evolution.

In February 2004, the NRC issued review guidelines, Guidance on the Assessment of a BWR SLC System for pH Control" (Reference 25), for assessing the acceptability of a BWR SLC system for pH control. These guidelines have been the basis of several recent Requests for Additional Information (RAIs). PPL has evaluated the SLC system against these guidelines. The assessment is provided in Attachment 10, Calculation EC-053-1012, "Assessment of SLC System for Suppression Pool pH Control." Based on the response in the calculation, SSES satisfies the criteria for the acceptability of the SLC system for pH control.

No hardware changes are necessary to use SLC in this new functional mode. However, a change to the SLC TS is proposed in this LAR to require that two SLC subsystems shall be operable in Modes 1, 2, and 3 (see Attachment 5). This proposed TS change supports the SLC function as credited in the AST LOCA analysis.

4.7.2 Operator Actions

There are new manual operator actions proposed as part of this LAR that are not currently considered in the SSES design basis and must be directed by new station procedures that will be written and approved before SSES AST implementation. The operator action assumed in the proposed AST dose consequence analyses is the initiation of the SLC system for boron injection to maintain the suppression pool water pH above 7.0, precluding iodine re-evolution. TS Sections 3.1.7, "Standby Liquid Control (SLC) System" and 3.3.6.1, "Primary Containment Isolation Instrumentation" and their Bases were also revised to address this change in the SLC system requirements (see Attachments 5 through 7).

4.7.3 NUREG-0737

The guidance provided in Regulatory Positions 4.1 and 4.2 of RG 1.183 was utilized, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1 of RG 1.183 (Reference 2), concerning NUREG-0737 (Ref. 5).

The acceptance criteria for the various NUREG-0737 items generally reference General Design Criteria 19 (GDC 19) from Appendix A to 10 CFR Part 50 or specify criteria derived from GDC-19. These criteria are generally specified in terms of whole body dose, or its equivalent to any body organ.

An evaluation (EC-RADN-1134, "Impact of AST on Current NUREG-0737 Radiological Evaluations that use TID-14844 DBA-LOCA Releases", Revision 0 – see Reference 35 Attachment 2 and Attachment 10 for a copy of the calculation) was performed to identify potential impacts of applying AST methodologies in accordance with NUREG-0737. The evaluation considered the comparative radiation levels from an AST at current licensed power and the existing TID-14844 methodology source terms (such as airborne activity in the Reactor Building and also as activity in the suppression pool water). The evaluation demonstrates that the current calculated doses (based on TID-14844 source terms) bound the doses that would be calculated based on AST source terms. The following NUREG-0737 Sections were investigated for AST impact:

- The current radiological dose analyses for post-accident vital area access and post-accident sampling (NUREG-0737, Item II.B.2 and Item II.B.3).
- The current radiological dose analyses for the post-accident containment high range radiation monitors (NUREG-0737, Item II.F.1).
- The Control Room post-accident radiological dose analyses for emergency support facility upgrades and Control Room habitability (NUREG-0737, Items III.A.1.2 and III.D.3.4).
- The post-accident sources of radiation and radioactivity outside the primary containment in terms of impact on dose analysis related to integrity of systems outside containment likely to contain radioactive material (NUREG-0737, Item III.D.1.1).

<u>Item II.B.2 (Post Accident Access Shielding)/ Item II.B.3 (Post Accident Sampling Capability)</u>

Post Accident Vital Area Access and Sampling – Post-accident personnel missions resulting in mission doses (including post-accident sampling) have been previously identified. The implementation of the AST methodology does not result in any new operator missions. Plant calculations used in support of plant post-accident vital area access (prepared in accordance with NUREG-0737, Items II.B.2 and II.B.3) were evaluated for impact by AST. The evaluated mission doses would be less than 5 rem TEDE.

Item II.F.1 (Accident Monitoring Instrumentation)

Post-Accident Radiation Monitor – The containment high range radiation monitors used to monitor post-accident primary containment radiation levels were evaluated for the impact of AST. The monitors continue to provide their design function and envelope the projected radiation rates.

Item III.A.1.2 (Emergency Response Facilities)

On-Site Emergency Centers

The Control Room is the primary location for the initial assessment and coordination of corrective actions for all emergency conditions. The Control Room is equipped with the display and controls for all critical plant systems, radiological and meteorological monitoring systems, and all station communication systems.

Off-site emergency functions initially served by the Control Room are transferred to the Technical Support Center (TSC) or Emergency Operations Facility (EOF) for an Alert, a Site Area Emergency, or a General Emergency as deemed appropriate by the Emergency Director. The primary consideration is to ensure that the number of personnel involved with the emergency in the Control Room shall not impair the safe and orderly shutdown of the reactor or the operation of plant safety systems.

The Operations Support Center (OSC) is the primary on-site assembly area for operations support team personnel during an emergency. There are two OSC areas (1) a primary OSC located separate from the Control Room and TSC and a backup OSC located in the Control Structure. The primary OSC is located on the first floor of the South Administrative Building. The backup OSC occupies 340 square feet adjacent to the Control Room on elevation 729'-1" of the Control Structure.

The OSC is utilized initially as the central location for assembly and accountability of on-shift emergency team personnel. If or when the TSC is activated, all non-operations support team personnel assemble and are accounted for in the OSC (or TSC if the backup OSC is being utilized). TSC personnel assess the need for emergency team personnel and based on this assessment, the OSC dispatches available team personnel and calls in additional personnel as needed. The OSC and TSC assembly areas will be monitored continuously for habitability. If these areas become uninhabitable, retained personnel will be directed to alternate holding areas.

The primary OSC is monitored for radiation exposure using a local area radiation monitor (ARM) and a continuous air monitor (CAM). The radiation dose will be limited to less than 5 rem TEDE, at which time the primary OSC will be relocated to the backup OSC. Applicable criteria are specified in General Design Criterion 19, Standard Review Plan NUREG-0800, Section 6.4, 10 CFR 50.67, and NUREG-0737, Item II.B.2. The current licensing basis for the habitability of the primary OSC remains valid. Based on the overall reduction in Control Room operator dose due to AST methodology, the use of a local ARM and CAM, and the ability to evacuate the primary OSC, an updated quantitative assessment of the primary OSC dose based on the AST source term was not performed.

A backup OSC area exists in the Control Structure and would be used in the event the primary OSC becomes uninhabitable. Backup OSC personnel are protected from radiological hazards, including direct shine and airborne activities for postulated accident conditions to the same degree as Control Room personnel. Applicable criteria are specified in General Design Criterion 19, Standard Review Plan NUREG-0800, Section 6.4, 10 CFR 50.67, and NUREG-0737, Item II.B.2. Therefore, current licensing basis for the habitability of the backup OSC remains valid and an updated quantitative assessment of the backup OSC dose based on the AST source term was not performed. Based on the analysis performed for the DBA LOCA, limited access control shall be required along the east wall of the OCS due to shine from core spray piping located in the adjacent RB. Personnel exposure in Room C-402 shall be limited to ≤ 0.738 rem TEDE for the duration of the accident. This is accomplished by designating an area 5' from the CRHE east wall as a limited entry zone. Emergency Plan station procedures shall be revised to address OSC access control prior to AST implementation (see Attachment 8, Item 1).

The TSC is a controlled access area that provides working space and facilities for approximately 25 personnel. These personnel provide guidance to plant operations personnel for management of emergency conditions and accident mitigation. The TSC is located in the existing Control Room mezzanine above the Control Room at elevation 741'-1" of the Control Structure. The TSC is within approximately two minutes travel time of the Control Room by stairs.

TSC personnel are protected from radiological hazards, including direct shine and airborne activities for postulated accident conditions to the same degree as Control Room personnel. Applicable criteria are specified in General Design Criterion 19, Standard Review Plan NUREG-0800, Section 6.4, 10 CFR 50.67, and NUREG-0737, Item II.B.2.

Shielding for the TSC is the same as for the Control Room for total dose to occupants from direct shine and airborne. Exposure will not exceed 5 rem TEDE for the duration of the accident. This is in accordance with General Design Criterion 19, Standard Review Plan NUREG-0800, Section 6.4, 10 CFR 50.67, and NUREG-0737, Item II.B.2. Therefore, current licensing basis for the habitability of the backup OSC remains valid and an updated quantitative assessment of the backup OSC dose based on the AST source term was not performed. Based on the analysis performed for the DBA LOCA, limited access control shall be required along the east wall of the TSC due to shine from core spray piping located in the adjacent RB. Personnel exposure in NRC Conference Room (room C-414) and the Electrical Room (room C-413) shall be limited to ≤ 0.738 rem TEDE for the duration of the accident. This is accomplished by designating an area 5' from the CRHE east wall as a limited entry zone. Emergency Plan station procedures shall be revised to address TSC access control prior to AST implementation (see Attachment 8, Item 1).

Off-Site Emergency Center

The Emergency Operations Facility (EOF) provides continuous management of PPL activities during radiological emergencies that may have off-site impact. The EOF is located at a distance greater than 10 miles from the site. Exposure will not exceed 5 rem TEDE for the duration of the accident. This is in accordance with General Design Criterion 19, Standard Review Plan NUREG-0800, Section 6.4, 10 CFR 50.67, and NUREG-0737, Item II.B.2. Therefore, current licensing basis for the habitability of the EOF remains valid and an updated quantitative assessment of the EOF dose based on the AST source term was not performed.

<u>Item III.D.3.4 (Control Room Habitability Requirements)</u>

Control Room Radiation Protection – The resultant doses to the Control Room for each of the four DBAs analyzed for AST have been determined. In each case, the Control Room dose is less than 5 rem TEDE. Based on the analysis performed for the DBA LOCA, limited access control shall be required along the east wall of the Control Room due to shine from core spray piping located in the adjacent RB. Personnel exposure in the Office (Room C-401) and the Operational Support Center (Room C-402) shall be limited to ≤ 0.738 rem TEDE for the duration of the accident. This is accomplished by designating an area 5' from the CRHE east wall as a limited entry zone. Emergency Plan station procedures shall be revised to address OSC access control prior to AST implementation (see Attachment 8, Item 1).

<u>Item III.D.1.1 (Leakage Control)</u>

Radioactive Sources Outside the Primary Containment – The DBA LOCA Control Room dose analysis, as well as that for offsite doses, considers the effects of coolant leakage outside the primary containment and (for the Control Room and TSC dose analyses only) the shine contribution from the Reactor Building, ECCS piping, the external cloud, and other source term bearing systems and/or components.

4.7.4 Equipment Qualification

The source term associated with environmental qualification of equipment will remain consistent with previous commitments under 10 CFR 50.49. As stated in the cover letter to this submittal, the SSES application to implement the AST methodology is requested with one exception. That exception is TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," will continue to be used as the radiation dose basis for equipment qualification.

5.0 REGULATORY SAFETY ANALYSIS (10 CFR 50.92 Evaluation)

The 10 CFR 50.92 Evaluation is included as Attachment 9 of this submittal.

6.0 ENVIRONMENTAL CONSIDERATIONS (10 CFR 50.21 Evaluation)

The 10 CFR 50.21 Evaluation is included as Attachment 9 of this submittal.

7.0 REFERENCES

1. U.S. Atomic Energy Commission Technical Information Document TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 23, 1962.

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

3. NUREG-0800, Section SRP 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000.

4. General Electric Company (GE) Report, NEDC-31858P-A, "Boiling Water Reactor Owners Group (BWROG) Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," Revision 2.

5. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

6. USNRC, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Regulatory Guide 1.145, Revision 1, November 1982.

7. USNRC, "Standard Review Plan For the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 6.4, "Control Room Habitability System," NUREG-0800, USNRC, 1987.

8. J.V. Ramsdell and C.A. Simonen, "Atmospheric Relative Concentrations in Building Wakes," NUREG-6331, Revision 1, USNRC, May 1997 (ARCON96 computer code).

9. USNRC Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

10. USNRC Regulatory Guide 1.194, "Atmospheric Relative Concentrations For Control Room Radiological Habitability Assessments At Nuclear Power Plants," June 2003.

11. NUREG/CR-0200, "SAS2H: A Coupled one-Dimensional Depletion and Shielding Analysis Module," Revision 6, Volume 1, Section S2, March 2000.

12. NUREG/CR-0200, "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," Revision 6, Volume 2, Section F7, March 2000.

13. EMF-85-74(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Power Corporation, February 1988, including Supplement, 1(P)(A) and Supplement 2(P)(A).

14. PPL Calculation EC-PUPC-1001, "NEDC-32161P, General Electric Power Uprate Engineering Report for Susquehanna Steam Electric Station," Revision 6, 5/24/04.

15. NRC Standard Review Plan Section 15.6.5, "Radiological Consequences of a Design Basis Loss-of-Coolant Accident," Revision 2, July, 1981.

16. NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal and Dose Estimation," April 1998 and Supplement 1, 2, & 3 dated June 8, 1999.

17. AEB-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term," December 9, 1998.

18. ~~J. E. Cline, "MSIV Leakage Iodine Transport Analysis," Letter Report, 3/26/1991.~~

19. NEDC-31858P-A, "BWROG Report for Increasing Main Steam Isolation Valve Leakage Rate Limits and Elimination of Leakage Control Systems," August 1999.

20. Letter from Brenda Mozafari, Division of Licensing Project Management, Office of Nuclear Reactor Regulation to Mr. J. S. Keenan, Brunswick Steam Electric Plant, Carolina Power & Light Company, Southport, North Carolina, Brunswick Steam Electric Plant, Units 1 and 2 – Issuance of Amendment RE: Alternative Source Term (TAC Nos. MB2570 and MD2571) dated 3/30/02.

21. PLI-93490, EPU Failed Fuel Pin Impacts for AST Events FEHA and CRDA, April 19, 2005.

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

23. USNRC Regulatory Guide 1.52, "Design, Testing and Maintenance Criteria For Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration And Adsorption Units Of Light-Water-Cooled Nuclear Power Plants," Revision 2, March 1978.

24. PPL SSES FSAR, Section 9.3.5, "Standby Liquid Control System," Revision 61.

25. NRC issued Review Guidelines, "Guidance on the Assessment of a BWR SLC System for pH Control," dated February 12, 2004.

7.0 REFERENCES - Continued

26. SSES Amendment 252 to License NPF-14 and 217 to License NPF-22, "Application for Technical Specification Improvement to Eliminate Requirements for Post Accident Sampling Stations for Boiling Water Reactors Using the Consolidated Line Item Improvement Process," dated 3/3/2003.

27. NRC approved Industry/Technical Specification Task Force Standard Technical Specification Change Traveler, TSTF-413, "Elimination of Requirements for Post Accident Sampling System (PASS)."

28. SSES Amendment 151 to License NPF-14 and 121 to License NPF-22, "Susquehanna Steam Electric Station, Units 1 and 2 (TAC Nos. M91013 and M91014)," dated 8/15/1995.

29. USNRC Regulatory Guide 1.3, "Assumption Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," Revision 2, June 1974.

30. USNRC Regulatory Guide 1.5 (Safety Guide 5), "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors," Revision 0, 3/10/71.

31. USNRC Regulatory Guide 1.25 (Safety Guide 25), "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage facility for Boiling and Pressurized Water Reactors," Revision 0, 3/23/72.

32. PPL Calculation EC-RADN-1038, "Radioactive Material Source Term Evaluation for Normal Conditions with Hydrogen Water Chemistry," Revision 0.

33. USNRC Regulatory Guide 1.98, "Assumptions Used For Evaluating The Potential Radiological Consequences Of A Radioactive Offgas System Failure In A Boiling Water Reactor," March 1976.

34. SSES Technical Specification 3.7.5, "Main Condenser Off Gas," Amendments 151 and 178, Bases for Improved Specification B 37.5.

35. PPL Calculation EC-RADN-1134, "Impact of AST on Current NUREG-0737 Radiological Evaluations that use TID-14844 DBA-LOCA Releases," Revision 0.

36. TSTF-448, Revision 2, BWOG-111, R0, "Technical Specification Task Force – Improved Standard Technical Specifications Change Traveler."

7.0 REFERENCES - Continued

37. PAVAN, "An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," NUREG/CR-2858, November, 1982.

38. USNRC Regulatory Guide 1.49, "Power Levels of Nuclear Power Plants," December, 1973.

39. PLA-5916, SSES Final resolution to Generic letter 2003-01 Control Room Habitability, Docket Nos. 50-387 and 50-388, June 28, 2005.

Attachment 3 to PLA-5963

Regulatory Guide 1.183
Compliance Table

Regulatory Guide 1.183 Compliance

Regulatory Guide 1.183 Sections 3 through 7 and Appendices A, B, C and D provide methodologies and assumptions that are acceptable to the NRC staff related to design basis radiological analyses for Alternate Source Term. Compliance with Regulatory Guide 1.183 positions are discussed below:

Please note: the information provided in this table is based on the calculations provided in Attachment 10.

RG 1.183 Section	Regulatory Guide 1.183 Position	Basis of Compliance
3.	ACCIDENT SOURCE TERM This section provides an AST that is acceptable to the NRC staff. The data in Regulatory Positions 3.2 through 3.5 are fundamental to the definition of an AST. Once approved, the AST assumptions or parameters specified in these positions become part of the facility's design basis. Deviations from this guidance must be evaluated against Regulatory Position 2. After the NRC staff has approved an implementation of an AST, subsequent changes to the AST will require NRC staff review under 10 CFR 50.67.	Conforms Accident source terms were developed based on the guidance of Section 3 of RG 1.183.
3.1	Fission Product Inventory The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty. Note: the uncertainty factor used in determining the core inventory should be that value provided in Appendix K to 10 CFR Part 50, typically 1.02. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. Note that some radionuclides, such as Cs-137, equilibrium will not be reached prior to fuel offload. Thus, the maximum inventory at the end of life should be used. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 (Ref. 17) or ORIGEN-ARP (Ref. 18). Core inventory factors (Ci/MWt) provided in TID14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels.	Conforms The inventory of fission products in the reactor core and available for release to the containment was based on the maximum full power operation of the core with proposed extended power values for fuel enrichment, fuel burnup, and an assumed core power equal to the proposed extended rated thermal power (core thermal power of 4032 MWt [102% (ECCS evaluation uncertainty) of 3952 MWt], which corresponds to an average fuel assembly power of 5.28 MWt. The fission product source terms calculated are slightly conservative (~2%) because the uranium mass assumed is higher than the actual ATRIUM-10 assembly mass.

For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors from the facility's core operating limits report (COLR) or Technical Specifications should be applied in determining the inventory of the damaged rods.

No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.

The maximum inventory at the end of life was used. The core inventory was determined utilizing a version of the ORIGEN code (SAS2H/ORIGIN-S) to generate the source terms for the ATRIUM-10 fuel. The SAS2H/ORIGIN-S code comprises an advances version of ORIGIN, and is consistent with the source term code recommendations given by the USNRC for generation of alternate source terms in RG 1.183.

3.2 Release Fractions

Note: the release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup of 62,000 MWD/MTU. The data in this section may not be applicable to cores containing mixed oxide (MOX) fuel.

The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs are listed in Table 1 for BWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1.

For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor.

Table 1
BWR Core Inventory Fraction Released Into Containment

Group	Gap Release Phase	Early In-vessel Phase	Total
Noble Gases	0.05	0.95	1.0
Halogens	0.05	0.25	0.3
Alkali Metals	0.05	0.20	0.25
Tellurium Metals	0.00	0.05	0.05
Ba, Sr	0.00	0.02	0.02
Noble Metals	0.00	0.0025	0.0025
Cerium Group	0.00	0.0005	0.0005
Lanthanides	0.00	0.0002	0.0002

Conforms

For the LOCA event, the core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases in Table 1 were utilized.

For non-LOCA events, the fraction of the core inventory assumed to be in the gap by radionuclide group in Table 3 were utilized in conjunction with the maximum core radial peaking factor. The CRDA was evaluated per Note 11 of RG 1.183 (the gap fractions are assumed to be 10% for iodines and noble gases).

Per Siemens Power Corporation EMF-85-74(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," February 1088, including Supplement, 1(P)(A) and Supplement 2(P)(A) (Reference 13), the NRC approved licensing for ATRIUM-10 fuel is 62,000 MWD/MTU for extended burnup design – peak fuel rod exposure and 54,000 MWD/MTU for extended burnup design – peak bundle exposure. Thus, the requirement concerning the maximum linear heat generation rate not exceeding 6.3 kw/ft peak rod average power does not apply.

	For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor. **Table 3** **Non-LOCA Fraction of Fission Product Inventory in Gap** **Group** — **Fraction** I-131 — 0.08 Kr-85 — 0.10 Other Noble Gases — 0.05 Other Halogens — 0.05 Alkali Metals — 0.12 Note: Table 3 release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU. As an alternative, fission gas release calculations performed using NRC approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. For the BWR rod drop accident and the PWR rod ejection accident, the gap fractions are assumed to be 10% for iodines and noble gases.	
3.3	Timing of Release Phases Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at BWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase. For non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage.	Conforms Table 4 onset and duration of each sequential release phase for the DBA LOCA were utilized in the analysis in a linear manner.

	Table 4 **LOCA Release Phases** **Phase** / **Onset** / **Duration** Gap Release / 2 min / 0.5 hr Early In-Vessel / 0.5 hr / 1.5 hr In lieu of treating the release in a linear ramp manner, the activity for each phase can be modeled as being released instantaneously at the start of that release phase, i.e., in step increases. For facilities licensed with leak-before-break methodology, the onset of the gap release phase may be assumed to be 10 minutes. A licensee may propose an alternative time for the onset of the gap release phase, based on facility-specific calculations using suitable analysis codes or on an accepted topical report shown to be applicable to the specific facility. In the absence of approved alternatives, the gap release phase onsets in Table 4 should be used.	
3.4	Radionuclide Composition Table 5 lists the elements in each radionuclide group that should be considered in design basis analyses. **Table 5** **Radionuclide Groups** **Group** / **Elements** Noble Gases / Xe, Kr Halogens / I, Br Alkali Metals / Cs, Rb Tellurium Group / Te, Sb, Se, Ba, Sr Noble Metals / Ru, Rh, Pd, Mo, Tc, Co Lanthanides / La, Zr, Nd, Eu, Nb, Pm, Pr Sm, Y, Cm, Am Cerium Ce, Pu, Np	Conforms Table 5 elements in each radionuclide group were utilized in design basis analyses.
3.5	Chemical Form Of the radioiodines released from the reactor coolant system (RCS) to the containment in a postulated accident, 95 percent of the iodine released should be assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs	Conforms Of the radioiodines released from the reactor coolant system (RCS) to the containment in a postulated accident, 95 percent of the iodine released was assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide.

	other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.	With the exception of elemental and organic iodine and noble gases, fission products were assumed to be in particulate form. The same chemical form was assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs.
3.6	Fuel Damage in Non-LOCA DBAs The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases. The amount of fuel damage caused by a FHA is addressed in Appendix B of this guide.	Conforms The amount of fuel damage caused by non-LOCA design basis events was analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. The Recirculation Pump Seizure event is not included in this LAR and will be submitted as a separate document at a later date.
4.0	DOSE CALCULATION METHODOLOGY The NRC staff has determined that there is an implied synergy between the ASTs and total effective dose equivalent (TEDE) criteria, and between the TID-14844 source terms and the whole body and thyroid dose criteria, and therefore, they do not expect to allow the TEDE criteria to be used with TID-14844 calculated results. The guidance of this section applies to all dose calculations performed with an AST pursuant to 10 CFR 50.67. Certain selective implementations may not require dose calculations as described in Regulatory Position 1.3 of this guide.	Conforms The DBA analyses, based on ASTs, utilized the dose calculation methodology of Section 4 of RG 1.183.
4.1	Offsite Dose Consequences The following assumptions should be used in determining the TEDE for persons located at or beyond the boundary of the exclusion area (EAB):	Conforms The dose calculation methodology of Section 4.1 of RG 1.183 was utilized to calculate the offsite dose consequences.

4.1.1	The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides that are significant with regard to dose consequences and the released radioactivity. Note: The prior practice of basing inhalation exposure on only radioiodine and not including radioiodine in external exposure calculations is not consistent with the definition of TEDE and the characteristics of the revised source term.	Conforms The dose calculations determine the TEDE and consider all radionuclides, including progeny from the decay of parent radionuclides that are significant with regard to dose consequences.
4.1.2	The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 19). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 20), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE.	Conforms Conversion factors for isotopes other than the standard 60 isotopes of the RADTRAD computer program (default FGR 11 files) were taken from Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."
4.1.3	For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8×10^{-4} cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3×10^{-4} cubic meters per second.	Conforms Breathing rates provided in Section 4.1.3 of RG 1.183 were utilized to calculate the offsite dose consequences.
4.1.4	The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 21), provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.	Conforms Conversion factors for isotopes other than the standard 60 isotopes of the RADTRAD computer program (default FGR 12 files) were taken from Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil."

4.1.5	The TEDE should be determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67. The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release (see also Table 6). Note: With regard to the EAB TEDE, the maximum two-hour value is the basis for screening and evaluation under 10 CFR 50.59. Changes to doses outside of the two-hour window are only considered in the context of their impact on the maximum two-hour EAB TEDE.	Conforms The TEDE was determined for the most limiting person at the EAB. The maximum two-hour TEDE was determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods.
4.1.6	TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67.	Conforms The TEDE was determined for the most limiting receptor at the outer boundary of the low population zone (LPZ).
4.1.7	No correction should be made for depletion of the effluent plume by deposition on the ground.	Conforms No correction was made for depletion of the effluent plume by deposition on the ground.
4.2	Control Room Dose Consequences The following guidance should be used in determining the TEDE for persons located in the Control Room:	Conforms The DBA analyses utilized the guidance of Section 4.2 of RG 1.183 to determining the TEDE for persons located in the Control Room.
4.2.1	The TEDE analysis should consider all sources of radiation that will cause exposure to Control Room personnel. The applicable sources will vary from facility to facility, but typically will include: • Contamination of the Control Room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility.	Conforms The TEDE analysis considered all significant sources of radiation that will cause exposure to Control Room personnel.

	• Contamination of the Control Room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the Control Room envelope. • Radiation shine from the external radioactive plume released from the facility. • Radiation shine from radioactive material in the reactor containment. • Radiation shine from radioactive material in systems and components inside or external to the Control Room envelope, e.g., radioactive material buildup in recirculation filters.	
4.2.2	The radioactive material releases and radiation levels used in the Control Room dose analysis should be determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would result in non-conservative results for the Control Room.	Conforms The radioactive material releases and radiation levels used in the Control Room dose analysis were determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values as appropriate.
4.2.3	The models used to transport radioactive material into and through the Control Room, and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to Control Room personnel. The iodine protection factor (IPF) methodology of Reference 22 may not be adequately conservative for all DBAs and Control Room arrangements since it models a steady-state Control Room condition. Since many analysis parameters change over the duration of the event, the IPF methodology should only be used with caution. The NRC computer codes HABIT (Ref. 23) and RADTRAD (Ref. 24) incorporate suitable methodologies.	Conforms The models used to transport radioactive material into and through the Control Room, and the shielding models used to determine radiation dose rates from external sources, were developed to provide suitably conservative estimates of the exposure to Control Room personnel.
4.2.4	Credit for engineered safety features that mitigate airborne radioactive material within the Control Room may be assumed. Such features may include Control Room isolation or pressurization, or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP (Ref. 3) and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post-accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Ref. 25), for guidance. The Control Room design is often optimized for the DBA LOCA and the protection afforded for other accident sequences may not be as advantageous. In most designs, Control Room isolation is actuated by engineered safeguards feature (ESF) signals or radiation monitors (RMs). In some cases, the ESF signal is effective only for selected accidents, placing reliance on the RMs for the remaining	Conforms Credit for engineered safety features that mitigate airborne radioactive material within the Control Room were assumed (but only when a signal would automatically initiate the component/system), but based on conservative assumptions that increased the dose to the operator. Credit for engineered safety features varied for each of the analyzed DBAs.

	accidents. Several aspects of RMs can delay the Control Room isolation, including the delay for activity to build up to concentrations equivalent to the alarm setpoint and the effects of different radionuclide accident isotopic mixes on monitor response.	
4.2.5	Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.	Conforms Credit was not taken for the use of personal protective equipment or prophylactic drugs.
4.2.6	The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the Control Room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days. For the duration of the event, the breathing rate of this individual should be assumed to be 3.5×10^{-4} cubic meters per second. Note: This occupancy is modeled in the χ/Q values determined in Reference 22 and should not be credited twice. The ARCON96 Code (Ref. 26) does not incorporate these occupancy assumptions, making it necessary to apply this correction in the dose calculations.	Conforms Occupancy factors and breathing rate of RG 1.183, Section 4.2.6 were utilized to determine the doses to the hypothetical maximum exposed individual who is present in the Control Room. Control Room χ/Q values were determined utilizing the ARCON96 computer code or approved methodologies to calculate puff releases. Occupancy factors were included in the RADTRAD computer code for dose evaluations.
4.2.7	Control Room doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses. The DDE from photons may be corrected for the difference between finite cloud geometry in the Control Room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, DDE_{∞}, to a finite cloud dose, DDE_{finite}, where the Control Room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the Control Room (Ref. 22). $DDE_{finite} = \frac{DDE_{\infty} V^{0.338}}{1173}$ Equation 1	Conforms The DDE from photons was corrected for the difference between finite cloud geometry in the Control Room and the semi-infinite cloud assumption used in calculating the dose conversion factors by Equation 1 as necessary.

<table>
<tr>
<td>4.3</td>
<td>Other Dose Consequences

The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Ref. 2). Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.</td>
<td>Conforms

The guidance provided in Regulatory Positions 4.1 and 4.2 was used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737.

Exception – The TID-14844 source term will continue to be used as the basis for the equipment qualification program.</td>
</tr>
<tr>
<td>4.4</td>
<td>Acceptance Criteria

The radiological criteria for the EAB, the outer boundary of the LPZ, and for the Control Room are in 10 CFR 50.67. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. The Control Room criterion applies to all accidents. For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6.

The acceptance criteria for the various NUREG-0737 (Ref. 2) items generally reference General Design Criteria 19 (GDC 19) from Appendix A to 10 CFR Part 50 or specify criteria derived from GDC-19. These criteria are generally specified in terms of whole body dose, or its equivalent to any body organ. For facilities applying for, or having received, approval for the use of an AST, the applicable criteria should be updated for consistency with the TEDE criterion in 10 CFR 50.67(b)(2)(iii).

Table 6
Accident Dose Criteria

<table>
<tr><th>Accident or Case</th><th>AB and LPZ Dose Criteria</th><th>Analysis Release Duration</th></tr>
<tr><td>LOCA</td><td>25 rem TEDE</td><td>30 days for containment, ECCS, and MSIV (BWR) leakage</td></tr>
<tr><td>BWR Main Steam Line Break</td><td></td><td>Instantaneous puff</td></tr>
<tr><td>Fuel Damage or Pre-incident Spike</td><td>25 rem TEDE</td><td></td></tr>
<tr><td>Equilibrium Iodine Activity</td><td>2.5 rem TEDE</td><td></td></tr>
</table></td>
<td>Conforms

The DBAs were updated for consistency with the TEDE criterion in 10 CFR 50.67(b)(2)(iii).</td>
</tr>
</table>

	Accident or Case / **Dose Criteria** / **Analysis Release Duration** BWR Rod Drop Accident pathway / 6.3 rem TEDE / 24 hours Fuel Handling Accident / 6.3 rem TEDE / 2 hours The column labeled "Analysis Release Duration" is a summary of the assumed radioactivity release durations identified in the individual appendices to this guide. Refer to these appendices for complete descriptions of the release pathways and durations.	
5.0	ANALYSIS ASSUMPTIONS AND METHODOLOGY	
5.1	General Considerations	
5.1.1	Analysis Quality The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. These design basis analyses were structured to provide a conservative set of assumptions to test the performance of one or more aspects of the facility design. Many physical processes and phenomena are represented by conservative, bounding assumptions rather than being modeled directly. The staff has selected assumptions and models that provide an appropriate and prudent safety margin against unpredicted events in the course of an accident and compensate for large uncertainties in facility parameters, accident progression, radioactive material transport, and atmospheric dispersion. Licensees should exercise caution in proposing deviations based upon data from a specific accident sequence since the DBAs were never intended to represent any specific accident sequence – the proposed deviation may not be conservative for other accident sequences.	Conforms Analyses performed per 10 CFR 50, Appendix B and the guidance consistent with RG 1.183.
5.1.2	Credit for Engineered Safeguard Features Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by Technical Specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in	Conforms Credit was taken for Engineered Safeguard Features. The single active component failure that resulted in the most limiting radiological

	emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.	consequences was assumed. Assumptions regarding the occurrence and timing of a loss of offsite power were selected with the objective of maximizing the postulated radiological consequences.
5.1.3	Assignment of Numeric Input Values The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be non-conservative in another portion of the same analysis. For example, assuming minimum containment system spray flow is usually conservative for estimating iodine scrubbing, but in many cases may be non-conservative when determining sump pH. Sensitivity analyses may be needed to determine the appropriate value to use. As a conservative alternative, the limiting value applicable to each portion of the analysis may be used in the evaluation of that portion. A single value may not be applicable for a parameter for the duration of the event, particularly for parameters affected by changes in density. For parameters addressed by Technical Specifications, the value used in the analysis should be that specified in the technical specifications. If a range of values or a tolerance band is specified, the value that would result in a conservative postulated dose should be used. If the parameter is based on the results of less frequent surveillance testing, e.g., steam generator nondestructive testing (NDT), consideration should be given to the degradation that may occur between periodic tests in establishing the analysis value. Note that for some parameters, the Technical Specification value may be adjusted for analysis purposes by factors provided in other regulatory guidance. For example, ESF filter efficiencies are based on the guidance in Regulatory Guide 1.52 (Ref. 25) and in Generic Letter 99-02 (Ref. 27) rather than the surveillance test criteria in the Technical Specifications. Generally, these adjustments address potential changes in the parameter between scheduled surveillance tests.	Conforms The numeric values that were chosen as inputs to the analyses required by 10 CFR 50.67 were selected with the objective of determining a conservative postulated dose. For a range of values, the value that resulted in a conservative postulated dose was used.
5.1.4	Applicability of Prior Licensing Basis The NRC staff considers the implementation of an AST to be a significant change to the design basis of the facility that is voluntarily initiated by the licensee. In order to issue a license amendment authorizing the use of an AST and the TEDE dose criteria, the NRC staff must make a current finding of compliance with regulations applicable to the amendment. The characteristics of the ASTs and the revised dose calculational methodology may be incompatible with many of the analysis assumptions and methods currently reflected in the facility's design basis analyses. The NRC staff may find that new or unreviewed issues are created by a particular site-specific	Conforms Licensee has ensured that analysis assumptions and methods are compatible with the AST and the TEDE criteria.

	implementation of the AST, warranting review of staff positions approved subsequent to the initial issuance of the license. This is not considered a backfit as defined by 10 CFR 50.109, "Backfitting." However, prior design bases that are unrelated to the use of the AST, or are unaffected by the AST, may continue as the facility's design basis. Licensees should ensure that analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.	
5.2	Accident-Specific Assumptions The appendices to this regulatory guide provide accident-specific assumptions that are acceptable to the staff for performing analyses that are required by 10 CFR 50.67. The DBAs addressed in these attachments were selected from accidents that may involve damage to irradiated fuel. This guide does not address DBAs with radiological consequences based on technical specification reactor or secondary coolant-specific activities only. The inclusion or exclusion of a particular DBA in this guide should not be interpreted as indicating that an analysis of that DBA is required or not required. Licensees should analyze the DBAs that are affected by the specific proposed applications of an AST. The NRC staff has determined that the analysis assumptions in the appendices to this guide provide an integrated approach to performing the individual analyses and generally expects licensees to address each assumption or propose acceptable alternatives. Such alternatives may be justifiable on the basis of plant-specific considerations, updated technical analyses, or, in some cases, a previously approved licensing basis consideration. The assumptions in the appendices are deemed consistent with the AST identified in Regulatory Position 3 and internally consistent with each other. Although licensees are free to propose alternatives to these assumptions for consideration by the NRC staff, licensees should avoid use of previously approved staff positions that would adversely affect this consistency. The NRC is committed to using probabilistic risk analysis (PRA) insights in its regulatory activities and will consider licensee proposals for changes in analysis assumptions based upon risk insights. The staff will not approve proposals that would reduce the defense in depth deemed necessary to provide adequate protection for public health and safety. In some cases, this defense in depth compensates for uncertainties in the PRA analyses and addresses accident considerations not adequately addressed by the core damage frequency (CDF) and large early release frequency (LERF) surrogate indicators of overall risk.	Conforms Licensee analyzed the DBAs that are affected by the specific proposed applications of an AST, utilizing the guidance provided in the appendices of RG 1.183.

5.3	Meteorology Assumptions Atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the Control Room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. Methodologies that have been used for determining χ/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19" (Refs. 6, 7, 22, and 28). References 22 and 28 should be used if the FSAR χ/Q values are to be revised or if values are to be determined for new release points or receptor distances. Fumigation should be considered where applicable for the EAB and LPZ. For the EAB, the assumed fumigation period should be timed to be included in the worst 2-hour exposure period. The NRC computer code PAVAN (Ref. 29) implements Regulatory Guide 1.145 (Ref. 28) and its use is acceptable to the NRC staff. The methodology of the NRC computer code ARCON96 (Ref. 26) is generally acceptable to the NRC staff for use in determining Control Room χ/Q values. Meteorological data collected in accordance with the site-specific meteorological measurements program described in the facility FSAR should be used in generating accident χ/Q values. Additional guidance is provided in Regulatory Guide 1.23, "Onsite Meteorological Programs" (Ref. 30). All changes in χ/Q analysis methodology should be reviewed by the NRC staff.	Conforms New χ/Qs for the EAB, the LPZ, and the Control Room were developed, based on more recent meteorological data, utilizing the guidance of RGs 1.23, 1.145, 1.194. The ABS Consulting, Inc. WINDOWS computer code and the PAVAN computer code were used in developing the offsite χ/Qs. The ARCON96 computer program was utilized in developing the CRHE χ/Qs. The Control Room air intake was also relocated. The MSLB Control Room χ/Q was calculated utilizing the puff methodology of RG 1.194.
6.0	ASSUMPTIONS FOR EVALUATING THE RADIATION DOSES FOR EQUIPMENT QUALIFICATION The assumptions in Appendix I to this guide are acceptable to the NRC staff for performing radiological assessments associated with equipment qualification. The assumptions in Appendix I will supersede Regulatory Positions 2.c(1) and 2.c(2) and Appendix D of Revision 1 of Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants" (Ref. 11), for operating reactors that have amended their licensing basis to use an alternative source term. Except as stated in Appendix I, all other assumptions, methods, and provisions of Revision 1 of Regulatory Guide 1.89 remain effective. The NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted. Until such time as this generic issue is resolved, licensees may use either the AST or the TID14844 assumptions for performing the required EQ analyses. However, no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST vs. TID14844) on EQ doses pending the outcome of the evaluation of the generic issue.	N/A. An AST assessment was not performed for equipment qualification. TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," will continue to be used as the radiation dose basis for equipment qualification and radiation zone maps/shielding calculations.

D.	**IMPLEMENTATION** The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide. Except in those cases in which an applicant or licensee proposes an acceptable alternative method for complying with the specified portions of the NRC's regulations, the methods described in this guide will be used in the evaluation of submittals related to the use of ASTs in radiological consequence analyses at operating power reactors.	Conforms The AST analysis utilized the guidance of RG 1.183 in the AST evaluations and did not use alternative method for complying with the specified portions of the NRC's regulations.
Appendix A	**ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LWR LOSS-OF-COOLANT ACCIDENT** The assumptions in this appendix are acceptable to the NRC staff for evaluating the radiological consequences of loss-of-coolant accidents (LOCAs) at light water reactors (LWRs). These assumptions supplement the guidance provided in the main body of this guide. Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 defines LOCAs as those postulated accidents that result from a loss of coolant inventory at rates that exceed the capability of the reactor coolant makeup system. Leaks up to a double-ended rupture of the largest pipe of the reactor coolant system are included. The LOCA, as with all design basis accidents (DBAs), is a conservative surrogate accident that is intended to challenge selective aspects of the facility design. Analyses are performed using a spectrum of break sizes to evaluate fuel and ECCS performance. With regard to radiological consequences, a large-break LOCA is assumed as the design basis case for evaluating the performance of release mitigation systems and the containment and for evaluating the proposed siting of a facility.	Conforms An analysis was performed utilizing the guidance of Appendix A and appropriate sections in the main body of RG 1.183 to evaluate a LOCA.
	SOURCE TERM ASSUMPTIONS	
1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms Assumptions regarding core inventory and the release of radionuclides from the fuel were per Regulatory Position 3 of RG 1.183.
2	If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine species, including those from iodine re-evolution, for sump or suppression pool pH values less than 7 will be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids and bases created during the LOCA event, e.g., radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.	Conforms The sump or suppression pool pH is controlled at values of 7 or greater.

	ASSUMPTIONS ON TRANSPORT IN PRIMARY CONTAINMENT	
3	Acceptable assumptions related to the transport, reduction, and release of radioactive material in and from the drywell in BWRs are as follows:	Conforms The analysis utilized the assumptions related to the transport, reduction, and release of radioactive material in and from the drywell in BWRs per RG 1.183.
3.1	The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the drywell in BWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell to the wetwell. The release into the drywell should be assumed to terminate at the end of the early in-vessel phase.	Conforms The radioactivity released from the fuel was assumed to mix instantaneously and homogeneously throughout the free air volume of the drywell as it was released. This distribution was not adjusted because ventilation exchange is not limited. The suppression pool free air volume was included after 2 hours. The release into the drywell was terminated at the end of the early in-vessel phase.
3.2	Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the Standard Review Plan (SRP), NUREG-0800 (Ref. A-1) and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Ref. A-2). The latter model is incorporated into the analysis code RADTRAD (Ref. A-3). The prior practice of deterministically assuming that a 50% plateout of iodine is released from the fuel is no longer acceptable to the NRC staff as it is inconsistent with the characteristics of the revised source terms.	Conforms The 10^{th} percentile Power's Aerosol Decontamination Model was conservatively used in the analysis for natural deposition.
3.3	Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP (Ref. A-1) may be credited. Acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays"1 (Ref. A-4). This simplified model is incorporated into the analysis code RADTRAD (Refs. A-1 to A-3).	N/A No credit is conservatively taken in this analysis for fission product reduction due to the initiation of the drywell sprays.

	The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown. The SRP sets forth a maximum decontamination factor (DF) for elemental iodine based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached. The reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays. The maximum activity to be used in determining the DF is defined as the iodine activity in the columns labeled "Total" in Tables 1 and 2 of this guide multiplied by 0.05 for elemental iodine and by 0.95 for particulate iodine (i.e., aerosol treated as particulate in SRP methodology). This document describes statistical formulations with differing levels of uncertainty. The removal rate constants selected for use in design basis calculations should be those that will maximize the dose consequences. For BWRs, the simplified model should be used only if the release from the core is not directed through the suppression pool. Iodine removal in the suppression pool affects the iodine species assumed by the model to be present initially.	
3.4	Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. A-5 and A-6). The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.	N/A. No credit is taken in this analysis for reduction of airborne radioactivity in the containment by in-containment recirculation filter systems.
3.5	Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs should generally not be credited. However, the staff may consider such reduction on an individual case basis. The evaluation should consider the relative timing of the blowdown and the fission product release from the fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool (Ref. 7). Analyses should consider iodine re-evolution if the suppression pool liquid pH is not maintained greater than 7.	Conforms No credit is taken in this analysis for reduction of airborne radioactivity in the containment by suppression pool scrubbing.
3.6	Reduction in airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above, should be evaluated on an individual case basis. See Section 6.5.4 of the SRP (Ref. A-1).	N/A

		No credit is taken in this analysis for reduction of airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above.
3.7	The primary containment (i.e., drywell for Mark I and II containment designs) should be assumed to leak at the peak pressure Technical Specification leak rate for the first 24 hours. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50% of the Technical Specification leak rate. Leakage from sub-atmospheric containments is assumed to terminate when the containment is brought to and maintained at a sub-atmospheric condition as defined by Technical Specifications. For BWRs with Mark III containments, the leakage from the drywell into the primary containment should be based on the steaming rate of the heated reactor core, with no credit for core debris relocation. This leakage should be assumed during the two-hour period between the initial blowdown and termination of the fuel radioactivity release (gap and early in-vessel release phases). After two hours, the radioactivity is assumed to be uniformly distributed throughout the drywell and the primary containment.	Conforms Per SSES Technical Specifications, Section 3.6.1.1 and Primary Containment and Technical Specification Bases, B3.6.1.1, the leakage rate for the primary containment is defined as 1% by weight of containment air per 24 hours @ 45 psig. In accordance with RG 1.183, Section 3.7: "The primary containment (i.e., drywell for Mark I and II containment designs) should be assumed to leak at the peak pressure Technical Specification leak rate for the first 24 hours. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50% of the Technical Specification leak rate". Based on the significant reduction of the calculated internal pressure of the primary containment at 24 hours into the LOCA, the 50% reduction in leak rate was utilized in the analysis.
3.8	If the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The purge release evaluation should assume that 100% of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the Technical Specification reactor coolant system equilibrium activity. Iodine spikes need not be considered. If the purge system is not isolated before the onset of the gap release phase, the release fractions associated with the gap release and early in-vessel phases should be considered as applicable.	N/A. The primary containment is not routinely purged during power operations.

	ASSUMPTIONS ON DUAL CONTAINMENTS	
4	For facilities with dual containment systems, the acceptable assumptions related to the transport, reduction, and release of radioactive material in and from the secondary containment or enclosure buildings are as follows:	Conforms The LOCA analysis utilized the assumptions related to the transport, reduction, and release of radioactive material in and from the secondary containment of Section 4 of RG 1.183.
4.1	Leakage from the primary containment should be considered to be collected, processed by engineered safety feature (ESF) filters, if any, and released to the environment via the secondary containment exhaust system during periods in which the secondary containment has a negative pressure as defined in Technical Specifications. Credit for an elevated release should be assumed only if the point of physical release is more than two and one-half times the height of any adjacent structure.	Conforms SSES Units 1 & 2 proposed Technical Specification B 3.6.4.1.4 state that one SGTS subsystem should draw down the Reactor Building including Zones I, II and III or Zones I and III to greater than or equal to 0.25 inch of vacuum water gauge in less than or equal to 300 seconds. A 10 minute Reactor Building drawdown time is conservatively used in the LOCA analysis. The release is assumed to occur at ground level.
4.2	Leakage from the primary containment is assumed to be released directly to the environment as a ground-level release during any period in which the secondary containment does not have a negative pressure as defined in Technical Specifications.	Conforms Prior to the completion of the secondary containment drawdown, primary containment leakage, excluding the portion of the leakage which bypasses secondary containment, is assumed to be released to the environment without credit for filtration by the SGTS. The release is assumed to occur at ground level.
4.3	The effect of high wind speeds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5% of the total number of hours in the data set. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded only 5% or 95% of the total numbers of hours in the data set, whichever is conservative for the intended use (e.g., if high temperatures are limiting, use those exceeded only 5%).	Conforms Certified meteorological data consisting of wind speeds of 1-hour average value that is exceeded only 5% of the total number of hours in the data set was utilized. Ambient temperatures used in these assessments were

		1-hour average values that are exceeded only 5% or 95% of the total numbers of hours in the data set.
4.4	Credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise, the leakage from the primary containment should be assumed to be transported directly to exhaust systems without mixing. Credit for mixing, if found to be appropriate, should generally be limited to 50%. This evaluation should consider the magnitude of the containment leakage in relation to contiguous building volume or exhaust rate, the location of exhaust plenums relative to projected release locations, the recirculation ventilation systems, and internal walls and floors that impede stream flow between the release and the exhaust.	Conforms Credit for dilution in the secondary containment was included in the LOCA analysis and limited to 50% of the minimum free air volume.
4.5	Primary containment leakage that bypasses the secondary containment should be evaluated at the bypass leak rate incorporated in the Technical Specifications. If the bypass leakage is through water, e.g., via a filled piping run that is maintained full, credit for retention of iodine and aerosols may be considered on a case-by-case basis. Similarly, deposition of aerosol radioactivity in gas-filled lines may be considered on a case-by-case basis.	Conforms The bypass leak rate was based on Technical Specification SR 3.6.1.3.11 and a PPL station procedure. The bypass leakage is not through water. Deposition of aerosol radioactivity in gas-filled lines was not considered.
4.6	Reduction in the amount of radioactive material released from the secondary containment because of ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms After secondary containment drawdown, leakage through the SGTS was credited. Filter efficiency was based on RG 1.52 and GL 99-02.
	ASSUMPTIONS ON ESF SYSTEM LEAKAGE	
5	ESF systems that recirculate sump water outside of the primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. This release source may also include leakage through valves isolating interfacing systems (Ref. A-7). The radiological consequences from the postulated leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions are acceptable for evaluating the consequences of leakage from ESF components outside the primary containment:	Conforms This analysis utilized the ESF leakage assumptions in Section 5 of RG 1.183.
5.1	With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Table 1 of this guide) should be assumed to instantaneously and homogeneously mix in the suppression pool (in BWRs) at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in	Conforms With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Table 1 of

	containment to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are non-conservative with regard to the buildup of sump activity.	RG 1.183) were assumed to instantaneously and homogeneously mix in the suppression pool for this analysis.
5.2	The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the Technical Specifications, or licensee commitments to Item III.D.1.1 of NUREG-0737 (Ref. A-8), would require declaring such systems inoperable. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump miniflow return to the refueling water storage tank.	Conforms There is no TS limit for ESF leakage outside primary containment. TS 5.5.2, "Primary Sources Outside Containment" discusses a program that provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. PPL nuclear department quality assurance procedure and numerous Surveillance Engineering Procedures declare such systems inoperable for ESF leakage when the leak rate exceeds 5 gpm. This meets FSAR Commitments 15.6.5.5.1.2 and 18.1.69. These procedures will be revised to reflect a value of 2.5 gpm (see Section 8.0 concerning regulatory commitments) before implementation of the AST. An additional leakage of 15 gpm was assumed through the control rod drives and SDV (not ESF) but included in this evaluation for conservative reasons). Consequently, the analysis used an assumed leakage of 20 gpm (2 X 2.5 + 15).
5.3	With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.	Conforms With the exception of iodine, all radioactive materials in the recirculating liquid were assumed to be retained in the liquid phase.

5.4	If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, h, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment: $$FF = \frac{h_{f_1} - h_{f_2}}{h_{fg}}$$ *Where:* h_{f1} is the enthalpy of liquid at system design temperature and pressure; h_{f2} is the enthalpy of liquid at saturation conditions (14.7 psia, 212°F); and h_{fg} is the heat of vaporization at 212°F.	N/A. The temperature of the leakage does not exceed 212°F.
5.5	If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.	Conforms The temperature of the leakage does not exceed 212°F and a flash fraction of 10% was assumed for the iodine.
5.6	The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms The radioiodine that is postulated to be available for release to the environment was assumed to be 97% elemental and 3% organic. Following drawdown of secondary containment, credit is taken for the filtration of secondary containment releases by the SGTS. The SGTS complies with the guidance of RG 1.52 and Generic letter 99-02.
	ASSUMPTIONS ON MAIN STEAM ISOLATION VALVE LEAKAGE IN BWRS	
6	For BWRs, the main steam isolation valves (MSIVs) have design leakage that may result in a radioactivity release. The radiological consequences from postulated MSIV leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions are acceptable for evaluating the consequences of MSIV leakage.	Conforms The MSIV leakage pathway was included with the other identified leakage pathways to determine the total calculated radiological consequences from the LOCA.

6.1	For the purpose of this analysis, the activity available for release via MSIV leakage should be assumed to be that activity determined to be in the drywell for evaluating containment leakage (see Regulatory Position 3). No credit should be assumed for activity reduction by the steam separators or by iodine partitioning in the reactor vessel.	Conforms For this analysis, the activity available for release via MSIV leakage was assumed to be that activity determined to be in the drywell for evaluating containment leakage. No credit was assumed for activity reduction by the steam separators or by iodine partitioning in the reactor vessel. The wetwell free air volume was included with the drywell free air volume after 2 hours.
6.2	All the MSIVs should be assumed to leak at the maximum leak rate above which the Technical Specifications would require declaring the MSIVs inoperable. The leakage should be assumed to continue for the duration of the accident. Postulated leakage may be reduced after the first 24 hours, if supported by site-specific analyses, to a value not less than 50% of the maximum leak rate.	Conforms The MSIVs were assumed to leak at the maximum leak rate above which the Technical Specifications would require declaring the MSIVs inoperable (≤ 100 SCFH from any one valve or ≤ 300 SCFH total from four valves). The SSES analysis assumes one MSIV is faulted with a flow of 100 SCFH. The remaining flow is evenly split between the remaining MSIV's. The leakage was assumed to continue for the duration of the accident. Postulated leakage was reduced after the first 24 hours by 50% of the maximum leak rate. See Section 3.7 of Appendix A above.
6.3	Reduction of the amount of released radioactivity by deposition and plateout on steam system piping upstream of the outboard MSIVs may be credited, but the amount of reduction in concentration allowed will be evaluated on an individual case basis. Generally, the model should be based on the assumption of well-mixed volumes, but other models such as slug flow may be used if justified.	Conforms The model for the removal of iodine in the main steam line only assumes horizontal runs of piping in determining plateout. Additionally, since aerosol plateout is a mechanistic settling process, only the bottom ½ of the inside surface area of the lines is applicable for plateout. Since the bottom half of a circular pipe has sides which are inclined, the area for aerosol plateout is modeled as the

		projected area of the diameter of the pipe [diameter X length] in lieu of the actual surface area [π X diameter X length]. Another potential issue related to the phenomenon of aerosol settling is the possibility for steam condensation in the piping to wash out and re-evolve some of the settled aerosols. While the actual process would be difficult to quantify, a factor of 2 reduction in the conservatively calculated projected pipe surface area is used in RADTRAD to provide additional margin. Therefore, the aerosol settling area is defined as ½ X pipe diameter X length. The aerosol settling velocity is conservatively set equal to ¼ of the 10th percentile value from AEB 98-03. Only the portion of the main steam lines from the primary containment isolation valve to the point where the drain piping takes off is used for plateout. Plateout is not credited in the faulted line.
6.4	In the absence of collection and treatment of releases by ESFs such as the MSIV leakage control system, or as described in paragraph 6.5 below, the MSIV leakage should be assumed to be released to the environment as an unprocessed, ground- level release. Holdup and dilution in the Turbine Building should not be assumed.	Conforms The MSIV leakage was assumed to be released to the environment as an unprocessed, ground-level release. Holdup and dilution in the Turbine Building was not assumed.
6.5	A reduction in MSIV releases that is due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by offgas systems, may be credited if the components and piping systems used in the release path are capable of performing their safety function during and following a safe shutdown earthquake (SSE). The amount of reduction allowed will be evaluated on an individual case basis. References A-9 and A-10 provide guidance on acceptable models.	Conforms A reduction in MSIV releases that is due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser was credited, based on References A-9 and A-10.

	ASSUMPTION ON CONTAINMENT PURGING	
7	The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. Reduction in the amount of radioactive material released via ESF filter systems may be taken into account provided that these systems meet the guidance in Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	N/A. Primary containment purging as a combustible gas or pressure control measure was analyzed and not deemed to be required. Containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis.
Appendix B	**ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT** This appendix provides assumptions acceptable to the staff for evaluating the radiological consequences of a fuel handling accident at light water reactors. These assumptions supplement the guidance provided in the main body of this guide.	Conforms An analysis was performed utilizing the guidance of Appendix B and appropriate sections in the main body of RG 1.183 to evaluate a fuel and an equipment accident.
1	SOURCE TERM Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide. The following assumptions also apply.	Conforms Assumptions regarding core inventory and the release of radionuclides from the fuel are taken from Regulatory Position 3 of RG 1.183.
1.1	The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.	Conforms Case 1 (FHA) considers a dropped fuel assembly unit (fuel assembly, channel, grapple and mast) weighing 1500 lbs. falling a distance of 32.95 feet onto the core. The number of failed rods is given as 156 rods for the Atrium 10 fuel assemblies. To conservatively address the issue of lead fuel assemblies (whether in the reactor or the in the spent fuel pool) radiological dose results are included which assume that another Atrium 10 assembly representing a lead use assembly (LUA) completely fails resulting in a total of 254.8 failed rods.

		Case 2 (EHA) assumes an object weighing 1100 lbs. is dropped 150 feet onto the core. The dropped assembly and the core are assumed to be ATRIUM-10 fuel assemblies. The number of failed rods is given as 366 rods considering the Atrium 10 fuel assemblies and 460.8 rods considering the case for Atrium 10 + 1 LUA.
1.2	The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.	Conforms The fission product release from the breached fuel is based on Regulatory Position 3.2 of RG 1.183 and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that were considered include xenons, kryptons, halogens, cesiums, and rubidiums. Please note, RG 1.183, Appendix B, Section 3, the pool DF for particulates which includes cesiums, and rubidiums is infinite. Therefore, they are neglected from further consideration in the analysis.
1.3	The chemical form of radioiodine released from the fuel to the spent fuel pool should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel is assumed to completely dissociate in the pool water. Because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This is assumed to occur instantaneously. The NRC staff will consider, on a case-by-case basis, justifiable mechanistic treatment of the iodine release from the pool.	Conforms All iodine released to the spent fuel pool dissociates and re-evolves as elemental iodine instantaneously.
2	WATER DEPTH If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and	The minimum depth of water above the fuel is 21 feet. Utilizing the methodology referenced in RG 1.183, Appendix B (Staff Technical Paper, Evaluation of Fission Product Release and Transport, G. Burley, 1971), an overall pool DF of 138.46 (used 138) was calculated.

	organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method (Ref. B-1).	The actual DF calculation is included in Attachment 10, Calculation EC-RADN-1126, Section 3.14. The maximum fuel rod pressurization is < 1200 psig. Therefore, pressure does not impact the pool DF used in analysis. The iodine speciation above the water for the 21' depth was calculated to be 79% elemental and 21% organic iodine.
3	NOBLE GASES The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).	Conform Noble gas DF = 1 Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).
4	FUEL HANDLING ACCIDENTS WITHIN THE FUEL BUILDING For fuel handling accidents postulated to occur within the fuel building, the following assumptions are acceptable to the NRC staff.	Conforms Fuel handling and an equipment handling accidents were evaluated within the Reactor Building outside containment.
4.1	The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period.	Conforms The radioactive material that escapes from the fuel pool to the Reactor Building is assumed to be released to the environment over a 2-hour time period.
4.2	A reduction in the amount of radioactive material released from the fuel pool by engineered safety feature (ESF) filter systems may be taken into account provided these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2, B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses Note: These analyses should consider the time for the radioactivity concentration to reach levels corresponding to the monitor setpoint, instrument line sampling time, detector response time, diversion damper alignment time, and filter system actuation, as applicable.	Conforms The activity transport from the pool to the environment is via the SGTS filters. Based on a conservative analysis using realistic assumptions and parameters, it was determined that the Refueling Floor High Exhaust Duct Radiation Monitors, Refueling Floor Wall Exhaust Duct Radiation Monitors and the

		Railroad Access Shaft Exhaust Duct Radiation Monitor will sense the FHA/EHA event and provide the required signals to the SGTS. An analysis demonstrated that the isolation damper closure time is less than the air travel time. Therefore, the isolation damper will close prior to the activity reaching the damper. The Standby Gas Treatment System charcoal filters provide 8 inches of charcoal for filtration. Per Table 2 of Regulatory Guide 1.52, an SGTS charcoal filter efficiency of 99% is assumed for all species of iodine.
4.3	The radioactivity release from the fuel pool should be assumed to be drawn into the ESF filtration system without mixing or dilution in the fuel building. If mixing can be demonstrated, credit for mixing and dilution may be considered on a case-by-case basis. This evaluation should consider the magnitude of the building volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the pool, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the pool and the exhaust plenums.	Conforms No mixing or dilution in the Reactor Building assumed.
5	FUEL HANDLING ACCIDENTS WITHIN CONTAINMENT For fuel handling accidents postulated to occur within the containment, the following assumptions are acceptable to the NRC staff.	N/A. Fuel handling and an equipment handling accidents were evaluated within the Reactor Building outside containment.
5.1	If the containment is isolated during fuel handling operations, no radiological consequences need to be analyzed. Note: Containment isolation does not imply containment integrity as defined by Technical Specifications for non-shutdown modes. The term isolation is used here collectively to encompass both containment integrity and containment closure, typically in place during shutdown periods. To be credited in the analysis, the appropriate form of isolation should be addressed in Technical Specifications.	N/A. Fuel handling and an equipment handling accidents were evaluated within the Reactor Building outside containment.
5.2	If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment, no radiological consequences need to be analyzed.	N/A. Fuel handling and an equipment handling accidents were evaluated within the Reactor Building outside containment.

5.3	If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open), the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period. Note: The staff will generally require that technical specifications allowing such operations include administrative controls to close the airlock, hatch, or open penetrations within 30 minutes. Such administrative controls will generally require that a dedicated individual be present, with necessary equipment available, to restore containment closure should a fuel handling accident occur. Radiological analyses should generally not credit this manual isolation.	N/A. Fuel handling and an equipment handling accidents were evaluated within the Reactor Building outside containment.
5.4	A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2 and B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.	N/A. Fuel handling and an equipment handling accidents were evaluated within the Reactor Building outside containment.
5.5	Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50% of the containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the reactor cavity and the exhaust plenums.	N/A. Fuel handling and an equipment handling accidents were evaluated within the Reactor Building outside containment.
Appendix C	**ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A BWR ROD DROP ACCIDENT** This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a rod drop accident at BWR light-water reactors. These assumptions supplement the guidance provided in the main body of this guide.	Conforms An analysis was performed utilizing the guidance of Appendix C and appropriate sections in the main body of RG 1.183 at full power, assuming fuel rod breach and melt. Please note, a second postulated accident was evaluated at low power operation with the mechanical vacuum pump (MVP) running. At low power with fewer than 30 rods failing, main steam line dose rates may be too low to be reliably sensed by the MSLRMs to generate a trip signal for the MVP. Failure to trip the MVP would result in an unfiltered release of

		fission products to the environment from the Turbine Building vent stack. A detailed discussion of this event is given in Attachment 10, Calculation EC-RADN-1127 of this submittal.
1	Assumptions acceptable to the NRC staff regarding core inventory are provided in Regulatory Position 3 of this guide. For the rod drop accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and on the assumption that 100% of the noble gases and 50% of the iodines contained in that fraction are released to the reactor coolant.	Conforms The analysis conservatively assumes that 2000 rods experience cladding damage. Currently, 1000 rods are assumed in the CRDA design basis accident for the pre-power uprate. Compliance check calculations are performed for each unit and cycle to verify that the number of rods is less than 1000. For these compliance checks, rods with 170 cal/gm or more energy deposited in the fuel are assumed to fail. It is expected that that the number of rods will not substantially change based on favorable results from compliance checking analyses and on scoping studies performed for extended power uprate. In order to establish a conservative bound for assessing accident doses, it is assumed that 2000 fuel rods fail. Although no fuel melt occurs at SSES for this event, the analysis assumes 0.77% of the fuel melts. This fuel melt assumption is intended to ensure compatibility with the same assumption made in GE's Topical Report NEDO-31400A, which evaluated the elimination of certain main steam radiation monitor (MSLRMs) safety functions. For the breached fuel rods, it is assumed that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting assumes that 100% of the noble gases and 50% of the iodines contained in that fraction are released to the reactor coolant.

		Please note, that in RAI 21 for the Limerick AST, the NRC recommends a typical value of 1% for failed fuel. Although the analysis utilizes a value of 0.77% (described above), PPL believes that the source term conservatisms in the calculation; (1) 2000 rods experience clad damage instead of 1000 rods, (2) no fuel melt at SSES, and (3) inclusion of the solids released from the melted fuel are in accordance with the fractions shown in Regulatory Guide 1.183, Table 1 provide more than sufficient margin. In addition, the resultant doses are only 0.49, 0.19, and 0.05 rem TEDE at the Control Room, EAB, and LPZ respectively. An increase in the percent of the melted fuel from 0.77% to 1.0% would not alter acceptance to regulatory guidelines.
2	If no or minimal fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity (typically 4 μCi/gm DE I-131) allowed by the Technical Specifications. The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum Technical Specification values, whichever maximizes the radiological consequences. In determining the dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.	N/A. Postulated accident assumes fuel clad failure and fuel melt.
3	The assumptions acceptable to the NRC staff that are related to the transport, reduction, and release of radioactive material from the fuel and the reactor coolant are as follows:	Conforms Transport assumptions of RG 1.183, Appendix C utilized in the analysis.
3.1	The activity released from the fuel from either the gap or from fuel pellets is assumed to be instantaneously mixed in the reactor coolant within the pressure vessel.	Conforms The activity released from the fuel from either the gap or from fuel pellets is assumed to be instantaneously mixed in the reactor coolant within the pressure vessel.

3.2	Credit should not be assumed for partitioning in the pressure vessel or for removal by the steam separators.	Conforms No credit assumed for partitioning in the pressure vessel or for removal by the steam separators.
3.3	Of the activity released from the reactor coolant within the pressure vessel, 100% of the noble gases, 10% of the iodine, and 1% of the remaining radionuclides are assumed to reach the turbine and condensers.	Conforms Of the activity released from the reactor coolant within the pressure vessel, 100% of the noble gases, 10% of the iodine, and 1% of the remaining radionuclides are assumed to reach the turbine and condensers.
3.4	Of the activity that reaches the turbine and condenser, 100% of the noble gases, 10% of the iodine, and 1% of the particulate radionuclides are available for release to the environment. The turbine and condensers leak to the atmosphere as a ground- level release at a rate of 1% per day for a period of 24 hours, at which time the leakage is assumed to terminate. No credit should be assumed for dilution or holdup within the Turbine Building. Radioactive decay during holdup in the turbine and condenser may be assumed. If there are forced flow paths from the turbine or condenser, such as unisolated motor vacuum pumps or unprocessed air ejectors, the leakage rate should be assumed to be the flow rate associated with the most limiting of these paths. Credit for collection and processing of releases, such as by off gas or standby gas treatment, will be considered on a case-by-case basis.	Conforms Of the activity that reaches the turbine and condenser, 100% of the noble gases, 10% of the iodine, and 1% of the particulate radionuclides are available for release to the environment. The turbine and condensers leak to the atmosphere as a ground- level release at a rate of 1% per day for a period of 24 hours, at which time the leakage is assumed to terminate. No credit assumed for dilution or holdup within the Turbine Building. Radioactive decay during holdup in the turbine and condenser was assumed. A second postulated accident was evaluated at low power operation with the mechanical vacuum pump running. The leakage pathway is unfiltered through the TB vent (assumes ground level release) at a leakage rate of 1212 percent per day from the condenser with no holdup in the TB.

3.5	In lieu of the transport assumptions provided in Paragraphs 3.2 through 3.4 above, a more mechanistic analysis may be used on a case-by-case basis. Such analyses account for the quantity of contaminated steam carried from the pressure vessel to the turbine and condensers based on a review of the minimum transport time from the pressure vessel to the first main steam isolation valve (MSIV) and considers MSIV closure time.	N/A. The MSIVs and main steam drain lines do not automatically trip closed following a CRDA. The operators manually scram the reactor and close all MSIVs and drain line valves in the event of a main steam line radiation monitor (MSLRM) high-high radiation alarm in accordance with station procedures. The MSLRMs in the original SSES design provided input to automatically trip the MSIVs and drain line valves closed. These MSLRM trip functions were eliminated as part of BWR generic efforts to reduce spurious reactor scrams and reactor vessel isolations (50.59 Safety Evaluation E-01-1, MSLRM MSIV Closure and Scram Deletion, June 19, 2003 – Amendment 151 to facility operating license NPF-14 and Amendment 121 to facility operating license NPF-22).
3.6	The iodine species released from the reactor coolant within the pressure vessel should be assumed to be 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic. The release from the turbine and condenser should be assumed to be 97% elemental and 3% organic.	Conforms The release from the turbine and condenser was 97% elemental and 3% organic.
Appendix D	**ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A BWR MAIN STEAM LINE BREAK ACCIDENT** This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a main steam line accident at BWR light water reactors. These assumptions supplement the guidance provided in the main body of this guide.	Conforms An analysis was performed utilizing the guidance of Appendix D and appropriate sections in the main body of RG 1.183 at hot standby, assuming no melt.

Regulatory Guide 1.183 Compliance Table

1	SOURCE TERM Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	Conforms Assumptions regarding core inventory and the release of radionuclides from the fuel are based on Regulatory Position 3 of RG 1.183. No fuel rod breach was assumed.
2	If no or minimal fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity allowed by Technical Specification. The iodine concentration in the primary coolant is assumed to correspond to the following two cases in the nuclear steam supply system vendor's standard Technical Specifications. The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum Technical Specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DE I- 131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.	Conforms The hypothetical event was based on the maximum coolant activity allowed by Technical Specification. In determining dose equivalent I-131 (DE I- 131), only the radioiodine associated with normal operations or iodine spikes were included.
2.1	The concentration that is the maximum value (typically 4.0 μCi/gm DE I-131) permitted and corresponds to the conditions of an assumed pre-accident spike, and	Conforms The analysis utilized the maximum concentration value (4.0 μCi/gm DE I-131) permitted per TS and corresponds to the conditions of an assumed pre-accident spike.
2.1	The concentration that is the maximum equilibrium value (typically 0.2 μCi/gm DE I-131) permitted for continued full power operation.	Conforms The analysis utilized the maximum equilibrium concentration value (0.2 μCi/gm DE I-131) permitted for continued full power operation.
3	The activity released from the fuel should be assumed to mix instantaneously and homogeneously in the reactor coolant. Noble gases should be assumed to enter the steam phase instantaneously.	Conforms The activity released from the fuel was assumed to mix instantaneously and homogeneously in the reactor coolant. Noble gases were assumed to enter the steam phase instantaneously.

4	TRANSPORT Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows.	Conforms An analysis was performed utilizing the transport guidance of Appendix D of RG 1.183 at hot standby, assuming no melt.
4.1	The main steam line isolation valves (MSIV) should be assumed to close in the maximum time allowed by Technical Specifications.	Conforms A MSIV closure time of 5.5 seconds (0.5 seconds for the valve to initiate the closure sequence and maximum of 5 seconds for full closure per TS 3.6.1.3)
4.2	The total mass of coolant released should be assumed to be that amount in the steam line and connecting lines at the time of the break plus the amount that passes through the valves prior to closure.	Conforms The total mass of coolant released was assumed to be that amount in the steam line and connecting lines at the time of the break plus the amount that passes through the valves prior to closure. The reactor is assumed to be in hot standby before the break to maximize the mass release.
4.3	All the radioactivity in the released coolant should be assumed to be released to the atmosphere instantaneously as a ground-level release. No credit should be assumed for plateout, holdup, or dilution within facility buildings.	Conforms All the radioactivity in the released coolant was assumed to be released to the atmosphere instantaneously as a ground-level release. No credit was assumed for plateout, holdup, or dilution within the Reactor Building steam tunnel.

Attachment 4 to PLA-5963

Regulatory Guide 1.194
Compliance Table

Regulatory Guide 1.194 Compliance

Regulatory Guide 1.194 Sections 3 through 7 and Table A-2 provide methodologies and assumptions that are acceptable to the NRC staff related to atmospheric relative concentrations for Control Room radiological habitability assessments at nuclear power plants. Compliance with Regulatory Guide 1.194 positions are discussed below:

Please Note: The information provided in this table is based on the calculations provided in Attachment 10.

RG 1.194 Section	Regulatory Guide 1.194 Position	Basis of Compliance
3.	CALCULATION OF χ/Q USING ARCON96 This section addresses the use of the ARCON96 code for calculating χ/Q values for design basis Control Room radiological habitability assessments. The ARCON96 code should be obtained and maintained under an appropriate software quality assurance program that complies with the applicable criteria of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 and applicable industry consensus standards to which the licensee has committed.	Conforms The ARCON96 code is maintained under a software quality assurance program that complies with Appendix B to 10 CFR Part 50 and applicable industry consensus standards.
3.1	Meteorological Data Input The meteorological data needed for χ/Q calculations include wind speed, wind direction, and a measure of atmospheric stability. These data should be obtained from an onsite meteorological measurement program based on the guidance of Safety Guide 23, "Onsite Meteorological Programs" (Ref. 12), that includes quality assurance provisions consistent with Appendix B to 10 CFR Part 50. The meteorological data set used in these assessments should represent hourly averages as defined in Safety Guide 23. Data should be representative of the overall site conditions and be free from local effects such as building and cooling tower wakes, brush and vegetation, or terrain. Collected data should be reviewed to identify instrumentation problems and missing or anomalous observations (see Ref. 13). The size of the data set used in the χ/Q assessments should be sufficiently	Conforms The meteorological data includes wind speed, wind direction, and a measure of atmospheric stability. These data were obtained from an onsite meteorological measurement program based on the guidance of Safety Guide 23, that includes quality assurance provisions consistent with Appendix B to 10 CFR Part 50.

	large such that it is representative of long-term meteorological trends at the site. The NRC staff considers 5 years of hourly observations to be representative of long-term trends at most sites. With sufficient justification of its representativeness, however, the minimum meteorological data set is one complete year (including all four seasons) of hourly observations. Wind direction should be expressed as the direction from which the wind is blowing (i.e., the upwind direction from the center of the site) referenced from true north. Atmospheric stability should be determined by the vertical temperature difference (ΔT) measured over the difference in height appropriate for the projected release height (including plume rise as applicable). A table of ΔT values in units of degrees Centigrade per 100 meters (°C/100m) versus stability class is given in Safety Guide 23 (Ref. 12). If other well-documented methodologies are used to estimate atmospheric stability (with appropriate justification), the models described in this guide may require modification. A well-documented methodology is one that is substantiated by diffusion data for conditions similar to those at the nuclear power plant site involved. Appendix A provides information on the structure and content of the meteorological data set and input parameters used by the ARCON96 code.	The meteorological data consists of five years of hourly data, covering the years from 1999 to 2003. Each record of the hourly data contains a location identifier, Julian day (1-366), hour (0 to 23), low-level direction, low-level speed, stability class (1=A to 7=G), upper level direction, and upper level speed. Wind speeds are entered in tenths of a reporting unit with no decimal. Wind directions are from 1 to 360 in degrees. Atmospheric stability was determined by the vertical temperature difference (ΔT) measured over the difference in height appropriate for the projected release height per Safety Guide 23.
3.2	Determination of Release Point (Source) Characteristics A 95th-percentile χ/Q value should be determined for each identified source-receptor combination. However, it may be possible to identify bounding combinations in order to reduce the needed calculational effort. In determining the bounding combinations, it will be necessary to consider the distance, direction, release mode, and height of the various release points to the environment in relation to the various Control Room intakes. Additional parameters, such as those used in establishing plume rise, may need to be considered in determining the bounding combination. For cases involving two or more release pathways associated with a single release source, a calculated composite value of χ/Q may be considered on a case-by-case basis if the licensee can demonstrate an acceptable modeling approach and justify the conservatism of any assumed weighting factors.	Conforms A total of 13 potential release points were evaluated, based on distance, direction, release mode, and height of the various release points to the environment in relation to the Control Room intake. It was determined that 5 of the 13 potential release points required the calculation of ARCON96 χ/Qs. There were no cases involving two or more release pathways associated with a single release source.

	Changes in associated parameters that could occur as a result of differences between normal operation and accident conditions, differences between accidents, differences that occur over the duration of the accident, single failure considerations, and considerations of loss of offsite power, consistent with accident sequences and descriptions, must all be considered in the characterization of the release points. The ARCON96 code provides options that allow an analyst to model ground-level, elevated stack, and vent-point source releases. In addition, the analyst can model diffuse area sources as a sub mode of the ground-level release type. These modes and limitations on their use are discussed in the positions that follow.	
3.2.1	Ground-Level Releases The ground-level release mode is appropriate for the majority of Control Room χ/Q assessments. If the release type is ground level, ARCON96 ignores all user inputs related to release velocity and radius. Release height is used to establish the plume slant path.	Conforms All release points are assumed to be ground level releases.
3.2.2	Elevated (Stack) Releases The stack release mode is appropriate for releases from a freestanding, vertical, uncapped stack that is outside the directionally dependent zone of influence of adjacent structures. Such a stack should be more than 2-1/2 times the height of the adjacent structures or be located: • more than 5L downwind of the trailing edge of upwind buildings, and • more than 2L upwind of the leading edge of downwind buildings, and • more than 0.5L crosswind of the closest edge of crosswind buildings Where L is the lesser of the height or width of the building creating the downwind, upwind, or crosswind wake. Since L will be dependent on wind direction for most building clusters, it will generally be necessary to assess the zone of influence for all directions within the 90° wind direction sector centered on the line of sight between the stack and the Control Room intake. If multiple intakes are involved such that upwind, downwind, and crosswind	N/A. See Section 3.2.1.

	orientations are confounded, 5L could be used for each orientation. Plume rise from buoyancy or mechanical jet effects are not calculated by ARCON96. The analyst may determine plume rise and add the amount of rise to the physical height of the stack to obtain an effective plume height as described in Regulatory Position 6 of this guide (Note: The plume rise may not be added to the physical height of the stack for the purpose of meeting the 2-1/2 times height criterion). Although ARCON96 does not determine plume rise, the input values of stack flow, radius, and vertical velocity are used by ARCON96 to assess downwash and to estimate a limiting χ/Q value. If the Control Room intake is located close to the base of a tall stack, the elevated release model in ARCON96 generates negligibly low χ/Q values. Although perhaps numerically correct, these model results may not be sufficiently conservative for a design basis assessment since the model does not adequately address meteorological conditions that could result in higher χ/Q values. Although the staff has previously suggested that licensees model fumigation as a mechanism to address this situation, the fumigation model did not appear to adequately estimate the effluent concentrations at the bases of industrial stacks. Concentrations greater than those predicted by ARCON96 could result from diurnal wind direction changes, meander, or stagnation. Therefore, the following procedure should be used to assess whether a particular stack-intake configuration is subject to this concern and to determine the appropriate χ/Q values. In addition to running ARCON96 to determine the elevated stack χ/Q values for the Control Room assessment, the analyst should calculate the maximum elevated stack χ/Q value (non-fumigation) using the methodology of Regulatory Guide 1.145 (Ref. 9) to determine the maximum χ/Q value at ground level for the 0-2 hour interval and for the 24-96 and 96-720 hour intervals. The NRC-sponsored code, PAVAN (Ref. 14), is acceptable to the staff for this assessment. For this assessment, the input parameters should be adjusted such that the effective release height is measured from the elevation of the Control Room outside air intake rather than plant grade. The same release point characterization and meteorological data sets used in ARCON96 should be used to determine the χ/Q values for several distances in each wind direction sector with the objective of identifying the maximum χ/Q value. Figure A.4 of Reference 15 may be useful in this regard. The maximum χ/Q value obtained for the 0-2 hour interval should be compared to the corresponding χ/Q value generated by ARCON96 and the higher value used in habitability assessments. The χ/Q values generated by ARCON96 for the 2-8 and the 8-24 hour intervals may be used without adjustment.	

	For the 24-96 hour and 96-720 hour intervals, the following expressions may be used to determine the effective χ/Q. This deterministic approach assumes that the stack plume reverses direction for 1 hour of each day for the duration of the event. The plume is assumed to fold over itself such that the ground level concentration is at its maximum value at the Control Room intake. $$\left(\frac{X}{Q}\right)_{24\to 96\ \mathrm{HRS}} = \frac{1*\left(\frac{X}{Q}\right)^{\mathrm{PAVAN}}_{24\to 96\ \mathrm{HRS}} + 23*\left(\frac{X}{Q}\right)^{\mathrm{ARCON96}}_{24\to 96\ \mathrm{HRS}}}{24} \qquad (1)$$ $$\left(\frac{X}{Q}\right)_{96\to 720\ hrs} = \frac{1*\left(\frac{X}{Q}\right)^{\mathrm{PAVAN}}_{96\to 720\ hrs} + 23*\left(\frac{X}{Q}\right)^{\mathrm{ARCON96}}_{96\to 720\ hrs}}{24} \qquad (2)$$	
3.2.3	Vent Releases The ARCON96 calculation of vent releases includes an algorithm to model mixed-mode releases as described in Regulatory Guide 1.111 (Ref. 10), which addresses χ/Q values used in the assessment of routine effluent releases. The development of this algorithm was based in part on limited field experiments. Given the limited experiment set, the results obtained with this algorithm may not be sufficiently conservative for accident evaluations. For this reason, the vent release mode should not be used in design basis assessments. This position is consistent with the guidance of Regulatory Guide 1.145 (Ref. 9) for offsite χ/Q values. These releases should be treated as a ground level release (Section 3.2.1) or as an elevated release (Section 3.2.2).	N/A. See Section 3.2.1.

3.2.4	Diffuse Area Sources The diffusion models in ARCON96 are based on point-source formulations. However, some release sources may be better characterized as area sources. Examples of possible area sources are postulated releases from the surface of a reactor or a secondary containment building. Typical assessments for loss-of-coolant accidents (LOCAs) have conservatively assumed that the containment structure could leak anywhere on the exposed surface. As such, these assessments typically used the shortest distance between the building surface and the Control Room intake and have treated the building as a point source. This approach may be unnecessarily conservative. A more reasonable approach, while still maintaining adequate conservatism, would be to model the building surface as a vertical planar area source. This approach is not intended to address dispersion resulting from building-induced turbulence. Treatment of a release as a diffuse source will be acceptable for design basis calculations if the guidance herein is followed. The staff may consider deviations from this guidance on a case-by-case basis.	N/A. The diffusion models are based on point-source formulations.
3.2.4.1	Diffuse source modeling should be used only for those situations in which the activity being released is homogeneously distributed throughout the building and when the assumed release rate from the building surface would be reasonably constant over the surface of the building. For example, steam releases within a Turbine Building with roof ventilators or louvered walls would generally not be suitable for modeling as a diffuse source. (See Regulatory Positions 3.2.4.7 and 3.2.4.8.).	N/A. The diffusion models are based on point-source formulations.
3.2.4.2	Since leakage is more likely to occur at a penetration, analysts must consider the potential impact of building penetrations exposed to the environment within this modeled area. If the penetration release would be more limiting, the diffuse area source model should not be used. Releases from personnel air locks and equipment hatches exposed to the environment, or containment purge releases prior to containment isolation, may need to be treated differently. It may be necessary to consider several cases to ensure that the χ/Q value for the most limiting location is identified.	N/A. The diffusion models are based on point-source formulations.

	Note: Penetrations that are enclosed within safety-related structures need not be considered in this evaluation if the release would be captured and released via a plant ventilation system, as ventilation system releases should have already been addressed as a separate release point.	
3.2.4.3	The total release rate (e.g., $Ci \cdot s^{-1}$) from the building atmosphere is to be used in conjunction with the diffuse area source χ/Q in assessments. This release rate is assumed to be equally distributed over the entire diffuse source area from which the radioactivity release can enter the environment. For freestanding containments, this would be the entire periphery above grade or above a building that surrounds the lower elevations of the containment. When a licensee can justify assuming collection of a portion of the release from the containment within the surrounding building, the total release from the containment may be apportioned between the exposed and enclosed building surfaces. Similarly, if the building atmosphere release is modeled through more than one simultaneous pathway (e.g., drywell leakage and main steam safety valve leakage in a BWR), only that portion of the total release released through the building surface should be used with the diffuse area χ/Q. The release rate should not be averaged or otherwise apportioned over the surface area of the building. For example, reducing the release rate by 50 percent because only 50 percent of the surface faces the Control Room intake would be inappropriate.	N/A. The diffusion models are based on point-source formulations.
3.2.4.4	ARCON96 uses two initial diffusion coefficients entered by the user to represent the area source. There are insufficient field measurements to mechanistically model these initial diffusion coefficients. The following deterministic equations should be used in the absence of site-specific empirical data. Note: See Regulatory Position 7 regarding the use of site-specific empirical measurements. $\sigma_{Y_0} = \frac{\text{Width}_{\text{area source}}}{6} \qquad (3)$ $\sigma_{Z_0} = \frac{\text{Height}_{\text{area source}}}{6} \qquad (4)$	N/A. The diffusion models are based on point-source formulations.

3.2.4.5	The height and width of the area source (e.g., the building surface) are taken as the maximum vertical and horizontal dimensions of the above-grade building cross-sectional area perpendicular to the line of sight from the building center to the Control Room intake. These dimensions are projected onto a vertical plane perpendicular to the line of sight and located at the closest point on the building surface to the Control Room intake. The release height is set at the vertical center of the projected plane. The source-to-receptor distance (slant path) is measured from this point to the Control Room intake.	N/A. The diffusion models are based on point-source formulations.
3.2.4.6	Intentional releases from a secondary containment (e.g., standby gas treatment systems (SGTS) at BWR reactors) or annulus ventilation systems in dual containment structures should be treated as a ground-level release or an elevated stack release, as appropriate. The diffuse area source model may be appropriate for time intervals for which the secondary containment or annulus ventilation system is not capable of maintaining the requisite negative pressure differential specified in Technical Specifications or in the FSAR. Secondary containment bypass leakage (i.e., leakage from the primary containment that bypasses the secondary containment and is not collected by the SGTS) should be treated as a ground-level release or an elevated stack release, as appropriate.	N/A. The diffusion models are based on point-source formulations.
3.2.4.7	A second possible application of the diffuse area source model is determining a χ/Q value for multiple (i.e., 3 or more) roof vents. This treatment would be appropriate for configurations in which (1) the vents are in a close arrangement, (2) no individual vent is significantly closer to the Control Room intake than the center of the area source, (3) the release rate from each vent is approximately the same, and (4) no credit is taken for plume rise. The distance to the receptor is measured from the closest point on the perimeter of the assumed area source. For assumed areas that are not circular, the area width is measured perpendicular to the line of sight from the center of the assumed source to the Control Room intake. The initial diffusion coefficient σ_{Y_o} is found by Equation 3; σ_{Z_o} is assumed to be 0.0. Note: The degree of significance will depend on the radius or width of the assumed area and the proximity of the vent cluster to the Control Room intake. As the radius decreases or the distance from the cluster to the Control Room intake increases, the less significance the position of any one vent has.	N/A. The diffusion models are based on point-source formulations.

3.2.4.8	A third possible application of the diffuse area source model is determining a χ/Q value for large louvered panels or large openings (e.g., railway doors on BWR Mark I plants) on vertical walls. This treatment would be appropriate for a louvered panel or opening when (1) the release rate from the building interior is essentially equally dispersed over the entire surface of the panel or opening and (2) assumptions of mixing, dilution, and transport within the building necessary to meet condition 1 are supported by the interior building arrangement. The staff has traditionally not allowed credit for mixing and holdup in Turbine Buildings because of the buoyant nature of steam releases and the typical presence of high volume roof exhaust ventilators. The distance to the receptor and the release height is measured from the center of the louvered panel or opening. Initial diffusion coefficients are found using Equations 3 and 4 assuming the width and height is that of the panel or opening rather than that of the building. If the area source and the intake are on the same building surface such that wind flows along the building surface would transport the release to the intake, the initial dispersion coefficient will need to be adjusted. If the included angle between the source-receptor line of sight and the vertical axis of the assumed source is less than 45 degrees, σ_{Y_o} should be set to 0.0. If the included angle between the source-receptor line of sight and the horizontal axis of the assumed source is less than 45 degrees, σ_{Z_o} should be set to 0.0.	N/A. The diffusion models are based on point-source formulations.
3.3	Determination of Control Room Intakes (Receptors) This section of the guide provides guidance to the meteorological analyst in applying models for determining χ/Q values that are appropriate for the as-built configuration of Control Room intakes. Radioactive materials released during an accident can enter the Control Room envelope via several potential pathways. These pathways may be intentional (e.g., ventilation system outside air intakes) and unintentional infiltration paths (e.g., doorways, envelope penetrations, leakage in ventilation system components). The applicable pathways will vary from site to site depending on the arrangement of the Control Room envelope in relation to other site buildings, the pressure differentials between these buildings and the Control Room, the configuration of Control Room ventilation systems, and the classification of the Control Room dose control (e.g., zone isolation with filtered pressurization, zone isolation with no pressurization). It may be necessary to determine χ/Q values for each potential pathway. However, the selection of one or more bounding intakes for the χ/Q evaluation may be sufficient to establish compliance with regulatory guidelines.	Conforms The Control Room envelope potential pathways were evaluated. It was determined that these pathways include ventilation system outside air intakes and unintentional infiltration paths. A single, bounding χ/Q value was determined for the two potential pathways.

3.3.1	Ventilation System Outside Air Intakes All Control Room ventilation systems draw makeup air from the environment during normal operations and many draw air from the environment for the purpose of supplying filtered pressurization air. The configuration of these systems may change between normal and emergency modes. In some configurations, normal ventilation outside air intakes isolate and different intakes open to supply pressurization air. Some intake dampers may have failure modes related to loss of ac power or single failures. These considerations should be evaluated in identifying the Control Room outside air intakes for which χ/Q values should be calculated.	Conforms Control Room ventilation system configuration for normal and emergency modes was considered when identifying the Control Room outside air intakes for which χ/Q values should be calculated.
3.3.2	Dual Ventilation Outside Air Intakes This section applies to Control Room ventilation system configurations that have two outside air intakes, each of which meets applicable design criteria of an engineered safeguards feature (ESF), including single-failure criterion, missile protection, seismic criteria, and operability under loss-of-offsite AC power conditions. Operability requirements should be provided in Technical Specifications. The outside air intakes should be located with the intent of providing a low contamination intake regardless of wind direction. The assurance of a low contamination outside air intake depends on release point configuration, building wake effects, terrain, and the possibility of wind stagnation or wind direction reversals. The two intakes should not be within the same wind direction window, defined as a wedge centered on the line of sight between the source and the receptor with the vertex located on the release point. If ARCON96 is used, the wedge angle is 90° (i.e., 45 degrees on either side of the line of sight). If the methods of Regulatory Position 4 are used, the size of the wedge is as given in Table 2. Figure 3 illustrates four examples of the interplay between Control Room intakes, release points, and wind direction windows. In addition, the analyst should consider χ/Q values for infiltration pathways as discussed in Regulatory Position 3.3.3. The methods of this regulatory position involve identification of the limiting and favorable intakes with regard to their χ/Q value. Because of the interplay of building wake, plume rise, wind direction frequency, intake flow rate, and other parameters, it may not be possible to identify the limiting or favorable intake by observation. In these situations, χ/Q	N/A. The Control Room ventilation system has a single outside air intake.

3.3.2.1	If both of the dual intakes are located within the same wind direction window, both intakes could be contaminated (See Figure 3(a)). In this case, the χ/Q values for each air intake should be calculated using ARCON96 as described in other sections of this guide and an effective χ/Q value calculated. Equation 5a should be used if the intake flow rates are equal. If the intake flow rates are not equal, but the imbalance does not shift between intakes, Equation 5b should be used. If the flow rate imbalance can shift between intakes, Equation 5c should be used. This calculation is repeated for each averaging time interval. $\overline{X/Q} = 0.5[(X/Q)_1 + (X/Q)_2]$ (5a) $\overline{X/Q} = \frac{F_1(X/Q)_1 + F_2(X/Q)_2}{F_1 + F_2}$ (5b) $\overline{X/Q} = \frac{\max(F_1,F_2) * \max[(X/Q)_1,(X/Q)_2] + \min(F_1,F_2) * \min[(X/Q)_1,(X/Q)_2]}{F_1 + F_2}$ (5c) Where: $\overline{X/Q}$ = Effective X/Q, s m^{-3} $(X/Q_1, (X/Q)_2$ = X/Q value for outside air intakes 1 and 2, s m^{-3} F_1, F_2 = Flow rate for outside air intakes 1 and 2, cfm	N/A. The Control Room ventilation system has a single outside air intake.
3.3.2.2	If the dual outside air intakes are not in the same wind direction window but cannot be isolated by design, the χ/Q values for the limiting outside air intake should be calculated for each time interval as described elsewhere in this guide. Equation 6a should be used if the intake flow rates are equal. If the intake flow rates are not equal, but the imbalance does not shift between intakes, Equation 6b should be used. If the flow rate imbalance can shift between intakes, Equation 6c should be used.	N/A. The Control Room ventilation system has a single outside air intake.

	$\overline{X/Q} = 0.5 \max [(X/Q)_1, (X/Q)_2]$ (6a) $\overline{X/Q} = \frac{\max[(F_1 (X/Q)_1, F_2 (X/Q)_2]}{F_1 + F_2}$ (6b) $\overline{X/Q} = \frac{\max (F_1, F_2) \max [(X/Q)_1, (X/Q)_2]}{F_1 + F_2}$ (6c)	
3.3.2.3	If the ventilation system design allows the operator to manually select the least contaminated outside air intake as a source of outside air makeup and close the other intake, the χ/Q values for each of the outside air intakes should be calculated for each time interval as described elsewhere in this guide. The χ/Q value for the limiting intake should be used for the time interval prior to intake isolation. This χ/Q value may be reduced by a factor of 2 to account for dilution by the flow from the other intake (see Equation 6a). The χ/Q values for the favorable intake are used for the subsequent time intervals. The χ/Q values for the favorable intake may be reduced by a factor of 4 to account for the dual inlet and the expectation that the operator will make the proper intake selection. This protocol should be used only if the dual intakes are in different wind direction windows and if there are redundant, ESF-grade radiation monitors within each intake, with Control Room indication and alarm, to monitor the intakes. The requisite steps to select the least contaminated outside air intake, and provisions for monitoring to ensure the least contaminated intake is in use throughout the event, should be addressed in procedures and in operator training. A conservative delay time should be assumed for the operator to complete the necessary actions. This delay period should consider: (1) the time for the operator to recognize the radiation monitor alarm and determine its validity (as provided for in the alarm response procedure), (2) delays associated with other accident response actions competing for the operator's attention, (3) the time needed to complete the actions, and (4) diesel generator sequencing time, if applicable. If actions are required outside the Control Room, delays associated with transit to the local control stations (including those delays caused by worker radiological protection controls associated with accident dose rates), and the availability of personnel should be considered.	N/A. The Control Room ventilation system has a single outside air intake.

	Note: The adjustment protocol and the numeric factors of this section are deterministic in nature and are expected to be conservative for most sites. Different factors may be considered on a case-by-case basis with sufficient justification.	
3.3.2.4	If the ventilation system design provides for automatic selection of the least contaminated outside air intake, the χ/Q values for the favorable intake should be calculated for each time interval as described elsewhere in this guide. The χ/Q values may be reduced by a factor of 10 to account for the ability to automatically select a "clean" intake. This protocol should be used only if the dual intakes are in different wind direction windows, there are redundant ESF-grade radiation monitors within each intake and an ESF-grade control logic and actuation circuitry is provided for the automatic selection of a clean intake throughout the event.	N/A. The Control Room ventilation system has a single outside air intake.
3.3.3	Infiltration Pathways Infiltration of contaminated air to a Control Room can be minimized by proper design and maintenance of the Control Room envelope (CRE). However, infiltration is always a possibility and the location and significance of these leakage pathways may warrant determination of χ/Q values. An unfiltered inleakage path of 100 cfm can admit the same quantity of radioactive material as a pressurization air intake having a flow of 2000 cfm through a 95 percent efficient filter. The situation can be further compounded if the χ/Q for the unfiltered pathway is more limiting than that for the Control Room outside air intake. The infiltration paths actually applicable to a particular facility will be identified via inleakage testing or CRE inspections and surveillances. Refer to Table H-1, "Determination of Vulnerability Susceptibility," of NEI 99-03, "Control Room Habitability Guidance" (Ref. 16), for further guidance on infiltration pathways. A 95th-percentile χ/Q value should be determined for each time interval for any infiltration path that could result in a significant intake of contaminated air into the CRE. Because of the interplay of source-to-receptor distance and direction, infiltration path flow rate, whether the path is filtered or unfiltered, and other considerations, it may not be possible to	Conforms The χ/Q determined for infiltration pathways was evaluated and it was determined that the use of the χ/Q calculated for the Control Room habitability envelope outside air intake was acceptable. A more complete discussion is provided in Attachment 2, Section 4.1.

	identify the potential impact of an infiltration path by observation. In these situations, χ/Q values should be calculated for each pathway and the limiting χ/Q value(s) identified. If there is sufficient margin available, it may be possible to calculate χ/Q values assuming the shortest distance between the release point and any identified point of infiltration on the outside of the CRE.	
3.4	Determination of Source-Receptor Distances and Directions When the combinations of release points and intakes have been identified, the direction and distance between the release point and the intake should be determined. Wind direction data are recorded as the direction from which the wind blows (e.g., a north wind blows from the north; a wind blowing out of the west is recorded with a direction of 270 degrees). The direction input to ARCON96 is the wind direction that would carry the plume from the release point to the intake. For example, an analyst standing at the intake facing west to the release point, would enter 270 degrees; an analyst facing north, would enter 360 degrees, etc. The source-to-receptor distance is the shortest horizontal distance between the release point and the intake. ARCON96 will use this distance and the elevations of the source and receptor to calculate the slant path. For an area source such as building surface, the shortest horizontal distance from the building surface to the Control Room intake is used as the source-to-receptor distance. For releases within building complexes, the shortest horizontal distance between the release point and the intake could be through intervening buildings. In these cases, it is acceptable to take the length of the shortest path (e.g., "taut string length") around or over the intervening building as the source-to-receptor distance. If the distance to the receptor is less than about 10 meters, the ARCON96 code and the procedures in Regulatory Position 4 should not be used to assess χ/Q values. These situations will need to be addressed on a case-by-case basis. Note: The site meteorological tower wind direction sensors are generally calibrated with reference to true north (360 degrees). Analysts should use caution in measuring directions on site engineering drawings since these drawings typically incorporate a plant grid and a plant "north" that may not align with true north. The source-to-receptor directions input to ARCON96 must use the same north reference as the wind direction observations.	Conforms Appropriate wind directions and source-receptor distances were input into ARCON96 for determination of the χ/Qs for each of the accidents analyzed. No taut string distances were used in the χ/Q determination. All distances to the new CRHE air intake are > 10m.

4.0	ALTERNATIVE PROCEDURES FOR GROUND-LEVEL RELEASES This regulatory position addresses alternative methods for determining χ/Q values for Control Room radiological habitability assessments. The methods in Regulatory Positions 4.1 to 4.3 are based on Murphy-Campe (Ref. 2) and the Standard Review Plan Chapter 6.4 (Ref. 3).	N/A. All ground level releases were determined per the preceding methodology.
4.1	Point Source-Point Receptor The 0-8 hour 95th-percentile (Note: The Murphy-Campe document identified this as the 5th-percentile /Q value.) χ/Q value for a single point source on the surface of the containment or other building and a single point receptor with a difference in elevation less than 30 percent of the building height may be estimated using Equation 7. $\frac{X}{Q} = \frac{1}{3\pi U \sigma_y \sigma_z}$ (7) Where: χ/Q = Relative concentration at plume centerline for time interval 0-8 hours, s m^{-3} 3 = Wake factor U = Wind speed at 10 meters, m s^{-1} σ_y, σ_z = Standard deviation, in meters, of the gas concentration in the horizontal and vertical cross wind directions evaluated at distance x and by stability class	N/A. All ground level releases were determined per the preceding ARCON96 methodology.
4.2	Diffuse Source-Point Receptor Equation 8 may be used when the activity is assumed to leak from many points on the surface of a building such as the containment in conjunction with a single point receptor. This equation is also appropriate for point source-point receptors where the difference in elevation between the source and the receptor is greater than 30 percent of the height of the	N/A. All ground level releases were determined per the preceding ARCON96 methodology and all sources were considered point sources.

	upwind building, typically the containment, which creates the most significant building wake impact. The equation is also applicable to a point source and volume receptor (e.g., an isolated Control Room with infiltration occurring at many locations). $$\frac{X}{Q} = \left[U \left(\pi \sigma_y \sigma_z + \frac{A}{K+2} \right) \right]^{-1} \quad (8)$$ Where: χ/Q =Relative concentration at plume centerline for time interval 0-8 hours, s m^{-3} U = Wind speed at 10 meters, m s^{-1} σ_y, σ_z = Standard deviation, in meters, of the gas concentration in the horizontal and vertical cross wind directions evaluated at distance x and by stability class $K = \frac{3}{(s/d)^{1.4}}$ s = Shortest distance between building surface and receptor location, m d = Diameter or width of building, m A = Cross-section area of building, m_2 The reference to "building" in the definitions of s, d, and A is to the diffuse source (e.g., containment). If the equation is used with a point source, the reference is to the building that has the greatest impact on the building wake. The values of the parameters σ_y, σ_z and U should be determined on the basis of the values of the site meteorological data. Some early analyses may have been based on generic meteorology conditions (e.g., F stability with wind speeds of 1.0 m·s^{-1}). If these early analyses are to be updated, the staff recommends that the ARCON96 code be used. If the ARCON96 code is not used, site-	

	specific hourly meteorological data should be used to determine the 95th-percentile χ/Q value. Figures 4 and 5 provide sigma values by stability category for distances greater than 10 meters. The data on these graphs should not be extrapolated for distances less than 10 meters.	
4.3	Point or Diffuse Source with Two Alternative Receptors Equations 7 and 8 of this guide may be used in conjunction with the procedures in Regulatory Position 3.3.2 to determine χ/Q values for Control Room designs having two or more Control Room outside air intakes, each of which meets the requirements of an engineered safety feature (ESF) including, as applicable, single-failure criteria for active components, seismic criteria, and missile criteria. If Equation 8 of this guide is used, the parameter K should be set to 0.0. In a change from previous practice, the staff no longer finds Equation 7 of Reference 2 to be acceptable for use in new applications.	N/A. All ground level releases were determined per the preceding ARCON96 methodology and there is only one receptor.
4.4	Determination of χ/Q Values for Other Time Intervals Equations 7 and 8 are used to determine χ/Q values for the first time interval of 0-8 hours. The χ/Q values for other time intervals are obtained by adjusting for long-term meteorological averaging of wind speed and wind direction. This is accomplished by multiplying the 0-8 hour time interval χ/Q value by a correction factor for wind speed and a correction factor for wind direction. Note: Previous guidance also provided for including a factor to account for personnel occupancy factors. Since typical radiological analysis codes provide the capability to enter these factors separately, the staff recommends that the factors not be included in the χ/Q value to avoid inadvertent double crediting.	N/A. All ground level releases were determined per the preceding ARCON96 methodology utilizing standard time intervals.
4.4.1	χ/Q Correction for Wind Speed Averaging This correction is defined as the ratio of the wind speed used to determine the 0-8 hour χ/Q value to the wind speed appropriate for each of the other time intervals. Column 2 of Table 1 tabulates the wind speed percentiles that correspond to each of these intervals. The hourly data should be arranged in order of increasing wind speed and the wind speed percentiles determined (i.e., the lowest wind speeds associated with the lowest percentiles).	N/A. All ground level releases were determined per the preceding ARCON96 methodology utilizing standard time intervals; consequently, no χ/Q correction is required per wind speed averaging.

	Include only the wind speed data associated with wind directions from sectors that result in receptor contamination. Table 2 tabulates the size of the minimum wind direction window to be used. From this ranking, identify the wind speed value for each interval that is not exceeded more than the stated percentage of the time. Divide this wind speed value into the 5th-percentile wind speed used to determine the 0-8 hour χ/Q to obtain the χ/Q correction factor for wind speed. The values shown in Column 1 of Table 1 are representative correction factors that may be used if hourly observation meteorological data are not available.	

Table 1

χ/Q Correction for Wind Speed Averaging

Time Interval	Column 1 Representative χ/Q Factors	Column 2 Corresponding Wind Speed Percentile
0-8 hours	1.0	5
8-24 hours	0.67	10
1-4 days	0.50	20
4-30 days	0.33	40

Table 2

Wind Direction Sectors

s/d Ratio	Minimum Window (Note: Centered on the source-to-receptor direction.)
>2.5	68°
1.25 – 2.5	90°
0.8 – 1.25	113°
0.6 – 0.8	135°
0.5 – 0.6	158°
0.35 – 0.5	180°
<0.35	225°

	The s/d is defined as: $$\frac{s}{d} = \frac{\text{Shortest distance between building surface and receptor location, m}}{\text{Diameter or Width of building, m}}$$ The reference to "building" in Equation 9 is to the diffuse source (e.g., containment). If the equation is used with a point source, the reference is to the building that has the greatest impact on the building wake.	
4.4.2	χ/Q Correction for Wind Direction Averaging The average wind direction frequency F is obtained by summing the annual average wind direction frequencies within the minimum window. Table 2 tabulates the size of the minimum wind direction window to be used. Column 2 of Table 3 is used to determine the χ/Q correction factor for wind direction for each time interval. Column 1 is used when F has not been determined. Table 3 Wind Direction Averaging Correction Time Interval / Column 1 Representative χ/Q Factors / Column 2 Equations for χ/Q Factors 0-8 hours / 1.0 / 1.0 8-24 hours / 0.88 / 0.75 + F/4 1-4 days / 0.75 / 0.50 + F/2 4-30 days / 0.5 / F	N/A. All ground level releases were determined per the preceding ARCON96 methodology utilizing standard time intervals; consequently, no χ/Q correction is required per wind direction averaging.

Table 3

Wind Direction Averaging Correction

Time Interval	Column 1 Representative χ/Q Factors	Column 2 Equations for χ/Q Factors
0-8 hours	1.0	1.0
8-24 hours	0.88	0.75 + F/4
1-4 days	0.75	0.50 + F/2
4-30 days	0.5	F

5.0	INSTANTANEOUS PUFF RELEASES The alternative method in this section may be used to model the release to the environment as an instantaneous puff release. One hundred percent of the radionuclides must be released directly to the environment over a period no longer than about 1 minute for a release to qualify as a puff release. Releases to enclosed buildings, intermittent releases that occur over a period longer than about 1 minute (e.g., releases from relief valves, atmospheric dumps), and releases that occur over a period longer than about 1 minute should be treated as continuous point source releases. The diffusion equation for an instantaneous puff ground level release, with no puff rise and no crosswind offset (i.e., center of puff is assumed to pass over Control Room intake), integrated over the duration of the puff passage is: Where: $$\frac{X}{Q}(x,u,k,h) = \frac{\int_0^T \frac{2}{(\sigma_z^2(x,k)+\sigma_i^2)^{1/2}\,(2\pi)^{3/2}\,(\sigma_{X,Y}^2(x,k)+\sigma_i^2)}}{\int_0^T f(t)dt} *$$ $$\exp\left[-\frac{1}{2}\left(\frac{(x-u*t)^2}{(\sigma_{x,y}^2\,(x,k)+\sigma_i^2)}+\frac{h^2}{(\sigma_z^2\,(x,k)+\sigma_i^2)}\right)\right]F(t)dt$$	Conforms The instantaneous puff release from the RB Unit 2 blowout panel was calculated for the main steam line break accident analysis per *the methodology to the left.* One hundred percent of the radionuclides were released directly to the environment over a period less than 1 minute. The RADTRAD model assumed a RB main steam tunnel volume of 5.611E+4 cf and an exit flow rate of 3.13E+6 cfm to the environment. The *effective* relative concentration calculated for the puff was input to RADTRAD. A detailed calculation is provided in Attachment 10, Calculation EC-RADN-1128, Sections 4.4 through 4.6.

	$\frac{X}{Q}(x, u, x, h)$ = Effective puff relative concentration, s m^{-3} X = Integrated concentration at control intake, Ci $m^{-3}s^{-1}$ Q_i = Relative quantity, for nuclide i, Ci x = Release point to receptor distance, m u = Wind speed, m/sec. Assume 1.0 m s^{-1} k = Stability Class. Assume F. h = Difference in elevation between the physical release point and the control room intake, m. If the control room intake is at a higher elevation than the release point and the puff is bouyant, assume h = 0. T = Time for trailing edge of puff to pass control room intake, sec. $= \frac{x + 3[\sigma_{x,y}(x, k) + \sigma_i]}{u}$ F = Control room total intake flow rate, cfm. (If the control room intake flow rate is constant over the period 0 to T seconds, the F(t) terms can be omitted from Equation 10. $\sigma_{x,y}(x, k)$ = Standard deviation, m, of the puff in the horizontal along the wind direction and cross – wind directions at the receptor locations. Use Figure 4 with the distance x and Stability Class K to determine $\sigma_{x,y}$ at the receptor, e.g., $\sigma_{x,y} = \sigma_y$. $\sigma_z(x, k)$ = Standard deviation, m, of the puff in the vertical cross – wind direction at the receptor location. Use Figure 5 with the distance x and Stability Class K to determine σ_z at the receptor.	

	σ_i = Initial standard deviation, m $= \left[\frac{2V}{(2\pi 2^{3/2}} \right]^{1/3}$ V = Initial puff volume (expanded to standard atmospheric conditions), m^3 (The puff dimensions that would exist when the puff is at the control room intake are assumed to exist during the entire puff transit. Equation 10 provides the *effective* relative concentration for the puff. This value can be input to dose assessment codes such as RADTRAD or HABIT as any value of X/Q would be if the intake flows, release duration, and release rates are modeled consistent with the inputs to Equation 10.	
6.0	PLUME RISE An applicant or licensee may propose adjustments to the release height for plume rise that are due to buoyancy or mechanical jet on a case-by-case basis. In order to credit these adjustments, the applicant or licensee must be able to demonstrate that the assumed buoyancy or vertical velocity of the effluent plumes will be maintained throughout the time intervals that plume rise is credited. Such justifications need to consider the availability of AC power, failure modes of dampers and ductwork, time-dependent release stream temperatures and pressures, and 95th-percentile wind speeds and ambient temperatures. (Note: As used here, 95th-percentile wind speed is that wind speed that is not exceeded more than 5 percent of the time. A 95th-percentile ambient temperature is that temperature that is not exceeded more than 5 percent of the time). Plume rise may be considered for freestanding stacks and for vents located on plant buildings. However, plume rise may not be used in demonstrating that a particular stack meets the 2-1/2 times the adjacent structure height criterion in Regulatory Position 3.2.2. A mixed-mode release model, such as that in Regulatory Guide 1.111 (Ref. 10), should not be used for design basis assessments. The plume rise may be determined through the use of the following set of equations (Ref. 17). The plume rise for plant vents is determined using Equation 11. The distance x is entered as the horizontal distance between the vent and the Control Room outside air intake.	N/A. Plume rise was not considered in the χ/Q determinations.

	The plume rise for isolated, free-standing stacks is calculated using Equations 11, 12, and 13. The distance x in Equation 11 should be based on the downwind location corresponding to the maximum X/Q value. See Regulatory Position 3.2.2. The plume rises calculated using Equations 12 and 13 should be compared and the larger plume rise identified. The result of this comparison is then compared to the plume rise determined using Equation 11 and the smaller plume rise selected for use. $$\Delta h = \left[\frac{3}{\beta_1^2} * \frac{F_m}{U^2} * x + \frac{3}{2\beta_1^2} * \frac{F_b}{U^3} * x^2 \right]^{1/3} \qquad (11)$$ $$\Delta h = 2.6 \left(\frac{F_b}{U_s} \right)^{1/3} \qquad (12)$$ $$\Delta h = 2.44 \left(\frac{F_m}{S} \right)^{1/4} \qquad (13)$$	

	Where: Δh = Plume rise, m F_m = Momentum flux parameter, $m^4\ s^{-2}$ $= \frac{\rho_0 V_0 w_0}{\pi \rho_a}$ β_1 = Dimensionless entrainment constant for momentum = 0.6 U = Wind speed at release height, $m\ s^{-1}$ x = Distance from release point to receptor, m F_b = Bouyancy flux parameter, $m^4\ s^{-3}$ $= \frac{g(\rho_a - \rho_0)V_0}{\pi \rho_a}$ w_0 = Effluent exit velocity, ms^{-1} V_0 = Volumetric release rate, $m^3\ s^{-1}$ ρ_0 = Effluent density after expansion to atmospheric pressure, kg m^{-3} ρ_a = Density of air, kg m^{-3} s = 0.0001 s^{-2} for A, B, C, and D stability; 0.00049 s^{-2} for E stability; 0.0013 s^{-2} for F stability; 0.002 s^{-2} for G stability g = Gravitational acceleration, 9.8 m s^{-2} Although ARCON96 processes ambient meteorological conditions on an hour-by-hour basis, the code cannot vary the other parameters that enter into a plume rise determination. For example, wind speed and stability class are varied hour by hour, but the density of air, the density of the effluent stream, and the vertical velocity are not varied hour-by-hour. As such, the analyst should ensure that these parameters are bounding for the entire period of the X/Q assessment or use individual time intervals to model the time-variant parameters. An alternative approach would be to calculate the plume rise for each hour independently of ARCON96 and to select a plume rise that is exceeded more than 95 percent of the time.	

	This rise is then added to the stack height as input to ARCON96. In lieu of mechanistically addressing the amount of buoyant plume rise associated with energetic releases from steam relief valves or atmospheric dump valves, the ground level X/Q value calculated with ARCON96 (on the basis of the physical height of the release point) may be reduced. (Note: This adjustment factor and the associated velocity ratio criterion are deterministic in nature and their selection was based on sensitivity analyses performed for typical steam release points at LWRs.) The adjustment factor should not be ratioed for different vertical velocity ratios) by a factor of 5. This reduction may be taken only if (1) the release point is uncapped and vertically oriented and (2) the time-dependent vertical velocity exceeds the 95^{th}-percentile wind speed. [Note: As used here, 95^{th}-percentile wind speed is that wind speed that is not exceeded more than 5 percent of the time. A 95^{th}-percentile ambient temperature is that temperature that is not exceeded more than 5 percent of the time (at the release point height) by a factor of 5.]	
7.0	USE OF SITE-SPECIFIC EXPERIMENTAL DATA The methods and parameters provided in this guide are acceptable for use for design basis Control Room habitability radiological assessments provided that all stated prerequisites and conditions are met. The staff believes that use of the guidance in this guide will result in X/Q values that are acceptably conservative. However, there may be circumstances in which these methods and parameters may not be advantageous for a particular plant configuration and site meteorological regimes and may lead to results that are deemed to be unnecessarily conservative. Licensees and applicants may opt to propose alternative methods and parameters such as those that are based in part on data obtained from site-specific experimental measurements. Data based on wind tunnel tests should be accompanied with an evaluation of the representativeness of the experiment results to the particular plant configuration and site meteorological regimes. These proposed alternatives, with supporting data, will be considered by the staff on a case-by-case basis. The staff recommends that licensees considering an experimental program request a meeting with the staff in advance of starting the program. The intent of this recommendation is to allow the staff and the licensee (or applicant) to discuss the proposed program, prior to resource expenditure, and for the staff to provide a preliminary assessment of the proposal. The staff's approval of the proposed alternative methods and	N/A. No experimental data was utilized to calculate the χ/Qs.

	parameters will not be granted, however, until the licensee or applicant completes the experimental program and dockets the proposal with supporting analyses and data for formal staff review. An acceptable experimental program should incorporate the following standards:	
7.1	The experimental program should be appropriately structured so as to provide data of appropriate quantity and quality to support data analysis and conclusions drawn from that data. The program should be developed by personnel who have educational and work experience credentials in air dispersion meteorology and modeling.	N/A. No experimental data was utilized to calculate the χ/Qs.
7.2	The experimental program should encompass a sufficient range of meteorological conditions applicable to the particular site so as to ensure that the data obtained address the site-specific meteorological regimes and the site-specific release point/receptor configurations that impact the Control Room X/Q values. Meteorological conditions observed at the particular site with a frequency of 5 percent or greater in a year should be addressed. Parameters derived from statistical analyses on the experimental data should represent the 95^{th}-percentile confidence level.	N/A. No experimental data was utilized to calculate the χ/Qs.
7.3	The experimental program, including data reduction and analysis, should incorporate applicable quality control criteria of Appendix B to 10 CFR Part 50. The products of the experimental program should be verified and validated.	N/A. No experimental data was utilized to calculate the χ/Qs.

Table A-2
ARCON96 INPUT PARAMETERS FOR DESIGN BASIS ASSESSMENTS

Parameter/Discussion/ Acceptable Input	Basis of Compliance
Lower Measurement Height, meters The value of this parameter is used by ARCON96 to adjust wind speeds for differences between the heights of the instrumentation and the release. Use the actual instrumentation height when known. Otherwise, assume 10 meters.	Conforms The actual instrumentation lower measurement height of 10 meters was utilized.
Upper Measurement Height, meters The value of this parameter is used by ARCON96 to adjust wind speeds for differences between the heights of the instrumentation and the release. Use the actual instrumentation height when known. Otherwise, use the height of the containment or the stack height, as appropriate. If wind speed measurements are available at more than two elevations, the instrumentation at the height closest to the release height should be used.	Conforms The actual instrumentation upper measurement height of 60 meters was utilized.
Wind Speed Units ARCON96 requires that wind speed be entered as miles per hour, m·s-1, or knots. Use the wind speed units that correspond to the units of the wind speeds in the meteorological data file.	Conforms Wind speed was entered as miles per hour, which corresponds to the units of the wind speeds in the meteorological data files.

Release Height, meters The value of the release height is used for three purposes in ARCON96: (1) to adjust wind speeds for differences between the heights of the instrumentation and the release, (2) to determine slant path for ground level releases, (3) to correct off-centerline data for elevated releases. Use the actual release heights whenever available. Plume rise from buoyancy and mechanical jet effects may be considered in establishing the release height if the analyst can demonstrate with reasonable assurance that the vertical velocity of the release will be maintained during the course of the accident. If actual release height is not available, set release height equal to intake height.	Conforms For the five source-receptor locations, the actual release height was utilized or the release height was conservatively assumed to be 0 (assume release height equals intake height). Plume rise from buoyancy and mechanical jet effects was not considered in establishing the release height.
Building Area, meters^2 ARCON96 uses the value of the building area in the high speed wind speed adjustment for ground-level and vent release models. Use the actual building vertical cross-sectional area perpendicular to the wind direction. Use default of 2000 m^2 if the area is not readily available. Do not enter zero. Use 0.01 m^2 if a zero entry is desired. *Note: This building area is for the building(s) that has the largest impact on the building wake within the wind direction window. This is usually, but need not always be, the reactor containment. With regard to the diffuse area source option, the building area entered here may be different from that used to establish the diffuse source.*	Conforms The actual building vertical cross-sectional area perpendicular to the wind direction was conservatively calculated. A value of 2,685 m^2 was utilized.

Vertical Velocity, meters/second In ARCON96, the value of the vertical velocity is used only in vent and stack release models. It is used for the downwash calculation. In the vent release model the velocity is used in the mixed-mode calculation. If the vertical velocity is set to zero, the maximum downwash will be calculated and the release height will be reduced by an amount equal to six times the stack radius. *Note: The vent release model should not be used for DBA accident calculations.* For stack release calculations only, use the actual vertical velocity if the licensee can demonstrate with reasonable assurance that the value will be maintained during the course of the accident (e.g., addressed by technical specifications), otherwise, enter zero. If the vertical velocity is set to zero, ARCON96 will reduce the stack height by 6 times the stack radius for all wind speeds. If this reduction is not desired, the stack radius should also be set to zero.	N/A. Vent and stack release models were not utilized. All releases were ground level.
Stack Flow, meters3/second ARCON96 uses the value of the stack flow in X/Q calculations for all 3 release types to ensure that the near field concentrations are no greater than the concentration at the release point. The impact diminishes with increasing distance. Use actual flow if it can be demonstrated with reasonable assurance that the value will be maintained during the course of the accident (e.g., addressed by Technical Specifications). Otherwise, enter zero. The flow is used in both elevated and ground-level release modes to establish a maximum X/Q value. This value is significant only if the flow is large and the distance from the release point to the receptor is small.	N/A. Vent and stack release models were not utilized. All releases were ground level.

Stack Radius, meters ARCON96 uses the value of the stack radius in downwash calculations in the vent and stack release modes. Use the actual stack internal radius when both the stack radius and vertical velocity are available. If the stack flow is zero, the radius should be set to zero.	N/A. Vent and stack release models were not utilized. All releases were ground level.
Distance to Receptor, meters The value of horizontal distance to the receptor from the release point is used in ARCON96 for calculating the slant range for ground level releases and the off-centerline correction factors for stack release models. Use the actual straight-line horizontal distance between the release point and the Control Room intake. For ground-level releases, it may be appropriate to consider flow around an intervening building if the building is sufficiently tall that it is unrealistic to expect flow from the release point to go over the building. *Note: If the distance to receptor is less than about 10 meters, ARCON96 should not be used to assess relative concentrations.*	Conforms The actual straight-line horizontal distances between the release point and the Control Room are utilized (no taut string distances even for releases in the building complex) to calculate the χ/Qs. All source to receptor distances are greater than 10 meters.
Intake Height, meters The value of the intake height is used in ARCON96 for calculating the slant range for ground level releases and the off-centerline correction factors for stack release models. Use the actual intake height. If the intake height is not available for ground level releases, assume the intake height is equal to the release height. For elevated releases, assume the height of the tallest site building.	Conforms The actual Control Room intake height was utilized to calculate the χ/Qs.

Elevation Difference, meters The value of this parameter is used by ARCON96 to normalize the release heights and the intake heights when the two heights are specified as "above grade" with different grades for the release point and intake height, or when one measurement is referenced to "above grade" and the other to "above sea level." Use zero unless it is known that the release heights are reported relative to different grades or reference data.	Conforms Actual release heights and the intake heights and terrain elevation differences were utilized as appropriate to determine χ/Qs.
Direction to Source, degrees ARCON96 uses the value of this parameter and the Wind Direction Window to establish which range of wind directions should be included in the assessment of the X/Q. Use the direction FROM the intake back TO the release point. (Wind directions are reported as the direction from which the wind is blowing. Thus, if the direction from the intake to the release point is north, a north wind will carry the plume from the release point to the intake.) Note: Some facilities have a "plant north" shown on site arrangement drawings that is different from "true north." The direction entered must have the same point of reference as the wind directions reported in the meteorological data. For ground-level releases, if the plume is assumed to flow around a building rather than over it, the direction may need to be modified to account for the redirected flow. In this case, the X/Q should be calculated assuming flow around and flow over (through) the building and the higher of the two X/Q s should be used.	Conforms Wind direction from the intake back to the release point was utilized to calculate the χ/Qs. Plant north and true north are considered the same at SSES. No analyses considered ground-level releases that flow around a building rather than over it.

Surface Roughness Length, meters ARCON96 uses the value of this parameter in adjusting wind speeds to account for differences in meteorological instrumentation height and release height. Use a value of 0.2 in lieu of the default value of 0.1 for most sites. (Reasonable values range from 0.1 for sites with low surface vegetation to 0.5 for forest-covered sites.)	Conforms A surface roughness length of 0.2 in lieu of the default value of 0.1 was utilized to calculate the χ/Qs.
Wind Direction Window, degrees Code Default ARCON96 uses the value of this parameter and the Direction to Source to establish which range of wind directions should be included in the assessment of the X/Q. Use the default window of 90 degrees (45 degrees on either side of line of sight from the source to the receptor).	Conforms Used the default values.
Minimum Wind Speed, meters/second Code Default ARCON96 uses the value of this parameter to identify calm conditions. Use the default wind speed of 0.5 m·s-1 (regardless of the wind speed units entered earlier), unless there is some indication that the anemometer threshold is greater than 0.6 m·s-1.	Conforms Used the default values.

Averaging Sector Width Constant Code Default ARCON96 uses the value of this parameter to prevent inconsistency between the centerline and sector average X/Q s for wide plumes. Has largest effect on ground level plumes. Although the default value is 4, a value of 4.3 is preferred. (A future revision to ARCON96 will change the default to 4.3)	Conforms An averaging sector width constant of 4.3 (the preferred value) was utilized to calculate the χ/Qs.
Initial Diffusion Coefficients, meters ARCON96 uses these parameters in modeling a diffuse source. These values will normally be set to zero. If the diffuse source option is being used, see Regulatory Position 2.2.4.	Conforms The initial diffusion coefficients are set to zero since only point sources were utilized in the evaluation.
Hours in Averages Code Default The values of this parameter were selected to provide results for desired periods and to provide a smooth X/Q curve. Use the default values.	Conforms Used the default values.
Minimum Number of Hours Code Default The default values of this parameter will allow processing with up to 10% missing data. Use the default values.	Conforms Used the default values.

Attachment 5 to PLA-5963

Safety Assessment for the Proposed Technical Specification And Bases Changes – Units 1 & 2

Proposed Technical Specification and Bases Changes

A description of each proposed TS change and the associated basis/safety assessment are included in Table 5-1. The preceding discussion in Attachment 1, the Safety Assessment in Attachment 2, and the calculations in Attachments 10 and 11 support these changes.

Table 5-1: Proposed Technical Specification and Bases Changes

Description and Safety Assessment for Specific Changes to TS and TS Bases		
Change #1	**Current Technical Specification:** Unit 1 & 2, Section 1.1: Definitions DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors shall be those listed in ICRP-30. Supplement to Part 1, Page 192 – 212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity." Existing calculations using conversion factors listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites" or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, are acceptable.	**Proposed Change:** Unit 1 & 2, Section 1.1: Definitions **DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) which alone would produce the same total effective dose equivalent (sum of committed effective dose equivalent {CEDE} from inhalation plus deep dose equivalent {DDE} or nominally equivalent to the effective dose equivalent {EDE} from external exposure {submersion}) as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The conversion factors that are used for this calculation of committed effective dose equivalent (CEDE) from inhalation shall be those listed in Table 2.1 of Federal Guidelines Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion", EPA, 1988, as described in Regulatory Guide 1.183. The factors in the column headed "effective" yield doses corresponding to the CEDE. The conversion factors that are used for the calculation of EDE (or DDE) from external exposure (submersion) shall be those listed in Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil", EPA, 1993, as described in Regulatory Guide 1.183. The factors in the column headed "effective" yield doses corresponding to the EDE.**

<table>
<tr><th colspan="3">Description and Safety Assessment for Specific Changes to TS and TS Bases</th></tr>
<tr><td>Change #1</td><td colspan="2">Basis / Safety Assessment:

Per USNRC RG 1.183, Section 4, "The NRC Staff has determined that there is an implied synergy between the ASTs and total effective dose equivalent (TEDE) criteria, and the TID-14844 source terms and the whole body and thyroid dose criteria, and therefore, they do not expect to allow the TEDE criteria to be used with the TID-14844 calculated results." Additionally, the NRC stated that "The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent from inhalation and the deep dose equivalent from external exposure." The NRC also recommended that inhalation and submergence conversion factors should be taken from Table 2.1 of FGR11 and Table III.1 of FRG 12. Consequently, the existing DOSE EQUIVALENT I-131 definition, based on TID-14844 source terms and 10 CFR 100, was revised to exclude those accidents re-evaluated for the AST and for those accidents re-evaluated for the AST (MSLBA, LOCA, CRDA, and FHA), the DOSE EQUIVALENT I-131 definition was rewritten to update the dose conversion factors to reflect RG 1.183 and 10 CFR 50.67 requirements.

These revised dose conversion factors were used in the re-analyses of the design basis accidents using AST methodology. The revised accident analyses use inhalation committed effective dose equivalent dose conversion factors from FGR 11 and external committed effective dose equivalent dose conversion factors from FGR 12. Dose conversion factors from Regulatory Guide 1.109, Revision I, are used in other calculations of dose equivalency. With the implementation of AST, the accident dose guidelines of 10 CFR 100 are superseded by the dose criteria of 10 CFR 50.67. The whole body and thyroid doses of 10 CFR 100 are replaced by the total effective dose equivalent (TEDE) criteria of 10 CFR 50.67. A conforming change to the definition is to delete the word "thyroid" from the definition. The analyses performed in support of this amendment request determined radiological consequences in terms of the TEDE dose quantity and were shown to be in compliance with the dose criteria of 10 CFR 50.67. These changes to the definition are acceptable because they reflect adoption of the dose conversion factors and dose consequences of the revised radiological analyses.</td></tr>
<tr><td>Change #2</td><td>Current Technical Specification:

Unit 1 & 2, Section 3.1.7
LCO 3.1.7
APPLICABILITY: MODES 1 and 2
ACTIONS: REQUIRED ACTION: D.1 Be in MODE 3
ACTIONS: COMPLETION TIME: D.1 12 hours</td><td>Proposed Change:

Unit 1 & 2, Section 3.1.7
LCO 3.1.7
APPLICABILITY: MODES 1, 2, and 3
ACTIONS: REQUIRED ACTION: D.1, Be in mode 3 AND D.2 Be in MODE 4
ACTIONS: COMPLETION TIME: D.1 = 12 hours
ACTIONS: COMPLETION TIME: D.2 36 hours</td></tr>
<tr><td>Change #2</td><td colspan="2">Basis / Safety Assessment:

Boron injection from the SLC system is required for suppression pool pH control during a DBA LOCA. The maintenance of a suppression pool pH level above 7.0 is important to prevent re-evolution of iodine from the suppression pool water. Consequently, operation of the SLC system was revised to address reactor modes and timing during a DBA LOCA.</td></tr>
</table>

<table>
<tr><th colspan="3">Description and Safety Assessment for Specific Changes to TS and TS Bases</th></tr>
<tr><td>Change #3</td><td>Current Technical Specification:

Unit 1 & 2, Section 3.3.6.1
Table 3.3.6.1-1 (page 5 of 6)
Item 5.e, SLC System Initiation
APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS = 1, 2</td><td>Proposed Change:

Unit 1 & 2, Section 3.3.6.1
Table 3.3.6.1-1 (page 5 of 6)
Item 5.e, SLC System Initiation
APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS = 1, 2, 3</td></tr>
<tr><td>Change #3</td><td colspan="2">Basis / Safety Assessment:

Boron injection from the SLC system is required for suppression pool pH control during a DBA LOCA. The maintenance of a suppression pool pH level above 7.0 is important to prevent re-evolution of iodine from the suppression pool water. Consequently, operation of the SLC system was revised to address reactor modes during a DBA LOCA.</td></tr>
<tr><td>Change #4</td><td>Current Technical Specification:

Unit 1 & 2, Section 3.7.3
LCO 3.7.3
ACTIONS: CONDITION A: One CREOAS subsystem inoperable
ACTIONS: CONDITION B: Two CREOAS subsystems inoperable due to inoperable Control Room habitability envelope boundary in MODES 1, 2, and 3.
ACTIONS: REQUIRED ACTION: B.1 Restore Control Room habitability envelope boundary to OPERABLE status.
ACTIONS: COMPLETION TIME: B.1: 24 hours
ACTIONS: CONDITION F: Two CREOAS subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.
SR 3.7.3.4, SURVEILLANCE: Verify each CREOAS subsystem can maintain a positive pressure of ≥ 0.125 inches water gauge relative to the outside atmosphere during the pressurization/filtration mode of operation at a flow rate ≤ 5810 cfm.
SR 3.7.3.4, FREQUENCY: 24 months on a STAGGERED TEST BASIS</td><td>Proposed Change:

Unit 1 & 2, Section 3.7.3
LCO 3.7.3
ACTIONS: CONDITION A: One CREOAS subsystem inoperable for reasons other than Condition B
ACTIONS: CONDITION B: One or more CREOAS subsystems inoperable due to inoperable Control Room habitability envelope boundary in MODE 1, 2, or 3
ACTIONS: REQUIRED ACTION: B.1 Implement mitigating actions AND B.2 Restore Control Room habitability envelope boundary to OPERABLE status
ACTIONS: COMPLETION TIME: B.1:Immediately B.2: 24 hours
ACTIONS: CONDITION F: Two CREOAS subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs. OR Required Action and associated Completion Time of Condition B not met during movement of irradiated fuel assemblies in secondary containment, during CORE ALTERATIONS, or during OPDRVs.
SR 3.7.3.4, SURVEILLANCE: Verify Control Room boundary integrity in accordance with the Control Room habitability Program.
SR 3.7.3.4, FREQUENCY: In accordance with the Control Room habitability Program.</td></tr>
</table>

Description and Safety Assessment for Specific Changes to TS and TS Bases	
Change #4	**Basis / Safety Assessment:** In NRC Generic Letter 2003-01, Licensees were alerted to findings at facilities that existing Technical Specifications surveillance requirements for the Control Room Emergency Filtration System (CREFS) may not be adequate. Specifically, the results of tracer gas tests at facilities indicated that the differential pressure surveillance is not a reliable method for demonstrating Control Room integrity. The Technical Specification Task Force and the Nuclear Energy Institute Control Room Habitability Task Force have developed proposed changes to the Improved Standard Technical Specifications (NUREGs 1430 through 1434) to replace the differential pressure surveillance with a tracer gas surveillance and to institute a Control Room Habitability Program that will ensure that Control Room Habitability is maintained. These changes were incorporated into TSTF-448, Revision 2, Technical Specification Task Force – Improved Standard Technical Specifications Change Traveler (Reference 12.2). As a result of this Traveler, TS Section 3.7.3 was added.

Description and Safety Assessment for Specific Changes to TS and TS Bases		
Change #5	**Current Technical Specification:** Unit 1 & 2, Section 5.5 The last item number is 5.5.12, "Primary Containment Leakage Rate Testing Program". A new section 5.5.13 was added.	**Proposed Change:** Unit 1 & 2, Section 5.5 **5.5.13 <u>Control Room Habitability Program</u>** **A Control Room Habitability Program shall be established and implemented to ensure that Control Room habitability is maintained such that, with an OPERABLE CREOAS System, Control Room occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge from outside the Control Room envelope. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the Control Room under accident conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:** **a. The definition of the Control Room envelope and the Control Room boundary;** **b. Requirements for maintaining Control Room boundary integrity, including configuration control, management of breaches, and preventive maintenance.** **c. Requirements for assessing Control Room habitability at the frequencies specified in Regulatory Guide 1.197 "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003.** **d. Requirements for determining the unfiltered air inleakage past the Control Room boundary into the Control Room envelope in accordance with the testing methods and at the frequencies specified in Regulatory Guide 1.197, Revision 0, May 2003.** **Continued next page**

Description and Safety Assessment for Specific Changes to TS and TS Bases		
Change #5	**Current Technical Specification:** Continued	**Proposed Change:** **Continued** **e. Measurement of the Control Room envelope positive pressure relative to outside atmosphere during the pressurization mode of operation by one subsystem of the CREOAS System every 24 months on a STAGGERED TEST BASIS. The results shall be trended and compared to the positive pressure measurements taken or to be taken during the Control Room inleakage testing. These evaluations shall be used as part of an assessment of Control Room boundary integrity between Control Room inleakage tests.** **f. The quantitative limits on unfiltered air inleakage past the Control Room boundary into the Control Room envelope. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph d. The unfiltered air inleakage limits must demonstrate that radiation dose and hazardous chemical exposure to the Control Room occupants will be within the assumptions in the licensing basis.** **g. Limitations on the use of compensatory measures to consider the CREOAS System OPERABLE when there are degraded or nonconforming conditions that result in the unfiltered air inleakage through the Control Room boundary into the Control Room envelope greater than the unfiltered inleakage assumed in the licensing basis analyses. Compensatory measures are interim actions used to maintain OPERABILITY of the CREOAS System until full qualification of the Control Room boundary is restored. Degraded or nonconforming conditions affecting the Control Room boundary integrity should be resolved in a time frame commensurate with the safety significance of the condition.**

Description and Safety Assessment for Specific Changes to TS and TS Bases		
Change #5	**Current Technical Specification:** Continued	**Proposed Change:** Continued **The program shall place additional limits on the use of compensatory measures which address a degraded or nonconforming Control Room barrier that results in unfiltered air inleakage into the Control Room envelope greater than the unfiltered air inleakage assumed in the licensing basis analysis for the following two conditions:** **1. When such compensatory measures may adversely affect the ability of the Control Room occupants to respond to an accident (including, but not limited to, the use of personal air filtration or bottled air systems), their use may be credited to support OPERABILITY of the CREAOS System until the next entry into MODE 2 following a refueling outage or for a maximum of 12 months, whichever is greater; and** **2. When such compensatory measures may complicate the response of the Control Room occupants to an accident (including, but not limited to, the use of potassium iodine, temporary system configurations, or manual actions), their use may be credited to support OPERABILITY of the CREAOS System for a maximum of 36 months.** **The provision of SR 3.0.2 is applicable to the Control Room inleakage testing frequencies.**

Description and Safety Assessment for Specific Changes to TS and TS Bases	
Change #5	**Basis / Safety Assessment:** In NRC Generic Letter 2003-01, Licensees were alerted to findings at facilities that existing technical specifications surveillance requirements for the Control Room Emergency Filtration System (CREFS) may not be adequate. Specifically the results of tracer gas tests at facilities indicated that the differential pressure surveillance is not a reliable method for demonstrating Control Room integrity. The Technical Specification Task force and the Nuclear Energy Institute Control Room habitability task Force have developed proposed changes to the Improved Standard Technical Specifications (NUREGs 1430 through 1434) to replace the differential pressure surveillance with a tracer gas surveillance and to institute a Control Room Habitability Program that will ensure that Control Room habitability is maintained. These changes were incorporated into TSTF-448, Revision 2, Technical Specification Task Force – Improved Standard Technical Specifications Change Traveler (Reference 12.2). As a result of this Traveler, TS Section 5.5.13 was added. Please note, that PPL is aware that this Traveler may be revised in the near future and require additional revisions to TS Section 5.5.13.

Description and Safety Assessment for Specific Changes to TS and TS Bases		
Change #6	**Current Technical Specification:** The TS Bases provide an explanation and rationale for associated TS requirements, and in some cases, how they are to be implemented. The current TS Bases are written based on the requirements of 10 CFR 100 and the determination of a thyroid dose and whole body dose. Specific sections revised are as follows: Units 1 & 2 Bases Section 2.1.1.3, SAFETY LIMIT VIOLATIONS & REFERENCES Section B 2.1.2, BACKGROUND, SAFETY LIMIT VIOLATIONS, & REFERENCES Section B 3.1.8, APPLICABLE SAFETY ANALYSES & REFERENCES Section B 3.2.3, APPLICABLE SAFETY ANALYSES Section B 3.3.6.1, APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY Section B 3.3.6.2, APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY Section B 3.3.7.1, APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY Section B 3.4.7, BACKGROUND, APPLICABLE SAFETY ANALYSES, LCO, ACTIONS, & REFERENCES Section B 3.6.1.1, APPLICABLE SAFETY ANALYSES, SURVEILLANCE REQUIREMENTS Section B 3.6.1.3, SURVEILLANCE REQUIREMENT Section B 3.7.3 Section B 3.7.5, APPLICABLE SAFETY ANALYSES & REFERENCES Section B 3.7.7, APPLICABLE SAFETY ANALYSES & REFERENCES Section B 3.9.6, BACKGROUND, APPLICABLE SAFETY ANALYSES, LCO, & REFERENCES	**Proposed Change:** Associated changes to the TS Bases were made to delete the thyroid dose and whole body dose requirements of 10 CFR 100. A general statement that doses would be maintained within regulatory limits replaced the deleted text. The TS Bases were clarified where applicable to include the Control Room dose. All these changes are administrative in nature.

Description and Safety Assessment for Specific Changes to TS and TS Bases		
Change #6	**Basis / Safety Assessment:** The accident dose guidelines of 10 CFR 100 are superseded by the dose criteria of 10 CFR 50.67 for accidents utilizing the AST methodology. The whole body and thyroid doses of 10 CFR 100 are replaced by the total effective dose equivalent (TEDE) criteria of 10 CFR 50.67, and references to 10 CFR 100 are replaced with 10 CFR 50.67 where applicable. The TS Bases were clarified where applicable to include the Control Room dose. This is a conforming change. Other changes were made to the TS Bases for clarity and to conform to the changes made to the associated TS. The revisions to the TS bases incorporate supporting information for the proposed TS changes. Bases do not establish actual requirements, and as such, do not change technical requirements of the TS. The Bases changes are therefore acceptable, since they administratively document the reasons and provide additional understanding for the associated TS requirements.	
Change #7	**Current Technical Specification:** The TS Bases provide an explanation and rationale for associated TS requirements, and in some cases, how they are to be implemented. The current TS Bases of Sections B 3.1.7 and B 3.3.6.1, APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY, subsection 5.e., are written such that actuation of the SLC system is not required during a DBA LOCA. Per the discussion in Attachment 2 of this LAR, actuation of the SLC system is required in order to maintain suppression pool pH ≥ 7 in order to prevent re-evolution of iodines from the pool.	**Proposed Change:** Associated changes to the TS Bases were made to conform to the new requirement that actuation of the SLC system is required during a DBA LOCA. All these changes are administrative in nature.
Change #7	**Basis / Safety Assessment:** Changes were made to the TS Bases for clarity and to conform to the changes made to the associated TS. The revisions to the TS bases incorporate supporting information for the proposed TS changes. Bases do not establish actual requirements, and as such, do not change technical requirements of the TS. The Bases changes are therefore acceptable, since they administratively document the reasons and provide additional understanding for the associated TS requirements.	
Change #8	**Current Technical Specification:** The TS Bases provide an explanation and rationale for associated TS requirements, and in some cases, how they are to be implemented. The current TS Bases of Section B 3.6.1.3, APPLICABLE SAFETY ANALYSES, discusses the MSLB event. Per the discussion in Attachment 2 of this LAR, the MSLB event occurs outside primary containment.	**Proposed Change:** Associated changes to the TS Bases were made to clarify that the MSLB event occurs outside primary containment. This change is administrative in nature.

<table>
<tr><th colspan="3">Description and Safety Assessment for Specific Changes to TS and TS Bases</th></tr>
<tr><td>Change #8</td><td colspan="2">Basis / Safety Assessment:

Changes were made to the TS Bases for clarity and to conform to the changes made to the associated TS. The revisions to the TS bases incorporate supporting information for the proposed TS changes. Bases do not establish actual requirements, and as such, do not change technical requirements of the TS. The Bases changes are therefore acceptable, since they administratively document the reasons and provide additional understanding for the associated TS requirements.</td></tr>
<tr><td>Change #9</td><td>Current Technical Specification:

The TS Bases provide an explanation and rationale for associated TS requirements, and in some cases, how they are to be implemented. The current TS Bases of Section B 3.6.4.1, SURVEILLANCE REQUIREMENTS, provides the maximum drawdown time required by the SGTS to establish and maintain the secondary containment to ≥ 0.25 inches of vacuum water gauge. Per Attachment 2 of this LAR, a drawdown time of 600 seconds was utilized in the DBA LOCA analysis.</td><td>Proposed Change:

Associated changes to the TS Bases were made to provide some relief for establishing secondary containment drawdown pressure and still provide significant margin with DBA LOCA drawdown time requirements. The maximum drawdown time was increased from 125 and 117 seconds for Zones I, II, & III and Zones I & III respectively, to 300 seconds for both cases.

This change is administrative in nature.</td></tr>
<tr><td>Change #9</td><td colspan="2">Basis / Safety Assessment:

Changes to the TS Bases were made to provide some relief for establishing secondary containment drawdown pressure and still provide significant margin with DBA LOCA drawdown time requirements. The surveillance requirement establishes a time of 300 seconds for the maximum drawdown time. The DBA LOCA analysis assumes a maximum drawdown time of 10 minutes for the unfiltered release to the environs. Consequently, the change in the allowable maximum drawdown time does not represent an increase in the calculated Control Room, EAB, or LPZ doses.

Changes were made to the TS Bases for clarity and to conform to the changes made to the associated TS. The revisions to the TS bases incorporate supporting information for the proposed TS changes. Bases do not establish actual requirements, and as such, do not change technical requirements of the TS. The Bases changes are therefore acceptable, since they administratively document the reasons and provide additional understanding for the associated TS requirements.</td></tr>
<tr><td>Change #10</td><td>Current Technical Specification:

The TS Bases provide an explanation and rationale for associated TS requirements, and in some cases, how they are to be implemented. The current TS Bases of Section B 3.7.3 discusses the Control Room emergency outside air supply (CREOAS) system.</td><td>Proposed Change:

Associated changes to the TS Bases were made to address the requirements of TSTF-448, Revision 2, Technical Specification Task Force – Improved Standard Technical Specifications Change Traveler and incorporate the Control Room Habitability Program.

This change is administrative in nature.</td></tr>
</table>

<table>
<tr><th colspan="3">Description and Safety Assessment for Specific Changes to TS and TS Bases</th></tr>
<tr><td>Change #10</td><td colspan="2">Basis / Safety Assessment:

Changes were made to the TS Bases for clarity and to conform to the changes made to the associated TS. The revisions to the TS bases incorporate supporting information for the proposed TS changes. Bases do not establish actual requirements, and as such, do not change technical requirements of the TS. The Bases changes are therefore acceptable, since they administratively document the reasons and provide additional understanding for the associated TS requirements.

In NRC Generic Letter 2003-01, Licensees were alerted to findings at facilities that existing technical specifications surveillance requirements for the Control Room Emergency Filtration System (CREFS) may not be adequate. Specifically, the results of tracer gas tests at facilities indicated that the differential pressure surveillance is not a reliable method for demonstrating Control Room integrity.

The Technical Specification Task force and the Nuclear Energy Institute Control Room Habitability Task Force have developed proposed changes to the Improved Standard Technical Specifications (NUREGs 1430 through 1434) to replace the differential pressure surveillance with a tracer gas surveillance and to institute a Control Room Habitability Program that will ensure that Control Room habitability is maintained.

These changes were incorporated into TSTF-448, Revision 2, Technical Specification Task Force – Improved Standard Technical Specifications Change Traveler (Reference 12.2). As a result of this Traveler, TS Section 3.7.3 was revised. Please note, that PPL is aware that this Traveler may be revised in the near future and require additional revisions to TS Section 3.7.3.</td></tr>
<tr><td>Change #11</td><td>Current Technical Specification:

The TS Bases provide an explanation and rationale for associated TS requirements, and in some cases, how they are to be implemented. The current TS Bases of Section B 3.9.6, APPLICABLE SAFETY ANALYSES, provides the fuel rod gap release fractions per RG 1.25 for a FHA. Per Attachment 2 of this LAR, the FHA was revised to reflect the new release fractions of RG 1.183.</td><td>Proposed Change:

Associated changes to the TS Bases were made to update the FHA to reflect RG 1.183 fuel rod gap release fractions. The original analysis assumes that 10% of the total fuel rod iodine inventory in the gap is available for release. Per RG 1.183 requirements, 8% of the I-131, and 5% of the I-132, I-133, I-134, & I-135 inventory is available for release from the gap.

This change is administrative in nature.</td></tr>
<tr><td>Change #11</td><td colspan="2">Basis / Safety Assessment:

Changes were made to the TS Bases for clarity and to conform to the changes made to the associated TS. The revisions to the TS bases incorporate supporting information for the proposed TS changes. Bases do not establish actual requirements, and as such, do not change technical requirements of the TS. The Bases changes are therefore acceptable, since they administratively document the reasons and provide additional understanding for the associated TS requirements.</td></tr>
</table>

Attachment 6 to PLA-5963

Proposed Technical Specification Changes
Units 1 & 2 Mark-ups

Table 6-1: List of Proposed Technical Specification Changes (Marked ups)

Units 1 & 2 Sections	Title
1.1	Definitions
3.1.7	Standby Liquid Control (SLC) System
3.3.6.1	Primary Containment Isolation Instrumentation
3.7.3	Control Room Emergency Outside Air Supply (CREOAS) System
5.5	Programs and Manuals

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm, interlock, display, and trip functions, and channel failure trips. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:

a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and

b. Control rod movement, provided there are no fuel assemblies in the associated core cell.

Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)

The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131

INSERT 1 →

~~DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors shall be those listed in ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."~~

(continued)

Insert 1 to DOSE EQUIVALENT I-131 definition:

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) which alone would produce the same total effective dose equivalent (sum of committed effective dose equivalent {CEDE} from inhalation plus deep dose equivalent {DDE} or nominally equivalent to the effective dose equivalent {EDE} from external exposure {submersion}) as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The conversion factors that are used for this calculation of committed effective dose equivalent (CEDE) from inhalation shall be those listed in Table 2.1 of Federal Guidelines Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion", EPA, 1988, as described in Regulatory Guide 1.183. The factors in the column headed "effective" yield doses corresponding to the CEDE. The conversion factors that are used for the calculation of EDE (or DDE) from external exposure (submersion) shall be those listed in Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil", EPA, 1993, as described in Regulatory Guide 1.183. The factors in the column headed "effective" yield doses corresponding to the EDE.

DOSE EQUIVALENT I-131 (continued)

~~Existing calculations using conversion factors listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites" or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, are acceptable.~~

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

END OF CYCLE RECIRCULATION PUMP TRIP (EOC RPT) SYSTEM RESPONSE TIME

The EOC RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by the associated turbine stop valve limit switch or from when the turbine control valve hydraulic oil control oil pressure drops below the pressure switch setpoint to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

ISOLATION SYSTEM RESPONSE TIME

The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

(continued)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, ~~and~~ 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Concentration of sodium pentaborate in solution < 13.6 weight percent but within limits of Figure 3.1.7-1.	A.1 Restore concentration of sodium pentaborate in solution to within limits >13.6 weight percent.	72 hours <u>AND</u> 10 days from discovery of failure to meet the LCO
B. One SLC subsystem inoperable for reasons other than Condition A.	B.1 Restore SLC subsystem to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO.
C. Two SLC subsystems inoperable for reasons other than Condition A.	C.1 Restore one SLC subsystem to OPERABLE status.	8 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4	12 hours 36 hours

Table 3.3.6.1-1 (page 5 of 6)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Reactor Water Cleanup (RWCU) System Isolation					
a. RWCU Differential Δ Flow – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 67 gpm
b. RWCU Penetration Area Temperature - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤ 137°F
c. RWCU Pump Area Temperature – High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤ 154°F
d. RWCU Heat Exchanger Area Temperature – High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤ 154°F
e. SLC System Initiation	1,2, 3	2[(b)]	I	SR 3.3.6.1.5	NA
f. Reactor Vessel Water Level -Low Low, Level 2	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ -45 inches
g. RWCU Flow - High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 472 gpm
h. Manual Initiation	1,2,3	1	G	SR 3.3.6.1.5	NA

(continued)

(b) SLC System Initiation only inputs into one of the two trip systems.

3.7 PLANT SYSTEMS

3.7.3 Control Room Emergency Outside Air Supply (CREOAS) System

LCO 3.7.3 Two CREOAS subsystems shall be OPERABLE.

-----NOTE-----
The control room habitability envelope boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREOAS subsystem inoperable, for reasons other than Condition B.	A.1 Restore CREOAS subsystem to OPERABLE status.	7 days
B. ~~Two~~ One or more CREOAS subsystems inoperable due to inoperable control room habitability envelope boundary in MODE~~S~~ 1, 2 ~~and~~ or 3.	Insert 1 B.~~1~~ 2 Restore control room habitability envelope boundary to OPERABLE status.	Immediately 24 hours
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

(continued)

Insert 1: B.1 Implement mitigating actions

AND

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Two CREOAS subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs. OR Required Action and associated Completion Time of Condition B not met during movement of irradiated fuel assemblies in secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ------------------------------------	
	F.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	F.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	F.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Operate each CREOAS filter train for ≥ 10 continuous hours with the heaters operable.	31 days
SR 3.7.3.2	Perform required CREOAS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.3.3	Verify each CREOAS subsystem actuates on an actual or simulated initiation signal.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.7.3.4~~	~~Verify each CREOAS subsystem can maintain a positive pressure of ≥ 0.125 inches water gauge relative to the outside atmosphere during the pressurization/filtration mode of operation at a flow rate of ≤ 5810 cfm.~~	~~24 months on a STAGGERED TEST BASIS~~
SR 3.7.3.4	Verify control room boundary integrity in accordance with the Control Room Habitability Program.	In accordance with the Control Room Habitability Program

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established, implemented, and maintained to comply with the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", dated September 1995, as modified by the following exception:

a. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the May 4, 1992 Type A test shall be performed no later than May 3, 2007.

The peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, is 45.0 psig.

The maximum allowable primary containment leakage rate, La, at Pa, shall be 1% of the primary containment air weight per day.

Leakage Rate Acceptance Criteria are:

a. Primary Containment leakage rate acceptance criterion is ≤ 1.0 La. During each unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 La for Type B and Type C tests and ≤ 0.75 La for Type A tests:

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is ≤ 0.05 La when tested at ≥ Pa.

   2) For each door, leakage rate is ≤ 5 scfh when pressurized to ≥ 10 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

Insert 2

Insert 2:

5.5.13 Control Room Habitability Program

A Control Room Habitability Program shall be established and implemented to ensure that control room habitability is maintained such that, with an OPERABLE CREOAS System, control room occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge from outside the control room envelope. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the control room envelope and the control room boundary;
b. Requirements for maintaining control room boundary integrity, including configuration control, management of breaches, and preventive maintenance.
c. Requirements for assessing control room habitability at the frequencies specified in Regulatory Guide 1.197 "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003.
d. Requirements for determining the unfiltered air inleakage past the control room boundary into the control room envelope in accordance with the testing methods and at the frequencies specified in Regulatory Guide 1.197, Revision 0, May 2003.
e. Measurement of the control room envelope positive pressure relative to outside atmosphere during the pressurization mode of operation by one subsystem of the CREOAS System every 24 months on a STAGGERED TEST BASIS. The results shall be trended and compared to the positive pressure measurements taken or to be taken during the control room inleakage testing. These evaluations shall be used as part of an assessment of control room boundary integrity between control room inleakage tests.
f. The quantitative limits on unfiltered air inleakage past the control room boundary into the control room envelope. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph d. The unfiltered air inleakage limits must demonstrate that radiation dose and hazardous chemical exposure to the control room occupants will be within the assumptions in the licensing basis.
g. Limitations on the use of compensatory measures to consider the CREOAS System OPERABLE when there are degraded or nonconforming conditions that result in the unfiltered air inleakage through the control room boundary into the control room envelope greater than the unfiltered inleakage assumed in the licensing basis analyses. Compensatory measures are interim actions used to maintain OPERABILITY of the CREOAS System until full qualification of the control room boundary is restored. Degraded or nonconforming conditions affecting the control room boundary integrity

should be resolved in a time frame commensurate with the safety significance of the condition. The program shall place additional limits on the use of compensatory measures which address a degraded or nonconforming control room barrier that results in unfiltered air inleakage into the control room envelope greater than the unfiltered air inleakage assumed in the licensing basis analysis for the following two conditions:

1. When such compensatory measures may adversely affect the ability of the control room occupants to respond to an accident (including, but not limited to, the use of personal air filtration or bottled air systems), their use may be credited to support OPERABILITY of the CREAOS System until the next entry into MODE 2 following a refueling outage or for a maximum of 12 months, whichever is greater; and
2. When such compensatory measures may complicate the response of the control room occupants to an accident (including, but not limited to, the use of potassium iodine, temporary system configurations, or manual actions), their use may be credited to support OPERABILITY of the CREAOS System for a maximum of 36 months.

The provision of SR 3.0.2 is applicable to the control room inleakage testing frequencies.

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm, interlock, display, and trip functions, and channel failure trips. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:

a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and

b. Control rod movement, provided there are no fuel assemblies in the associated core cell.

Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)

The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131

INSERT 1 →

~~DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors shall be those listed in ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."~~

(continued)

Insert 1 to DOSE EQUIVALENT I-131 definition:

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) which alone would produce the same total effective dose equivalent (sum of committed effective dose equivalent {CEDE} from inhalation plus deep dose equivalent {DDE} or nominally equivalent to the effective dose equivalent {EDE} from external exposure {submersion}) as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The conversion factors that are used for this calculation of committed effective dose equivalent (CEDE) from inhalation shall be those listed in Table 2.1 of Federal Guidelines Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion", EPA, 1988, as described in Regulatory Guide 1.183. The factors in the column headed "effective" yield doses corresponding to the CEDE. The conversion factors that are used for the calculation of EDE (or DDE) from external exposure (submersion) shall be those listed in Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil", EPA, 1993, as described in Regulatory Guide 1.183. The factors in the column headed "effective" yield doses corresponding to the EDE.

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	~~Existing calculations using conversion factors listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites" or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, are acceptable.~~
EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME	The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
END OF CYCLE RECIRCULATION PUMP TRIP (EOC RPT) SYSTEM RESPONSE TIME	The EOC RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by the associated turbine stop valve limit switch or from when the turbine control valve hydraulic oil control oil pressure drops below the pressure switch setpoint to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
ISOLATION SYSTEM RESPONSE TIME	The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

(continued)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, ~~and~~ 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Concentration of sodium pentaborate in solution < 13.6 weight percent but within limits of Figure 3.1.7-1.	A.1 Restore concentration of sodium pentaborate in solution to within limits > 13.6 weight percent.	72 hours AND 10 days from discovery of failure to meet the LCO
B. One SLC subsystem inoperable for reasons other than Condition A.	B.1 Restore SLC subsystem to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
C. Two SLC subsystems inoperable for reasons other than Condition A.	C.1 Restore one SLC subsystem to OPERABLE status.	8 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 36 hours

Table 3.3.6.1-1 (page 5 of 6)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
h. Manual Initiation	1,2,3	1	G	SR 3.3.6.1.5	NA
5. Reactor Water Cleanup (RWCU) System Isolation					
a. RWCU Differential Δ Flow – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 67 gpm
b. RWCU Penetration Area Temperature - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤ 137°F
c. RWCU Pump Area Temperature – High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤ 154°F
d. RWCU Heat Exchanger Area Temperature – High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤ 154°F
e. SLC System Initiation	1,2, 3	2(b)	I	SR 3.3.6.1.5	NA
f. Reactor Vessel Water Level -Low Low, Level 2	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ -45 inches
g. RWCU Flow - High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 472 gpm
h. Manual Initiation	1,2,3	1	G	SR 3.3.6.1.5	NA

(continued)

(b) SLC System Initiation only inputs into one of the two trip systems.

3.7 PLANT SYSTEMS

3.7.3 Control Room Emergency Outside Air Supply (CREOAS) System

LCO 3.7.3 Two CREOAS subsystems shall be OPERABLE.

------------------------------NOTES------------------------------

The control room habitability envelope boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREOAS subsystem inoperable for reasons other than Condition B.	A.1 Restore CREOAS subsystem to OPERABLE status.	7 days
B. ~~Two~~ One or more CREOAS subsystems inoperable due to inoperable control room habitability envelope boundary in MODE~~S~~ 1, 2, ~~and~~ or 3.	Insert 1 B.~~1~~2 Restore control room habitability envelope boundary to OPERABLE status.	Immediately 24 hours
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

(continued)

Insert 1: B.1. Implement mitigating actions

AND

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Two CREOAS subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs. OR Required Action and associated Completion Time of Condition B not met during movement of irradiated fuel assemblies in secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. -----------------------------------	
	F.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	F.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	F.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Operate each CREOAS filter train for ≥ 10 continuous hours with the heaters operable.	31 days
SR 3.7.3.2	Perform required CREOAS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.3.3	Verify each CREOAS subsystem actuates on an actual or simulated initiation signal.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.7.3.4~~	~~Verify each CREOAS subsystem can maintain a positive pressure of ≥ 0.125 inches water gauge relative to the outside atmosphere during the pressurization/filtration mode of operation at a flow rate of ≤ 5810 cfm.~~	~~24 months on a STAGGERED TEST BASIS~~
SR 3.7.3.4	Verify control room boundary integrity in accordance with the Control Room Habitability Program.	In accordance with the Control Room Habitability Program

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established, implemented, and maintained to comply with the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", dated September 1995, as modified by the following exception:

a. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the October 31, 1992 Type A test shall be performed no later than October 30, 2007.

The peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, is 45.0 psig.

The maximum allowable primary containment leakage rate, La, at Pa, shall be 1% of the primary containment air weight per day.

Leakage Rate Acceptance Criteria are:

a. Primary Containment leakage rate acceptance criterion is ≤ 1.0 La. During each unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 La for Type B and Type C tests and ≤ 0.75 La for Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is ≤ 0.05 La when tested at ≥ Pa,

   2) For each door, leakage rate is ≤ 5 scfh when pressurized to ≥ 10 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

Insert 2

Insert 2:

5.5.13 <u>Control Room Habitability Program</u>

A Control Room Habitability Program shall be established and implemented to ensure that control room habitability is maintained such that, with an OPERABLE CREOAS System, control room occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge from outside the control room envelope. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the control room envelope and the control room boundary;
b. Requirements for maintaining control room boundary integrity, including configuration control, management of breaches, and preventive maintenance.
c. Requirements for assessing control room habitability at the frequencies specified in Regulatory Guide 1.197 "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003.
d. Requirements for determining the unfiltered air inleakage past the control room boundary into the control room envelope in accordance with the testing methods and at the frequencies specified in Regulatory Guide 1.197, Revision 0, May 2003.
e. Measurement of the control room envelope positive pressure relative to outside atmosphere during the pressurization mode of operation by one subsystem of the CREOAS System every 24 months on a STAGGERED TEST BASIS. The results shall be trended and compared to the positive pressure measurements taken or to be taken during the control room inleakage testing. These evaluations shall be used as part of an assessment of control room boundary integrity between control room inleakage tests.
f. The quantitative limits on unfiltered air inleakage past the control room boundary into the control room envelope. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph d. The unfiltered air inleakage limits must demonstrate that radiation dose and hazardous chemical exposure to the control room occupants will be within the assumptions in the licensing basis.
g. Limitations on the use of compensatory measures to consider the CREOAS System OPERABLE when there are degraded or nonconforming conditions that result in the unfiltered air inleakage through the control room boundary into the control room envelope greater than the unfiltered inleakage assumed in the licensing basis analyses. Compensatory measures are interim actions used to maintain OPERABILITY of the CREOAS System until full qualification of the control room boundary is restored. Degraded or nonconforming conditions affecting the control room boundary integrity

should be resolved in a time frame commensurate with the safety significance of the condition. The program shall place additional limits on the use of compensatory measures which address a degraded or nonconforming control room barrier that results in unfiltered air inleakage into the control room envelope greater than the unfiltered air inleakage assumed in the licensing basis analysis for the following two conditions:

1. When such compensatory measures may adversely affect the ability of the control room occupants to respond to an accident (including, but not limited to, the use of personal air filtration or bottled air systems), their use may be credited to support OPERABILITY of the CREAOS System until the next entry into MODE 2 following a refueling outage or for a maximum of 12 months, whichever is greater; and
2. When such compensatory measures may complicate the response of the control room occupants to an accident (including, but not limited to, the use of potassium iodine, temporary system configurations, or manual actions), their use may be credited to support OPERABILITY of the CREAOS System for a maximum of 36 months.

The provision of SR 3.0.2 is applicable to the control room inleakage testing frequencies.

Attachment 7 to PLA-5963

For Information - Proposed Technical Specification Bases Changes Units 1 & 2 Mark-ups

Table 7-1: List of Proposed Technical Specification Bases Changes (Marked ups)

Units 1 & 2 Sections	Title
B 2.1.1	Reactor Core SLs
B 2.1.2	Reactor Coolant System (RCS) Pressure SL
B 3.1.7	Standby Liquid Control (SLC) System
B 3.1.8	Scram Discharge Volume (SDV) Vent and Drain Valves
B 3.2.3	Linear Heat Generation Rate (LHGR)
B 3.3.6.1	Primary Containment Isolation Instrumentation
B 3.3.6.2	Secondary Containment Isolation Instrumentation
B 3.3.7.1	Control Room Emergency Outside Air Supply (CREOAS) System Instrumentation
B 3.4.7	Reactor Coolant Specific Activity
B 3.6.1.1	Primary Containment
B 3.6.1.3	Primary Containment Isolation Valves (PCIVs)
B 3.6.4.1	Secondary Containment
B 3.7.3	Control Room Emergency Outside Air Supply (CREOAS) System
B 3.7.5	Main Condenser Offgas
B 3.7.7	Spent Fuel Storage Pool Water Level
B 3.9.6	Reactor Pressure Vessel (RPV) Water Level

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.3 Reactor Vessel Water Level (continued)

monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

regulatory

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of ~~10 CFR 100, "Reactor Site Criteria,"~~ limits ~~(Ref. 3)~~. Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

2. ANF 524 (P)(A), Revision 2, "Critical Power Methodology for Boiling Water Reactors," Supplement 1 Revision 2 and Supplement 2, November 1990.

3. ~~10 CFR 100~~. DELETED

4. EMF-1997 (P)(A), Revision 0, "ANFB-10 Critical Power Correlation," July 1998 and EMF-1997 (P)(A) Supplement 1 Revision 0, "ANFB-10 Critical Power Correlation: High Local Peaking Results," July 1998.

5. EMF-2158(P)(A), Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/Microburn-B2," October 1999.

(continued)

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and anticipated operational occurrences (AOOs).

During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." Following inception of unit operation, RCS components shall be pressure tested in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB, reducing the number of protective barriers designed to prevent radioactive releases from exceeding ~~the~~ regulatory limits ~~specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4)~~. If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor High Flux and Vessel Steam Dome Pressure—High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME, Boiler and Pressure Vessel Code, 1968 Edition, including Addenda through the summer of 1970 (Ref. 5), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The SL of 1325 psig, as measured in the reactor steam dome, is equivalent to 1375 psig at the lowest elevation of the RCS. The RCS inside containment is designed to the ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition with Addenda through summer of 1972 (Ref. 6), for the reactor recirculation piping, which permits a maximum pressure transient of 110% of design pressures of 1250 psig for suction piping and 1500 psig for discharge piping. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 110% of design pressures of 1250 psig for suction piping and 1500 psig for discharge piping. The most limiting of these allowances is the 110% of the suction piping design pressures; therefore, the SL on maximum allowable RCS pressure is established at 1325 psig as measured at the reactor steam dome.

APPLICABILITY

SL 2.1.2 applies in all MODES.

SAFETY LIMIT VIOLATIONS

regulatory

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of ~~10 CFR 100, "Reactor Site Criteria,"~~ limits ~~(Ref. 4)~~. Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also assures that the probability of an accident occurring during this period is minimal.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IW-5000.
4. ~~10 CFR 100~~. DELETED
5. ASME, Boiler and Pressure Vessel Code, Section III, 1968 Edition, Addenda summer of 1970.
6. ASME, Boiler and Pressure Vessel Code, Section III, 1971 Edition, Addenda summer of 1972.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

The SLC System is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. Insert 1 The SLC system satisfies the requirements of 10 CFR 50.62 (Ref. 1) for anticipated transient without scram.

The SLC System consists of a sodium pentaborate solution storage tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core. A smaller tank containing demineralized water is provided for testing purposes.

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods or if fuel damage occurs post-LOCA. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner or if fuel damage occurs post-LOCA. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron, which produces a concentration of 660 ppm of natural boron, in the reactor coolant at 68°F. To allow for potential leakage and imperfect mixing in the reactor system, an amount of boron equal to 25% of the amount cited above is added (Ref. 2). The volume versus concentration limits in Figure 3.1.7-1 and the temperature versus concentration limits in Figure 3.1.7-2 are calculated such that the required concentration is achieved accounting for dilution in the RPV with normal water level and including the water volume in

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

the residual heat removal shutdown cooling piping and in the recirculation loop piping. This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected. The minimum concentration of 13.6 weight percent ensures compliance with the requirements of 10 CFR 50.62 (Ref. 1).

INSERT 2

The SLC System satisfies the requirements of the NRC Policy Statement (Ref. 3) because operating experience and probabilistic risk assessments have shown the SLC System to be important to public health and safety. Thus, it is retained in the Technical Specifications.

LCO

The OPERABILITY of the SLC System provides backup capability for reactivity control independent of normal reactivity control provisions provided by the control rods. INSERT 3 The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE; each contains an OPERABLE pump, an explosive valve, and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, control rods are not able to be withdrawn (except as permitted by LCO 3.10.3 and LCO 3.10.4) since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to be OPERABLE when only a single control rod can be withdrawn.

INSERT 4

(continued)

BASES

ACTIONS

A.1

and suppression pool pH control

If the boron solution concentration is less than the required limits for compliance with 10 CFR 50.62 (Ref. 1) (≥ 13.6 weight percent) but greater than the concentration required for cold shutdown (original licensing basis), the concentration must be restored to within limits > 13.6 weight percent in 72 hours. It is not necessary under these conditions to enter Condition C for both SLC subsystems inoperable since they are capable of performing their original design basis functions. Because of the low probability of an event and the fact that the SLC System capability still exists for vessel injection under these conditions, the allowed Completion Time of 72 hours is acceptable and provides adequate time to restore concentration to within limits.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of concentration out of limits or inoperable SLC subsystems during any single continuous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, an SLC subsystem is inoperable and that subsystem is subsequently returned to OPERABLE, the LCO may already have been not met for up to 7 days. This situation could lead to a total duration of 10 days (7 days in Condition B, followed by 3 days in Condition A), since initial failure of the LCO, to restore the SLC System. Then an SLC subsystem could be found inoperable again, and concentration could be restored to within limits. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock," resulting in establishing the "time zero" at the time the LCO was initially not met instead of at the time Condition A was entered. The 10 day Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

B.1

If one SLC subsystem is inoperable for reasons other than Condition A, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to perform the

(continued)

BASES

ACTIONS

B.1 (continued)

and provide adequate buffering agent to the suppression pool.

shutdown function. However, the overall reliability is reduced because a single failure in the remaining OPERABLE subsystem could result in reduced SLC System shutdown capability. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of performing the intended SLC System functions and the low probability of an event occurring ~~concurrent with the failure of the Control Rod Drive (CRD) System to shut down the plant.~~ requiring SLC injection

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of concentration out of limits or inoperable SLC subsystems during any single continuous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, concentration is out of limits, and is subsequently returned to within limits, the LCO may already have been not met for up to 3 days. This situation could lead to a total duration of 10 days (3 days in Condition A, followed by 7 days in Condition B), since initial failure of the LCO, to restore the SLC System. Then concentration could be found out of limits again, and the SLC subsystem could be restored to OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock," resulting in establishing the "time zero" at the time the LCO was initially not met instead of at the time Condition B was entered. The 10 day Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

C.1

If both SLC subsystems are inoperable for reasons other than Condition A, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable given the low probability of an event occurring ~~concurrent with the failure of the control rods to shut down the reactor.~~ requiring SLC injection.

(continued)

BASES

ACTIONS (continued)

D.1

and MODE 4 within 36 hours

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant ~~must be brought to MODE~~ 3 within 12 hours. The allowed Completion Time are ~~of 12 hours is~~ reasonable, based on operating experience, ~~to reach MODE 3 from~~ full power conditions in an orderly manner and without challenging ~~plant systems.~~ to reach the required plant conditions from

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3

SR 3.1.7.1 through SR 3.1.7.3 are 24 hour Surveillances verifying certain characteristics of the SLC System (e.g., the volume and temperature of the borated solution in the storage tank), thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure that the proper borated solution volume and temperature, including the temperature of the pump suction piping, are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the sodium pentaborate remains in solution and does not precipitate out in the storage tank or in the pump suction piping. The temperature versus concentration curve of Figure 3.1.7-2 ensures that a 10°F margin will be maintained above the saturation temperature. An alternate method of performing SR 3.1.7.3 is to verify the OPERABILITY of the SLC heat trace system. This verifies the continuity of the heat trace lines and ensures proper heat trace operation, which ensure that the SLC suction piping temperature is maintained. The 24 hour Frequency is based on operating experience and has shown there are relatively slow variations in the measured parameters of volume and temperature.

SR 3.1.7.4 and SR 3.1.7.6

SR 3.1.7.4 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.7

Demonstrating that each SLC System pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1224 psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. Insert 5 This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this Surveillance is in accordance with the Inservice Testing Program.

SR 3.1.7.8 and SR 3.1.7.9

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The pump and explosive valve tested should be alternated such that both complete flow paths are tested every 48 months at alternating 24 month intervals. The Surveillance may be performed in separate steps to prevent injecting solution into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Demonstrating that all heat traced piping between the boron solution storage tank and the suction inlet to the injection

(continued)

Insert 1: Additionally, the SLC System is designed to provide sufficient buffering agent to maintain the suppression pool pH at or above 7.0 following a DBA LOCA involving fuel damage. Maintaining the suppression pool pH at or above 7.0 will mitigate the re-evolution of iodine from the suppression pool water following a DBA LOCA.

Insert 2: The SLC system is also used to control Suppression Pool pH in the event of a DBA LOCA by injecting sodium pentaborate into the reactor vessel. The sodium pentaborate is then transported to the suppression pool and mixed by ECCS flow recirculation through the reactor, out of the break and into the suppression chamber. The amount of sodium pentaborate solution that must be available for injection following a DBA LOCA is determined as part of the DBA LOCA radiological analysis. This quantity is maintained in the storage tank as specified in the Technical Specification.

Insert 3: and provides sufficient buffering agent to maintain the suppression pool pH at or above 7.0 following a DBA LOCA involving fuel damage.

Insert 4: A DBA LOCA that results in the release of radioactive material is possible in MODES 1, 2 and 3 therefore capability to buffer the suppression pool pH is required. In MODES 4 and 5 a DBA LOCA with a radioactive release need not be postulated.

Insert 5: Additionally, the minimum pump flow rate requirement ensures that adequate buffering agent will reach the suppression pool to maintain pH above 7.0.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV is a volume of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two SDVs (headers) and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. The two instrument volumes are connected to a common drain line with two valves in series. Each header is connected to a common vent line with two valves in series. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 1.

APPLICABLE SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all of the control rods are capable of scramming. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite and control room doses remain within ~~the~~ regulatory limits ~~of 10 CFR 100 (Ref. 2)~~; and

b. Open on scram reset to maintain the SDV vent and drain path open so that there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be ~~terminated~~ by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite and control room doses are well within ~~the~~ regulatory limits ~~of 10 CFR 100 (Ref. 2)~~, and adequate core cooling is maintained (Ref. 3). The SDV vent and drain valves allow continuous drainage of the SDV during normal plant operation

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.3 (continued)

30 seconds after receipt of a scram signal is based on the bounding leakage case evaluated in the accident analysis based on the requirements of Reference 2. Similarly, after receipt of a simulated or actual scram reset signal, the opening of the SDV vent and drain valves is verified. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1 and the scram time testing of control rods in LCO 3.1.3 overlap this Surveillance to provide complete testing of the assumed safety function. The 24 month Frequency is based on the need to perform portions of this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 4.6.
2. ~~10 CFR 100~~. DELETED
3. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981.
4. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation. Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during the normal operations identified in Reference 1.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References ~~1,~~ 2, 3, and 4. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of ~~the guidelines of 10 CFR, Parts 20, 50, and 100.~~ regulatory limits. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet; and

b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 3).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. A separate evaluation was performed to determine the limits of LHGR during anticipated operational occurrences. This limit,

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The penetrations which are isolated by the below listed functions can be determined by referring to the PCIV Table found in the Bases of LCO 3.6.1.3, "Primary Containment Isolation Valves."

Main Steam Line Isolation

1.a. Reactor Vessel Water Level—Low Low Low, Level 1

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of the MSIVs and other interfaces with the reactor vessel occurs to prevent offsite dose limits
from being exceeded. The Reactor Vessel Water Level—Low Low Low, Level 1 Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. The Reactor Vessel Water Level—Low Low Low, Level 1 Function associated with isolation is assumed in the analysis of the recirculation line break (Ref. 1). The isolation of the MSLs on Level 1 supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four level instruments that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low Low Low, Level 1 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low Low Low, Level 1 Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding
regulatory ~~10 CFR 100~~ limits.
and control room

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.c. Main Steam Line Flow—High (continued)

directly assumed in the analysis of the main steam line break (MSLB) (Ref. 1). The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite and control room doses do not exceed ~~the 10 CFR 100~~ regulatory limits.

The MSL flow signals are initiated from 16 instruments that are connected to the four MSLs. The instruments are arranged such that, even though physically separated from each other, all four connected to one MSL would be able to detect the high flow. Four channels of Main Steam Line Flow—High Function for each unisolated MSL (two channels per trip system) are available and are required to be OPERABLE so that no single instrument failure will preclude detecting a break in any individual MSL.

1.d. Condenser Vacuum—Low

The Allowable Value is chosen to ensure that offsite dose limits are not exceeded due to the break.

The Condenser Vacuum—Low Function is provided to prevent overpressurization of the main condenser in the event of a loss of the main condenser vacuum. Since the integrity of the condenser is an assumption in offsite dose calculations, the Condenser Vacuum—Low Function is assumed to be OPERABLE and capable of initiating closure of the MSIVs. The closure of the MSIVs is initiated to prevent the addition of steam that would lead to additional condenser pressurization and possible rupture of the diaphragm installed to protect the turbine exhaust hood, thereby preventing a potential radiation leakage path following an accident.

Condenser vacuum pressure signals are derived from four pressure instruments that sense the pressure in the condenser. Four channels of Condenser Vacuum-Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.f. Manual Initiation (continued)

There are four push buttons for the logic, two manual initiation push button per trip system. There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons.

Two channels of Manual Initiation Function are available and are required to be OPERABLE in MODES 1, 2, and 3, since these are the MODES in which the MSL isolation automatic Functions are required to be OPERABLE.

Primary Containment Isolation

2.a. Reactor Vessel Water Level - Low, Level 3

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 3 supports actions to ensure that offsite dose limits ~~of 10 CFR 100~~ are not exceeded. The Reactor Vessel Water Level—Low, Level 3 Function associated with isolation is implicitly assumed in the FSAR analysis as these leakage paths are assumed to be isolated post LOCA.

and control room

regulatory

Reactor Vessel Water Level—Low, Level 3 signals are initiated from level instruments that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low, Level 3 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low, Level 3 Allowable Value was chosen to be the same as the RPS Level 3 scram Allowable Value (LCO 3.3.1.1), since isolation of these valves is not critical to orderly plant shutdown.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.b. Reactor Vessel Water Level—Low Low, Level 2

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 2 supports actions to ensure that offsite dose limits ~~of 10 CFR 100~~ are not exceeded. The Reactor Vessel Water Level—Low Low, Level 2 Function associated with isolation is implicitly assumed in the FSAR analysis as these leakage paths are assumed to be isolated post LOCA.

and control room

regulatory

Reactor Vessel Water Level—Low Low, Level 2 signals are initiated from level instruments that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low Low, Level 2 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low Low, Level 2 Allowable Value was chosen to be the same as the ECCS Level 2 Allowable Value (LCO 3.3.5.1), since this may be indicative of a LOCA.

2.c. Reactor Vessel Water Level—Low Low Low, Level 1

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 1 supports actions to ensure the offsite dose limits ~~of 10 CFR 100~~ are not exceeded. The Reactor Vessel Water Level - Low Low Low, Level 1 Function associated with isolation is implicitly assumed in the FSAR analysis as these leakage paths are assumed to be isolated post LOCA.

and control room

regulatory

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.c. Reactor Vessel Water Level—Low Low Low, Level 1 (continued)

Reactor vessel water level signals are initiated from four level instruments that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low Low Low, Level 1 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low Low Low, Level 1 Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the associated penetrations isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding ~~10 CFR 100~~ limits.

regulatory   and control room

2.d. Drywell Pressure—High

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite and control room

dose limits ~~of 10 CFR 100~~ are not exceeded. The Drywell Pressure—High Function, associated with isolation of the primary containment, is implicitly assumed in the FSAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

regulatory

High drywell pressure signals are initiated from pressure instruments that sense the pressure in the drywell. Four channels of Drywell Pressure—High per Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be the same as the ECCS Drywell Pressure—High Allowable Value (LCO 3.3.5.1), since this may be indicative of a LOCA inside primary containment.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

5.e. SLC System Initiation

The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 4). SLC System initiation signals are initiated from the two SLC pump start signals.

There is no Allowable Value associated with this Function since the channels are mechanically actuated based solely on the position of the SLC System initiation switch.

Two channels (one from each pump) of the SLC System Initiation Function are available and are required to be OPERABLE only in MODES ~~1 and 2, since these are the only MODES where the reactor can be critical, with the exception of Special Operations LCO 3.10.8, and these MODES are~~ consistent with the Applicability for the SLC System (LCO 3.1.7).

1, 2, and 3 which is

As noted (footnote (b) to Table 3.3.6.1-1), this Function is only required to close the outboard RWCU isolation valve trip systems.

5.f. Reactor Vessel Water Level—Low Low, Level 2

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some interfaces with the reactor vessel occurs to isolate the potential sources of a break. The isolation of the RWCU System on Level 2 supports actions to ensure that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. The Reactor Vessel Water Level—Low Low, Level 2 Function associated with RWCU isolation is not directly assumed in the FSAR safety analyses because the RWCU System line break is bounded by breaks of larger systems (recirculation and MSL breaks are more limiting).

Reactor Vessel Water Level—Low Low, Level 2 signals are initiated from four level instruments that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6.b. Reactor Vessel Water Level—Low, Level ~~3~~ (continued)

In MODES 1 and 2, another isolation (i.e., Reactor Steam Dome Pressure—High) and administrative controls ensure that this flow path remains isolated to prevent unexpected loss of inventory via this flow path.

6.c Manual Initiation

The Manual Initiation push button channels introduce signals to RHR Shutdown Cooling System isolation logic that is redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific FSAR safety analysis that takes credit for this Function. It is retained for overall redundancy and diversity of the isolation function as required by the NRC in the plant licensing basis.

There are two push buttons for the logic, one manual initiation push button per trip system. There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons.

Two channels of the Manual Initiation Function are available and are required to be OPERABLE in MODES 3, 4, and 5, since these are the MODES in which the RHR Shutdown Cooling System Isolation automatic Function are required to be OPERABLE.

Traversing Incore Probe System Isolation

7.a Reactor Vessel Water Level - Low, Level 3

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 3 supports actions to ensure that offsite dose limits ~~of 10 CFR 100~~ are not exceeded. The Reactor Vessel Water Level - Low, Level 3 Function associated with isolation is implicitly assumed in the FSAR analysis as these leakage paths are assumed to be isolated post LOCA.

and control room

regulatory

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7.a Reactor Vessel Water Level - Low, Level ~~3~~ (continued)

Reactor Vessel Water Level - Low, Level 3 signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Two channels of Reactor Vessel Water Level - Low, Level 3 Function are available and are required to be OPERABLE to ensure that no single instrument failure can initiate an inadvertent isolation actuation. The isolation function is ensured by the manual shear valve in each penetration.

The Reactor Vessel Water Level - Low, Level 3 Allowable Value was chosen to be the same as the RPS Level 3 scram Allowable Value (LCO 3.3.1.1), since isolation of these valves is not critical to orderly plant shutdown.

~~7.b. Drywell Pressure - High~~

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits ~~of 10 CFR 100~~ are not exceeded. The Drywell Pressure - High Function, associated with isolation of the primary containment, is implicitly assumed in the FSAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

and control room

regulatory

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Two channels of Drywell Pressure - High per Function are available and are required to be OPERABLE to ensure that no single instrument failure can initiate an inadvertent actuation. The isolation function is ensured by the manual shear valve in each penetration.

The Allowable Value was selected to be the same as the ECCS Drywell Pressure - High Allowable Value (LCO 3.3.5.1), since this may be indicative of a LOCA inside primary containment.

(continued)

BASES

BACKGROUND (continued)	system initiates isolation of one automatic isolation valve (damper) and starts one SGT subsystem (including its associated reactor building recirculation subsystem) while the other trip system initiates isolation of the other automatic isolation valve in the penetration and starts the other SGT subsystem (including its associated reactor building recirculation subsystem). Each logic closes one of the two valves on each penetration and starts one SGT subsystem, so that operation of either logic isolates the secondary containment and provides for the necessary filtration of fission products.
APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY	The isolation signals generated by the secondary containment isolation instrumentation are implicitly assumed in the safety analyses of References 1 and 2 to initiate closure of valves and start the SGT System to limit offsite doses. and control room

Refer to LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," Applicable Safety Analyses Bases for more detail of the safety analyses.

The secondary containment isolation instrumentation satisfies Criterion 3 of the NRC Policy Statement. (Ref. 7) Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the secondary containment isolation instrumentation is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have the required number of OPERABLE channels with their setpoints set within the specified Allowable Values, as shown in Table 3.3.6.2-1. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. Each channel must also respond within its assumed response time, where appropriate.

Allowable Values are specified for each Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Reactor Vessel Water Level—Low Low, Level 2 (continued)

level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low Low, Level 2 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low Low, Level 2 Allowable Value was chosen to be the same as the High Pressure Coolant Injection/Reactor Core Isolation Cooling (HPCI/RCIC) Reactor Vessel Water Level—Low Low, Level 2 Allowable Value (LCO 3.3.5.1 and LCO 3.3.5.2), since this could indicate that the capability to cool the fuel is being threatened.

The Reactor Vessel Water Level—Low Low, Level 2 Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the Reactor Coolant System (RCS); thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, this Function is not required. In addition, the Function is also required to be OPERABLE during operations with a potential for draining the reactor vessel (OPDRVs) because the capability of isolating potential sources of leakage must be provided to ensure that offsite and control room dose limits are not exceeded if core damage occurs.

Reactor Vessel Water Level–Low Low, Level 2 will isolate the affected Unit's zone (i.e., Zone I for Unit 1 and Zone II for Unit 2) and Zone III.

2. Drywell Pressure—High

High drywell pressure can indicate a break in the reactor coolant pressure boundary (RCPB). An isolation of the secondary containment and actuation of the SGT System are initiated in order to minimize the potential of an offsite dose release. The isolation on high drywell pressure supports actions to ensure that any offsite releases are within the limits calculated in the safety analysis. However, the Drywell Pressure—High Function associated with

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3, 4, 5, 6, 7 Refuel Floor High Exhaust Duct, Refuel Floor Wall Exhaust Duct, and Railroad Access Shaft Exhaust Duct Radiation—High (continued)

The Exhaust Radiation—High signals are initiated from radiation detectors that are located on the ventilation exhaust ductwork coming from the refueling floor zones and the Railroad Access Shaft. The signal from each detector is input to an individual monitor whose trip outputs are assigned to an isolation channel. Eight channels of Refuel Floor High Exhaust Duct and Wall Exhaust Duct Radiation—High Function (four from Unit 1 and four from Unit 2) and two channels of Railroad Access Shaft Exhaust Duct Radiation - High Function (both from Unit 1) are available to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding.

The Refuel Floor Exhaust Radiation—High Functions are required to be OPERABLE during CORE ALTERATIONS, OPDRVs, and movement of irradiated fuel assemblies in the secondary containment, because the capability of detecting radiation releases due to fuel failures (due to a fuel handling accident) must be provided to ensure that offsite dose limits are not exceeded. and control room

The Railroad Access Shaft Exhaust Duct Radiation - High Function is only required to be OPERABLE during handling of irradiated fuel within the Railroad Access Shaft, and above the Railroad Access Shaft with the Railroad Access Shaft Equipment Hatch open. This provides the capability of detecting radiation releases due to fuel failures resulting from dropped fuel assemblies which ensures that offsite dose limits are not exceeded. and control room

Refuel Floor High and Wall Exhaust Duct and Railroad Access Shaft Exhaust Duct Radiation - High Functions will isolate Zone III of secondary containment.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The ability of the CREOAS System to maintain the habitability of the main control room is explicitly assumed for certain accidents as discussed in the FSAR safety analyses (Refs. 1 and 2). CREOAS System operation ensures that the radiation exposure of control room personnel, through the duration of any one of the postulated accidents, does not exceed ~~the~~ regulatory limits ~~set by GDC 19 of 10 CFR 50, Appendix A.~~

CREOAS System instrumentation satisfies Criterion 3 of the NRC Policy Statement. (Ref. 5)

The OPERABILITY of the CREOAS System instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.7.1-1. Each Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Values, where appropriate. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

Allowable Values are specified for each CREOAS System Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter reaches the setpoint, the associated device changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3, 4, 5, 6, 7 Refuel Floor High Exhaust Duct, Refuel Floor Wall Exhaust Duct and Railroad Access Shaft Exhaust Duct Radiation—High (continued)

Duct Radiation—High Function (four from Unit 1 and four from Unit 2), and two channels of the Railroad Access Shaft Exhaust Radiation - High Function (both from Unit 1) are available and are required to be OPERABLE when the associated Refuel Floor Exhaust System is in operation to ensure that no single instrument failure can preclude the initiation function.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding. The Refuel Floor Exhaust Duct and Wall Exhaust Duct Radiation—High are required to be OPERABLE during CORE ALTERATIONS, OPDRVs, and movement of irradiated fuel assemblies in the secondary containment, because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded. and control room

The Railroad Access Shaft Exhaust Duct Radiation - High Function is only require to be OPERABLE during handling of irradiated fuel within the Railroad Access Shaft, and above the Railroad Access Shaft with the Railroad Access Shaft Equipment Hatch open, because the capability of detecting radiation releases due to fuel failures (dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded. and control room

8. Main Control Room Outside Air Intake Radiation—High

The main control room outside air intake radiation monitors measure radiation levels at the control structure outside air intake duct. A high radiation level may pose a threat to main control room personnel; thus, automatically initiating the CREOAS System. The Control Room Air Inlet Radiation—High Function consists of two independent monitors. Two channels of Control Room Air Inlet Radiation—High are available and are required to be OPERABLE to ensure that no single instrument failure can preclude CREOAS System initiation. The Allowable Value was selected to ensure protection of the control room personnel.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Specific Activity

BASES

BACKGROUND

During circulation, the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks into the reactor coolant and activation of corrosion products in the reactor coolant. These radioactive materials in the reactor coolant can plate out in the RCS, and, at times, an accumulation will break away to spike the normal level of radioactivity. The release of coolant during a Design Basis Accident (DBA) could send radioactive materials into the environment.

Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure that in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within ~~the~~ regulatory limits ~~of 10 CFR 100 (Ref. 1)~~.

This LCO contains iodine specific activity limits. The iodine isotopic activities per gram of reactor coolant are expressed in terms of a DOSE EQUIVALENT I-131. The allowable levels are intended to limit the 2 hour radiation dose to an individual at the site boundary to ~~a small fraction of the 10 CFR 100~~ regulatory limits.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving radioactive material in the primary coolant are presented in the FSAR (Ref. 2). The specific activity in the reactor coolant (the source term) is an initial condition for evaluation of the consequences of an accident due to a main steam line break (MSLB) outside containment. No fuel damage is postulated in the MSLB accident, and the release of radioactive material to the environment is assumed to end when the main steam isolation valves (MSIVs) close completely.

This MSLB release forms the basis for determining offsite and control room doses (Ref. 2). The limits on the specific activity of the primary coolant ensure that the 2 hour ~~thyroid and whole body~~ doses at the site boundary, resulting from an MSLB outside containment during steady state operation, will not exceed ~~40% of the dose guidelines of 10 CFR 100~~ regulatory limits.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limits on specific activity are values from a parametric evaluation of typical site locations. These limits are conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site.

RCS specific activity satisfies Criterion 2 of the NRC Policy Statement (Ref. 3).

LCO

The specific iodine activity is limited to ≤ 0.2 μCi/gm DOSE EQUIVALENT I-131. This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so any release of radioactivity to the environment during an MSLB ~~is less than a small fraction of the~~ regulatory ~~10 CFR 100~~ limits. will not exceed

APPLICABILITY

In MODE 1, and MODES 2 and 3 with any main steam line not isolated, limits on the primary coolant radioactivity are applicable since there is an escape path for release of radioactive material from the primary coolant to the environment in the event of an MSLB outside of primary containment

In MODES 2 and 3 with the main steam lines isolated, such limits do not apply since an escape path does not exist. In MODES 4 and 5, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.

ACTIONS

A.1 and A.2

When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is ≤ 4.0 μCi/gm, samples must be analyzed for DOSE EQUIVALENT I-131 at least once every 4 hours. In addition, the specific activity must be restored to the LCO limit within 48 hours. The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with the normal processing systems.

(continued)

and control room

BASES

APPLICABLE SAFETY ANALYSES (continued)

This MSLB release forms the basis for determining offsite doses (Ref. 2). The limits on the specific activity of the primary coolant ensure that the 2 hour ~~thyroid and whole body~~ dose~~s~~ at the site boundary, resulting from an MSLB outside containment during steady state operation, will not exceed ~~10% of the dose guidelines of 10 CFR 100~~.

regulatory limits

The limits on specific activity are values from a parametric evaluation of typical site locations. These limits are conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site.

RCS specific activity satisfies Criterion 2 of the NRC Policy Statement (Ref. 3).

LCO

The specific iodine activity is limited to ≤ 0.2 μCi/gm DOSE EQUIVALENT I-131. This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so any release of radioactivity to the environment during an MSLB ~~is less than a small fraction of the 10 CFR 100~~ limits.

regulatory

will not exceed

APPLICABILITY

In MODE 1, and MODES 2 and 3 with any main steam line not isolated, limits on the primary coolant radioactivity are applicable since there is an escape path for release of radioactive material from the primary coolant to the environment in the event of an MSLB outside of primary containment.

In MODES 2 and 3 with the main steam lines isolated, such limits do not apply since an escape path does not exist. In MODES 4 and 5, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

B.1, B.2.1, B.2.2.1, and B.2.2.2

If the DOSE EQUIVALENT I-131 cannot be restored to $\leq 0.2\ \mu Ci/gm$ within 48 hours, or if at any time it is $> 4.0\ \mu Ci/gm$, it must be determined at least once every 4 hours and all the main steam lines must be isolated within 12 hours. Isolating the main steam lines precludes the possibility of ~~releasing radioactive material to the environment in an amount that is more than a small fraction of the requirements of 10 CFR 100~~ exceeding regulatory dose limits during a postulated MSLB accident.

Alternatively, the plant can be placed in MODE 3 within 12 hours and in MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

The Completion Time of once every 4 hours is the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines in an orderly manner and without challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for placing the unit in MODES 3 and 4 are reasonable, based on operating experience, to achieve the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level.

This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.

REFERENCES

1. ~~10 CFR 100.11, 1973.~~ DELETED

2. FSAR, Section 15.6.4.

3. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

BASES (continued)

APPLICABLE SAFETY ANALYSES

The safety design basis for the primary containment is that it must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage.

Analytical methods and assumptions involving the primary containment are presented in References 1 and 2. The safety analyses assume a nonmechanistic fission product release following a DBA, which forms the basis for determination of offsite doses. The fission product release is, in turn, based on an assumed leakage rate from the primary containment. OPERABILITY of the primary containment ensures that the leakage rate assumed in the safety analyses is not exceeded.

and control room

The maximum allowable leakage rate for the primary containment (L_a) is 1.0% by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_a) of 45 psig.

Primary containment satisfies Criterion 3 of the NRC Policy Statement. (Ref. 6)

LCO

Primary containment OPERABILITY is maintained by limiting leakage to $\leq 1.0 \cdot L_a$, except prior to each startup after performing a required Primary Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.

Individual leakage rates specified for the primary containment air lock are addressed in LCO 3.6.1.2.

Leakage requirements for MSIVs and Secondary containment bypass are addressed in LCO 3.6.1.3.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.1 (continued)

to meet these SRs must be evaluated against the Type A, B, and C acceptance criteria of the Primary Containment Leakage Rate Testing Program. As left leakage prior to each startup after performing a required leakage test is required to be $< 0.6\ L_a$ for combined Type B and C leakage, and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

and control room

As noted in table B 3.6.1.3-1, an exemption to Appendix J is provided that isolation barriers which remain water filled or a water seal remains in the line post-LOCA are tested with water and the leakage is not included in the Type B and C $0.60\ L_a$ total.

SR 3.6.1.1.2

Maintaining the pressure suppression function of primary containment requires limiting the leakage from the drywell to the suppression chamber. Thus, if an event were to occur that pressurized the drywell, the steam would be directed through the downcomers into the suppression pool. This SR measures drywell to suppression chamber leakage to ensure that the leakage paths that would bypass the suppression pool are within allowable limits. The allowable limit is 10% of the acceptable SSES $A/\sqrt{k}$ design valve. For SSES, the $A/\sqrt{k}$ design value is .0535 ft^2.

Satisfactory performance of this SR can be achieved by establishing a known differential pressure between the drywell and the suppression chamber and determining the leakage. The leakage test is performed when the 10 CFR 50, Appendix J, Type A test is performed in accordance with the Primary Containment Leakage Rate Testing Program. This testing Frequency was developed considering this test is performed in conjunction with the Integrated Leak rate test and also in view of the fact that component failures that might have affected this test are identified by other primary containment SRs. Two consecutive test failures,

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The PCIVs LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory, and establishing the primary containment boundary during major accidents. As part of the primary containment boundary, PCIV OPERABILITY supports leak tightness of primary containment. Therefore, the safety analysis of any event requiring isolation of primary containment is applicable to this LCO.

The DBAs that result in a release of radioactive material within primary containment are a LOCA and a main steam line break (MSLB). In the analysis for each of these accidents, it is assumed that PCIVs are either closed or close within the required isolation times following event initiation. This ensures that potential paths to the environment through PCIVs (including primary containment purge valves) are minimized. ~~Of the events analyzed in Reference 1, the MSLB is the most limiting event due to radiological consequences.~~ The closure time of the main steam isolation valves (MSIVs) for a MSLB outside primary containment is a significant variable from a radiological standpoint. The MSIVs are required to close within 3 to 5 seconds since the 5 second closure time is assumed in the analysis. The safety analyses assume that the purge valves were closed at event initiation. Likewise, it is assumed that the primary containment is isolated such that release of fission products to the environment is controlled.

The DBA analysis assumes that within the required isolation time leakage is terminated, except for the maximum allowable leakage rate, L_a.

The single failure criterion required to be imposed in the conduct of unit safety analyses was considered in the original design of the primary containment purge valves. Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred.

The primary containment purge valves may be unable to close in the environment following a LOCA. Therefore, each of the purge valves is required to remain closed during MODES 1, 2, and 3 except as permitted under Note 2 of SR 3.6.1.3.1. In this case, the single failure criterion remains applicable to the primary containment purge valve

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.5 (continued)

full closure isolation time is demonstrated by SR 3.6.1.3.7. The isolation time test ensures that the valve will isolate in a time period less than or equal to that assumed in the Final Safety Analyses Report. The isolation time and Frequency of this SR are in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.6

For primary containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B, (Ref. 3), is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between primary containment and the environment), a Frequency of 184 days was established. The acceptance criteria for these valves is defined in the Primary Containment Leakage Rate Testing Program, 5.5.12.

The SR is modified by a Note stating that the primary containment purge valves are only required to meet leakage rate testing requirements in MODES 1, 2, and 3. If a LOCA inside primary containment occurs in these MODES, purge valve leakage must be minimized to ensure offsite radiological release is within limits. At other times when the purge valves are required to be capable of closing (e.g., during handling of irradiated fuel), pressurization concerns are not present and the purge valves are not required to meet any specific leakage criteria.

SR 3.6.1.3.7

Verifying that the isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The isolation time test ensures that the MSIV will isolate in a time period that does not exceed the times assumed in the DBA analyses. This ensures that the calculated radiological consequences of these events remain within ~~10 CFR 100~~

regulatory → limits. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.1.4 and SR 3.6.4.1.5

The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. To ensure that all fission products are treated, SR 3.6.4.1.4 verifies that the SGT System will rapidly establish and maintain a pressure in the secondary containment that is less than the pressure external to the secondary containment boundary. This is confirmed by demonstrating that one SGT subsystem will draw down the secondary containment to ≥ 0.25 inches of vacuum water gauge in less than or equal to the maximum time allowed. This cannot be accomplished if the secondary containment boundary is not intact. SR 3.6.4.1.5 demonstrates that one SGT subsystem can maintain ≥ 0.25 inches of vacuum water gauge for at least 1 hour at less than or equal to the maximum flow rate permitted for the secondary containment configuration that is operable. The 1 hour test period allows secondary containment to be in thermal equilibrium at steady state conditions. As noted, both SR 3.6.4.1.4 and SR 3.6.4.1.5 acceptance limits are dependent upon the secondary containment configuration when testing is being performed. The acceptance criteria for the SRs based on secondary containment configuration is defined as follows:

SECONDARY CONTAINMENT TEST CONFIGURATION	MAXIMUM DRAWDOWN TIME(SEC) (SR 3.6.4.1.4 ACCEPTANCE CRITERIA)	MAXIMUM FLOW RATE (CFM) (SR 3.6.4.1.5 ACCEPTANCE CRITERIA)
Zones I, II and III.	300 ≤ ~~425~~ Seconds (Zones I, II, and III)	≤ 4000 CFM (From Zones I, II, and III)
Zones I and III.	300 ≤ ~~147~~ Seconds (Zones I and III)	≤ 2885 CFM (From Zones I and III)

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.3 Control Room Emergency Outside Air Supply (CRE0AS) System

BASES

BACKGROUND

The CREOAS System provides a protected~~radiologically controlled~~ environment from which operators can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or smoke from outside the control room envelope.~~can be safely operated following a Design Basis Accident (DBA).~~ This ~~radiologically controlled~~ protected environment is termed the~~e~~ habitability envelope and is comprised of Control Structure floor elevations 697'-0" through 783'-0" including the stairwells as described in FSAR Section 6.4 (Ref. 1).

The safety related function of CREOAS System includes two independent and redundant high efficiency air filtration subsystems for emergency treatment of outside supply air and a control room boundary which limits the inleakage of unfiltered air. Each subsystem consists of an electric heater, a prefilter, an upstream high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section, a downstream HEPA filter, a CREOAS fan, a control structure heating and ventilation fan, a control room floor cooling fan, a computer room floor cooling fan, and the associated ductwork ~~and~~ dampers, doors, barriers and instrumentation. Prefilters and HEPA filters remove particulate matter, which may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay. With the exception of the CREOAS fan, all other CREOAS subsystem fans operate continuously to maintain the affected compartments environment. These other ventilation fans operate independently of the CREOAS fans and are required to operate to ensure a positive pressure in the control structure is maintained utilizing filtered outside air supplied by the CREOAS fans.

The habitability envelope is protected for normal operation, natural events, and accident conditions. The habitability envelope is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the habitability envelope. The integrity of the habitability envelope must be maintained to limit the inleakage of unfiltered air into the habitability envelope. The habitability envelope and habitability boundary are defined in the Control Room Habitability Program.

Upon receipt of the initiation signal(s) (indicative of conditions that could result in radiation exposure to occupants~~plant operators~~ in the habitability envelope), the CREOAS System automatically switches to the pressurization/filtration mode of operation to prevent infiltration of contaminated air into the habitability envelope. A system of dampers aligns

(continued)

BASES

the outside air intake to the CREOAS fan suction and filter train. Outside air is taken in at the normal ventilation intake and passed through one of the charcoal adsorber filter subsystems. The filtered air leaving the CREOAS filtration train is routed to the inlet of the other ventilation fans for distribution.

One of the CREOAS System design requirements is to maintain the ~~control room~~habitability envelope environment for a 30 day continuous occupancy after a DBA without exceeding ~~5 rem~~ dose regulatory limits~~whole body dose or its equivalent to any part of the body~~. A single CREOAS subsystem with an intact control structure habitability envelope will pressurize the habitability envelope (which includes the control

(continued)

BASES

BACKGROUND (continued)

room) to greater than or equal to 0.125 inches water gauge to prevent infiltration of air from surrounding buildings. CREOAS System operation in maintaining the habitability envelope environment is discussed in the FSAR, Chapters 6 and 9, (Refs. 1 and 2, respectively)

APPLICABLE SAFETY ANALYSES

The ability of the CREOAS System to maintain the habitability of the control structure habitability envelope is an explicit assumption for the safety analyses presented in the FSAR, Chapters 6 and 15 (Refs. 1 and 3, respectively). The pressurization/ filtration mode of the CREOAS System is assumed to operate following a loss of coolant accident~~,~~ and fuel handling accident~~, and control rod drop accident~~, as discussed in the FSAR, Section 6.4.1 (Ref. 1). The radiological doses to occupants~~plant operators~~ in the habitability envelope as a result of the various DBAs are summarized in Reference 3. No single active failure will cause the loss of outside or recirculated air.

The CREOAS System satisfies Criterion 3 of the NRC Policy Statement. (Ref. 4)

LCO

Two redundant subsystems of the CREOAS System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disable the other subsystem. Total system failure or loss of habitability boundary integrity could result in exceeding a dose regulatory limits~~of 5 rem whole body or equivalent~~ to occupants~~plant operators~~ in the habitability envelope in the event of a DBA.

The CREOAS System is considered OPERABLE when the individual components necessary to limit~~control~~ operator exposure are OPERABLE~~ in both subsystems~~. Both subsystems are considered OPERABLE when:

a. Both filter trains each consisting of a CREOAS fan heater, a HEPA filter, and charcoal adsorber which is not excessively restricting flow is OPERABLE; and

b. Both Control Structure Heating and Ventilation fans, Computer Room Floor Cooling fans, and Control Room Floor Cooling fans are OPERABLE; and

c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

d. Neither Smoke Removal Fan (0V104A/B) is in operation.

(continued)

BASES

LCO (continued)

One subsystem is considered OPERABLE when:

a. One filter train consisting of a CREOAS fan, heater, a HEPA filter, and charcoal adsorber which is not excessively restricting flow is OPERABLE; and

b. The 'A' Control Structure Heating and Ventilation fan (0V103A) and the 'A' Computer Room Floor Cooling fan (0V115A) and the 'A' Control Room Floor Cooling fan (0V117A) are OPERABLE

OR

The 'B' Control Structure Heating and Ventilaiton fan (0V103B) and the 'B' Computer Room Floor Cooling fan (0V115B) and the 'B' Control Room Floor Cooling fan (0V117B) are OPERABLE

(These fans are not dedicated to either CREOAS subsystem. As a result when any one set of fans is not OPERABLE, one arbitrarily determined CREOAS subsystem is not OPERABLE): and

c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

d. Neither Smoke Removal Fan (0V104A/B) is in operation.

~~In addition, the habitability envelope must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors to maintain a positive pressure.~~ Note the habitability envelope can not be maintained with a smoke removal fan (0V104A or 0V104B) in operation. In order for the CREOAS subsystems to be considered OPERABLE, the integrity of the habitability boundary must be maintained such that control room occupant dose from a large radioactive release does not exceed regulatory limits and the control room occupants are protected from hazardous chemicals and smoke from outside the habitability boundary.

The LCO is modified by a Note allowing the control room habitability envelope boundary to be opened intermittently under administrative controls. This Note only applies to openings in the habitability envelope that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the control room operators. This individual

(continued)

BASES

will have a method to rapidly close the opening and to restore the habitability envelope to a condition equivalent to the design condition when a need for control room habitability envelope isolation is indicated.

APPLICABILITY

In MODES 1, 2, and 3, the CREOAS System must be OPERABLE to ensure that the habitability envelope will remain habitable ~~control operator exposure~~ during and following a DBA, since the DBA could lead to a fission product release.

APPLICABILITY (continued)

In MODES 4 and 5, the probability and consequences of a DBA are reduced because of the pressure and temperature limitations in these MODES. Therefore, maintaining the CREOAS System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During operations with potential for draining the reactor vessel (OPDRVs);

b. During CORE ALTERATIONS; and

c. During movement of irradiated fuel assemblies in the secondary containment.

ACTIONS

A.1

With one CREOAS subsystem inoperable for reasons other than an inoperable habitability boundary, the inoperable CREOAS subsystem must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE CREOAS subsystem is adequate to perform the habitability envelope~~its radiation~~ protection function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced CREOAS System capability. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and that the remaining subsystem can provide the required capabilities.

B.1 and B.2

If the unfiltered inleakage of potentially contaminated air past the habitability boundary and into the habitability envelope can result in occupants of the habitability envelope receiving doses greater than regulatory limits or the control room occupants are not protected from hazardous chemicals or

(continued)

BASES

smoke from outside the habitability boundary, the habitability boundary is inoperable. However, the control room habitability boundary may be considered OPERABLE, but degraded or nonconforming, when unfiltered air inleakage is greater than assumed in the licensing basis accident analyses if compensatory measures can ensure that the habitability envelope remains habitable for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section 2.7.3, (Ref. 5) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 6). Temporary analytical methods may also be used as compensatory measures (Ref. 7). The Control Room Habitability Program provides limits on use of compensatory measures to maintain the control room habitability boundary operable but degraded or nonconforming when the unfiltered air inleakage is greater than that assumed in the licensing basis analyses. When those limits are exceeded, Condition B must be entered. Actions must be taken to restore an OPERABLE habitability boundary within 24 hours. During the period that the habitability boundary is inoperable, mitigating actions must be implemented to lessen the effect to control room occupants from the potential hazards of a radiological or chemical event or a challenge from smoke exhaust to the habitability envelope. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable habitability boundary) should be preplanned for implementation upon entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. ~~If the control room habitability envelope boundary is inoperable in MODES 1, 2, and 3, the CREOAS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE control room habitability envelope boundary within 24 hours. During the period that the control room habitability envelope boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the Condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the control room habitability envelope boundary.~~

ACTIONS (continued)

C.1 and C.2

In MODE 1, 2, or 3, if the inoperable CREOAS subsystem or control room habitability envelope boundary cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status, the unit must be placed in at

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least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1, D.2.1, D.2.2, and D.2.3

The Required Actions of Condition D are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require either an entry into LCO 3.0.3 or a reactor shutdown in accordance with LCO 3.0.3.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, if the inoperable CREOAS subsystem cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CREOAS subsystem may be placed in the pressurization/filtration mode. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the habitability envelope~~control room~~. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

ACTIONS (continued)

E.1

If both CREOAS subsystems are inoperable in MODE 1, 2, or 3, for reasons other than an inoperable control room habitability envelope boundary (i. e., Condition B) the CREOAS System may not be capable of performing the intended function and the unit is in a condition outside the

(continued)

BASES

accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

F.1, F.2, and F.3

The Required Actions of Condition F are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require either an entry into LCO 3.0.3 or a reactor shutdown in accordance with LCO 3.0.3.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, with two CREOAS subsystems inoperable or if the habitability boundary cannot be restored to OPERABLE status within the required Completion Time, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require pressurization of the habitability envelope. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately Suspension of these activities shall not preclude completion of movement of a component to a safe position. If applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that a CREOAS fan in a standby mode starts on demand from the control room and continues to operate with flow through the HEPA filters and charcoal adsorbers. Standby systems should be checked periodically to ensure that they start and function properly. As the environmental and normal operating conditions of this system are not severe, testing each subsystem once every month provides an adequate check on this system. Monthly heater operation dries out any moisture that

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1 (continued)

has accumulated in the charcoal as a result of humidity in the ambient air. Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Furthermore, the 31 day Frequency is based on the known reliability of the equipment and the availability of two redundant

(continued)

BASES

subsystems.

SR 3.7.3.2

This SR verifies that the required CREOAS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.3.3

This SR verifies that on an actual or simulated initiation signal, each CREOAS subsystem starts and operates. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.7.1.5 overlaps this SR to provide complete testing of the safety function. The 24 month Frequency is consistent with industry practice and other filtration systems SRs.

SR 3.7.3.4

This SR verifies the integrity of the habitability boundary by testing for unfiltered air inleakage past the habitability boundary and into the habitability envelope. The details of the testing are specified in the Control Room Habitability Program.

Unfiltered air inleakage through the habitability boundary and into the habitability envelope greater than the amount assumed in the licensing basis accident analyses results in the control room habitability boundary being inoperable when habitability is not maintained (i.e., accident dose is greater than regulatory limits or the occupants are not protected from hazardous chemical and smoke.) However, the control room habitability boundary may be considered OPERABLE, but degraded or nonconforming, when unfiltered air inleakage is greater than assumed in the licensing basis accident analyses if compensatory measures can ensure that the habitability envelope remains habitable for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section 2.7.3, (Ref. 5) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 6). Temporary analytical methods may also be used as compensatory measures (Ref. 7). The Control Room Habitability Program provides limits on use of compensatory measures to maintain the control room habitability boundary operable but degraded or nonconforming when the unfiltered air inleakage is greater than that assumed in the licensing

(continued)

BASES

basis analyses. When those limits are exceeded, Condition B must be entered. ~~This SR verifies the integrity of the habitability envelope and the assumed inleakage rates of potentially contaminated air. The habitability envelope positive pressure, with respect to potentially contaminated adjacent areas (the turbine building), is periodically tested to verify proper function of the CREOAS System and the integrity of the habitability envelope. During the emergency mode of operation, the CREOAS System is designed to slightly pressurize the control structure ≥ 0.125 inches water gauge positive pressure with respect to the outside atmosphere to prevent unfiltered inleakage. The CREOAS System is designed to maintain this positive pressure at a flow rate of ≤ 5810 cfm to the control structure in the pressurization/filtration mode. The control structure habitability envelope is maintained when the control structure habitability envelope can be pressurized to ≥0.125 inches water gage positive pressure with respect to outside atmosphere. The Frequency of 24 months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration systems SRs.~~

(continued)

BASES

REFERENCES

1. FSAR, Chapter 6.
2. FSAR, Chapter 9.
3. FSAR, Chapter 15.
4. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).
5. Regulatory Guide 1.196, May, 2003.
6. NEI 99-03, "Control Room Habitability Assessment," March 2003.
7. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2005, "NEI Draft White Paper, use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability."

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Condenser Offgas

BASES

BACKGROUND

During unit operation, steam from the low pressure turbine is exhausted directly into the condenser. Air and noncondensible gases are collected in the condenser, then exhausted through the steam jet air ejectors (SJAEs) to the Main Condenser Offgas System. The offgas from the main condenser normally includes radioactive gases.

The Main Condenser Offgas System has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled by the offgas condenser; the water and condensibles are stripped out by the offgas condenser and moisture separator. The radioactivity of the remaining gaseous mixture (i.e., the offgas recombiner effluent) is monitored downstream of the moisture separator prior to entering the holdup line.

APPLICABLE SAFETY ANALYSES

The main condenser offgas radioactivity rate is an initial condition of the Main Condenser Offgas System failure event, discussed in the FSAR, Section 15.7.1 (Ref. 1). The analysis assumes a gross failure in the Main Condenser Offgas System that results in the rupture of the Main Condenser Offgas System pressure boundary. The radioactivity rate of the specified noble gases (Xe-133, Xe-135, Xe-138, Kr-85m, Kr-87, and Kr-88) is controlled to ensure that, during the event, the calculated offsite doses will be well within ~~the~~ limits ~~of 10 CFR 100 (Ref. 2)~~ or the NRC staff approved regulatory licensing basis.

The main condenser offgas limits satisfy Criterion 2 of the NRC Policy Statement. (Ref. 3)

LCO

To ensure compliance with the assumptions of the Main Condenser Offgas System failure event (Ref.1), the fission product release rate should be consistent with a specified noble gas release to the reactor coolant of

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B 3.7 PLANT SYSTEMS

B 3.7.7 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool design is found in the FSAR, Section 9.1 (Ref. 1). The assumptions of the fuel handling accident are found in the FSAR, Section 15.7.4 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the radiological consequences (calculated ~~whole body and thyroid~~ doses at the exclusion area and low population zone boundaries) are ~~≤ 25% of 10 CFR 100 (Ref. 4) exposure guidelines NUREG-0800 (Ref. 3~~) within the regulatory limits of 10 CFR 50.67 (Ref. 4). A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide ~~1.25~~ 1.183 (Ref. 5).

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. With an assumed minimum water level of 21 ft and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and that offsite and control room doses are maintained within allowable limits (Ref. 2). The consequences of a fuel handling accident over the spent fuel storage pool are no more severe than those of the fuel handling accident over the reactor core, as discussed in the FSAR, Section 15.7.4 (Ref. 2). The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The spent fuel storage pool water level satisfies Criteria 2 and 3 of the NRC Policy Statement (Ref. 6).

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 9.1.
2. FSAR, Section 15.7.4.
3. ~~NUREG-0800, Section 15.7.4, Revision 1, July 1981.~~ DELETED
4. 10 CFR ~~100.~~ 50.67
5. Regulatory Guide ~~1.25, March 1972.~~ 1.183, July 2000
6. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

B 3.9 REFUELING OPERATIONS

B 3.9.6 Reactor Pressure Vessel (RPV) Water Level

BASES

BACKGROUND

The movement of fuel assemblies or handling of control rods within the RPV requires a minimum water level of 22 ft above the top of the RPV flange. During refueling, this maintains a sufficient water level in the reactor vessel cavity and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to ≤ 25% of 10 CFR ~~100~~50.67 limits, as provided by the guidance of Reference ~~3~~1.

APPLICABLE SAFETY ANALYSES

During movement of fuel assemblies or handling of control rods, the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by Regulatory Guide 1.~~25~~183 (Ref. 1). A decontamination factor of ~~100~~38 ~~(Regulatory Position C.1.g of Ref. 1)~~ is used in the accident analysis for iodine. This relates to the assumption that 99.3% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the water. The fuel pellet to cladding gap is assumed to contain ~~10~~8% of the total fuel rod ~~iodine~~ I-131inventory and 5% of the total fuel rod I-132, I-133, I-134, and I-135 inventory (Ref. 1).

Analysis of the fuel handling accident inside containment is described in Reference 2. With an assumed minimum water level of 21 ft and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and that offsite and control room doses are maintained within allowable limits (Ref. 2).

While the worst case assumptions include the dropping of the irradiated fuel assembly being handled onto the reactor core, the possibility exists of the dropped assembly striking the RPV flange and releasing fission products. Therefore, the minimum depth for water coverage to ensure

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BASES

APPLICABLE SAFETY ANALYSES (continued)

acceptable radiological consequences is specified from the RPV flange. Since the worst case event results in failed fuel assemblies seated in the core, as well as the dropped assembly, dropping an assembly on the RPV flange will result in reduced releases of fission gases.

RPV water level satisfies Criterion 2 of the NRC Policy Statement (Ref. 5).

LCO

A minimum water level of 22 ft above the top of the RPV flange is required to ensure that the radiological consequences of a postulated fuel handling accident are within acceptable limits, as provided by the guidance of Reference ~~8~~. 1

APPLICABILITY

LCO 3.9.6 is applicable when moving fuel assemblies or handling control rods (i.e., movement with other than the normal control rod drive) within the RPV. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. Requirements for fuel handling accidents in the spent fuel storage pool are covered by LCO 3.7.7, "Spent Fuel Storage Pool Water Level."

ACTIONS

A.1

If the water level is < 22 ft above the top of the RPV flange, all operations involving movement of fuel assemblies and handling control rods within the RPV shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of fuel movement and handling control rods shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 22 ft above the top of the RPV flange ensures that the design basis for the

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.9.6.1

postulated fuel handling accident analysis during refueling operations is met. Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide ~~1.25, March 23, 1972.~~ 1.183, July 2000
2. FSAR, Section 15.7.4.
3. ~~NUREG-0800, Section 15.7.4.~~ deleted.
4. 10 CFR ~~100.11.~~ 50.67
5. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.3 Reactor Vessel Water Level (continued)

The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of regulatory ~~10 CFR 100, "Reactor Site Criteria,"~~ limits ~~(Ref. 3)~~. Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. ANFB 524 (P)(A), Revision 2, "Critical Power Methodology for Boiling Water Reactors," Supplement 1 Revision 2 and Supplement 2, November 1990.
3. ~~10 CFR 100.~~ DELETED
4. EMF-1997(P)(A), Revision 0, "ANFB-10 Critical Power Correlation," July 1998 and EMF-1997(P)(A) Supplement 1 Revision 0," ANFB-10 Critical Power Correlation: High Local Peaking Results," July 1998.
5. EMF-2158(P)(A), Rev. 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4 / MICROBURN-B2," October 1999.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and anticipated operational occurrences (AOOs).

During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." Following inception of unit operation, RCS components shall be pressure tested in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB, reducing the number of protective barriers designed to prevent radioactive releases from exceeding the regulatory limits ~~specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4)~~. If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor High Flux and Vessel Steam Dome Pressure—High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

(continued)

APPLICABLE SAFETY ANALYSES (continued)

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME, Boiler and Pressure Vessel Code, 1968 Edition, including Addenda through the summer of 1970 (Ref. 5), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The SL of 1325 psig, as measured in the reactor steam dome, is equivalent to 1375 psig at the lowest elevation of the RCS. The RCS inside containment is designed to the ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition with Addenda through summer of 1972 (Ref. 6), for the reactor recirculation piping, which permits a maximum pressure transient of 110% of design pressures of 1250 psig for suction piping and 1500 psig for discharge piping. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 110% of design pressures of 1250 psig for suction piping and 1500 psig for discharge piping. The most limiting of these allowances is the 110% of the suction piping design pressures; therefore, the SL on maximum allowable RCS pressure is established at 1325 psig as measured at the reactor steam dome.

APPLICABILITY

SL 2.1.2 applies in all MODES.

SAFETY LIMIT VIOLATIONS

regulatory

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of ~~10 CFR 100, "Reactor Site Criteria,"~~ limits ~~(Ref. 4)~~. Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also assures that the probability of an accident occurring during this period is minimal.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI,Article IW-5000.
4. ~~10 CFR 100.~~ DELETED
5. ASME, Boiler and Pressure Vessel Code, Section III,1968 Edition, Addenda summer of 1970.
6. ASME, Boiler and Pressure Vessel Code, Section III,1971 Edition, Addenda summer of 1972.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

The SLC System is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. Insert 1 The SLC system satisfies the requirements of 10 CFR 50.62 (Ref. 1) for anticipated transient without scram.

The SLC System consists of a sodium pentaborate solution storage tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core. A smaller tank containing demineralized water is provided for testing purposes.

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods or if fuel damage occurs post-LOCA. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner or if fuel damage occurs post-LOCA. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron, which produces a concentration of 660 ppm of natural boron, in the reactor coolant at 68°F. To allow for potential leakage and imperfect mixing in the reactor system, an amount of boron equal to 25% of the amount cited above is added (Ref. 2). The volume versus concentration limits in Figure 3.1.7-1 and the temperature versus concentration limits in Figure 3.1.7-2 are calculated such that the required concentration is achieved accounting for dilution in the RPV with normal water level and including the water volume in

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

the residual heat removal shutdown cooling piping and in the recirculation loop piping. This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected. The minimum concentration of 13.6 weight percent ensures compliance with the requirements of 10 CFR 50.62 (Ref. 1).

INSERT 2

The SLC System satisfies the requirements of the NRC Policy Statement (Ref. 3) because operating experience and probabilistic risk assessments have shown the SLC System to be important to public health and safety. Thus, it is retained in the Technical Specifications.

LCO

The OPERABILITY of the SLC System provides backup capability for ~~reactivity control independent of normal reactivity control provisions provided by the control rods.~~ INSERT 3 The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE; each contains an OPERABLE pump, an explosive valve, and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, control rods are not able to be withdrawn (except as permitted by LCO 3.10.3 and LCO 3.10.4) since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to be OPERABLE when only a single control rod can be withdrawn.

INSERT 4

(continued)

BASES (continued)

ACTIONS

A.1

and suppression pool pH control

If the boron solution concentration is less than the required limits for compliance with 10 CFR 50.62 (Ref. 1) (≥ 13.6 weight percent) but greater than the concentration required for cold shutdown (original licensing basis), the concentration must be restored to within limits > 13.6 weight percent in 72 hours. It is not necessary under these conditions to enter Condition C for both SLC subsystems inoperable since they are capable of performing their original design basis function. Because of the low probability of an event and the fact that the SLC System capability still exists for vessel injection under these conditions, the allowed Completion Time of 72 hours is acceptable and provides adequate time to restore concentration to within limits.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of concentration out of limits or inoperable SLC subsystems during any single continuous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, an SLC subsystem is inoperable and that subsystem is subsequently returned to OPERABLE, the LCO may already have been not met for up to 7 days. This situation could lead to a total duration of 10 days (7 days in Condition B, followed by 3 days in Condition A), since initial failure of the LCO, to restore the SLC System. Then an SLC subsystem could be found inoperable again, and concentration could be restored to within limits. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock," resulting in establishing the "time zero" at the time the LCO was initially not met instead of at the time Condition A was entered. The 10 day Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

B.1

If one SLC subsystem is inoperable for reasons other than Condition A, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to perform the

(continued)

BASES

ACTIONS

B.1 (continued)

and provide adequate buffering agent to the suppression pool.

shutdown function. However, the overall reliability is reduced because a single failure in the remaining OPERABLE subsystem could result in reduced SLC System shutdown capability. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of performing the intended SLC System functions and the low probability of an event occurring ~~concurrent with the failure of the Control Rod Drive (CRD) System to shut down the plant.~~ requiring SLC injection.

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of concentration out of limits or inoperable SLC subsystems during any single continuous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, concentration is out of limits, and is subsequently returned to within limits, the LCO may already have been not met for up to 3 days. This situation could lead to a total duration of 10 days (3 days in Condition A, followed by 7 days in Condition B), since initial failure of the LCO, to restore the SLC System. Then concentration could be found out of limits again, and the SLC subsystem could be restored to OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock," resulting in establishing the "time zero" at the time the LCO was initially not met instead of at the time Condition B was entered. The 10 day Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

C.1

If both SLC subsystems are inoperable for reasons other than Condition A, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable given the low probability of an event occurring ~~concurrent with the failure of the control rods to shut down the reactor.~~ requiring SLC injection.

(continued)

BASES

ACTIONS (continued)

D.1

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and MODE 4 within 36 hours. The allowed Completion Time ~~of 12 hours is~~ are reasonable, based on operating experience, to reach ~~MODE 3 from~~ the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3

SR 3.1.7.1 through SR 3.1.7.3 are 24 hour Surveillances verifying certain characteristics of the SLC System (e.g., the volume and temperature of the borated solution in the storage tank), thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure that the proper borated solution volume and temperature, including the temperature of the pump suction piping, are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the sodium pentaborate remains in solution and does not precipitate out in the storage tank or in the pump suction piping. The temperature versus concentration curve of Figure 3.1.7-2 ensures that a 10°F margin will be maintained above the saturation temperature. An alternate method of performing SR 3.1.7.3 is to verify the OPERABILITY of the SLC heat trace system. This verifies the continuity of the heat trace lines and ensures proper heat trace operation, which ensure that the SLC suction piping temperature is maintained. The 24 hour Frequency is based on operating experience and has shown there are relatively slow variations in the measured parameters of volume and temperature.

SR 3.1.7.4 and SR 3.1.7.6

SR 3.1.7.4 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.7

Demonstrating that each SLC System pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1224 psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this Surveillance is in accordance with the Inservice Testing Program.

Insert 5

SR 3.1.7.8 and SR 3.1.7.9

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The pump and explosive valve tested should be alternated such that both complete flow paths are tested every 48 months at alternating 24 month intervals. The Surveillance may be performed in separate steps to prevent injecting solution into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Demonstrating that all heat traced piping between the boron solution storage tank and the suction inlet to the injection

(continued)

Insert 1: Additionally, the SLC System is designed to provide sufficient buffering agent to maintain the suppression pool pH at or above 7.0 following a DBA LOCA involving fuel damage. Maintaining the suppression pool pH at or above 7.0 will mitigate the re-evolution of iodine from the suppression pool water following a DBA LOCA.

Insert 2: The SLC system is also used to control Suppression Pool pH in the event of a DBA LOCA by injecting sodium pentaborate into the reactor vessel. The sodium pentaborate is then transported to the suppression pool and mixed by ECCS flow recirculation through the reactor, out of the break and into the suppression chamber. The amount of sodium pentaborate solution that must be available for injection following a DBA LOCA is determined as part of the DBA LOCA radiological analysis. This quantity is maintained in the storage tank as specified in the Technical Specification.

Insert 3: and provides sufficient buffering agent to maintain the suppression pool pH at or above 7.0 following a DBA LOCA involving fuel damage.

Insert 4: A DBA LOCA that results in the release of radioactive material is possible in MODES 1, 2 and 3 therefore capability to buffer the suppression pool pH is required. In MODES 4 and 5 a DBA LOCA with a radioactive release need not be postulated.

Insert 5: Additionally, the minimum pump flow rate requirement ensures that adequate buffering agent will reach the suppression pool to maintain pH above 7.0.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV is a volume of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two SDVs (headers) and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. The two instrument volumes are connected to a common drain line with two valves in series. Each header is connected to a common vent line with two valves in series. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 1.

APPLICABLE SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all of the control rods are capable of scramming. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite and control room doses remain within ~~the~~ regulatory limits ~~of 10 CFR 100 (Ref. 2)~~; and

b. Open on scram reset to maintain the SDV vent and drain path open so that there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite and control room doses are well within ~~the~~ regulatory limits ~~of 10 CFR 100 (Ref. 2)~~, and adequate core cooling is maintained (Ref. 3). The SDV vent and drain valves allow continuous drainage of the SDV during normal plant operation

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.3 (continued)

30 seconds after receipt of a scram signal is based on the bounding leakage case evaluated in the accident analysis based on the requirements of Reference 2. Similarly, after receipt of a simulated or actual scram reset signal, the opening of the SDV vent and drain valves is verified. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1 and the scram time testing of control rods in LCO 3.1.3 overlap this Surveillance to provide complete testing of the assumed safety function. The 24 month Frequency is based on the need to perform portions of this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 4.6.
2. ~~10 CFR 100.~~ DELETED
3. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981.
4. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation. Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during the normal operations identified in Reference 1.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 1, 2, 3, and 4. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of regulatory limits ~~the guidelines of 10 CFR, Parts 20, 50, and 100~~. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet; and

b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 3).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. A separate evaluation was performed to determine the limits of LHGR during anticipated operational occurrences. This limit, Protection Against Power Transients (PAPT), defined in reference 4, provides the acceptance criteria for LHGRs calculated in evaluation of the AOOs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

The penetrations which are isolated by the below listed functions can be determined by referring to the PCIV Table found in the Bases of LCO 3.6.1.3, "Primary Containment Isolation Valves."

<u>Main Steam Line Isolation</u>

<u>1.a. Reactor Vessel Water Level—Low Low Low, Level 1</u>

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of the MSIVs and other interfaces with the reactor vessel occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level—Low Low Low, Level 1 Function is one of the many Functions ~~assumed to be OPERABLE and capable of~~ providing isolation signals. The Reactor Vessel Water Level—Low Low Low, Level 1 Function associated with isolation is assumed in the analysis of the recirculation line break (Ref. 1). The isolation of the MSLs on Level 1 supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four level instruments that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low Low Low, Level 1 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low Low Low, Level 1 Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite and control room doses from exceeding regulatory ~~10 CFR 100~~ limits.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.c. Main Steam Line Flow—High (continued)

and control room

cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed ~~the 10 CFR 100~~ limits.

regulatory

The MSL flow signals are initiated from 16 instruments that are connected to the four MSLs. The instruments are arranged such that, even though physically separated from each other, all four connected to one MSL would be able to detect the high flow. Four channels of Main Steam Line Flow—High Function for each unisolated MSL (two channels per trip system) are available and are required to be OPERABLE so that no single instrument failure will preclude detecting a break in any individual MSL.

1.d. Condenser Vacuum—Low

The Allowable Value is chosen to ensure that offsite dose limits are not exceeded due to the break.

The Condenser Vacuum—Low Function is provided to prevent overpressurization of the main condenser in the event of a loss of the main condenser vacuum. Since the integrity of the condenser is an assumption in offsite dose calculations, the Condenser Vacuum—Low Function is assumed to be OPERABLE and capable of initiating closure of the MSIVs. The closure of the MSIVs is initiated to prevent the addition of steam that would lead to additional condenser pressurization and possible rupture of the diaphragm installed to protect the turbine exhaust hood, thereby preventing a potential radiation leakage path following an accident.

Condenser vacuum pressure signals are derived from four pressure instruments that sense the pressure in the condenser. Four channels of Condenser Vacuum—Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is chosen to prevent damage to the condenser due to pressurization, thereby ensuring its integrity for offsite dose analysis. As noted (footnote (a) to Table 3.3.6.1-1), the channels are not required to be OPERABLE in MODES 2 and 3 when all main turbine stop valves (TSVs) are closed, since the potential for condenser overpressurization is minimized. Switches are provided to manually bypass the channels when all TSVs are closed.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Primary Containment Isolation

2.a. Reactor Vessel Water Level - Low, Level 3 and control room

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 3 supports actions to ensure that offsite dose limits ~~of 10 CFR 100~~ are not exceeded. The Reactor Vessel Water Level—Low, Level 3 Function associated with isolation is implicitly assumed in the FSAR analysis as these leakage paths are assumed to be isolated post LOCA. regulatory

Reactor Vessel Water Level—Low, Level 3 signals are initiated from level instruments that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low, Level 3 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low, Level 3 Allowable Value was chosen to be the same as the RPS Level 3 scram Allowable Value (LCO 3.3.1.1), since isolation of these valves is not critical to orderly plant shutdown.

2.b. Reactor Vessel Water Level—Low Low, Level 2 and control room

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 2 supports actions to ensure that offsite dose limits ~~of 10 CFR 100~~ are not exceeded. The Reactor Vessel Water Level— Low Low, Level 2 Function associated with isolation is implicitly assumed in the FSAR analysis as these leakage paths are assumed to be isolated post LOCA. regulatory

Reactor Vessel Water Level—Low Low, Level 2 signals are initiated from level instruments that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low Low, Level 2 Function are available and

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Reactor Vessel Water Level - Low Low, Level 2 (continued)

are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low Low, Level 2 Allowable Value was chosen to be the same as the ECCS Level 2 Allowable Value (LCO 3.3.5.1), since this may be indicative of a LOCA.

2.c. Reactor Vessel Water Level—Low Low Low, Level 1

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 1 supports actions to ensure the offsite dose regulatory limits ~~of 10 CFR 100~~ are not exceeded. The Reactor Vessel Water Level - Low Low Low, Level 1 Function associated with isolation is implicitly assumed in the FSAR analysis as these leakage paths are assumed to be isolated post LOCA. and control room

Reactor vessel water level signals are initiated from four level instruments that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level-Low Low Low, Level 1 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low Low Low, Level 1 Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the associated penetrations isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding ~~10 CFR 100~~ limits.

regulatory

and control room

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.d. Drywell Pressure—High

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite and control room dose regulatory limits ~~of 10 CFR 100~~ are not exceeded. The Drywell Pressure—High Function, associated with isolation of the primary containment, is implicitly assumed in the FSAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

High drywell pressure signals are initiated from pressure instruments that sense the pressure in the drywell. Four channels of Drywell Pressure—High per Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be the same as the ECCS Drywell Pressure—High Allowable Value (LCO 3.3.5.1), since this may be indicative of a LOCA inside primary containment.

2.e. SGTS Exhaust Radiation—High

High SGTS Exhaust radiation indicates possible gross failure of the fuel cladding. Therefore, when SGTS Exhaust Radiation High is detected, an isolation is initiated to limit the release of fission products. However, this Function is not assumed in any accident or transient analysis in the FSAR because other leakage paths (e.g., MSIVs) are more limiting.

The SGTS Exhaust radiation signals are initiated from radiation detectors that are located in the SGTS Exhaust. Two channels of SGTS Exhaust Radiation—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is low enough to promptly detect gross failures in the fuel cladding.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

5.e. SLC System Initiation

The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 4). SLC System initiation signals are initiated from the two SLC pump start signals.

There is no Allowable Value associated with this Function since the channels are mechanically actuated based solely on the position of the SLC System initiation switch.

Two channels (one from each pump) of the SLC System Initiation Function are available and are required to be OPERABLE only in MODES ~~1 and 2, since these are the only MODES where the reactor can be critical, with the exception of Special Operations LCO 3.10.8, and these MODES are~~ 1, 2, and 3 which is consistent with the Applicability for the SLC System (LCO 3.1.7).

As noted (footnote (b) to Table 3.3.6.1-1), this Function is only required to close the outboard RWCU isolation valve trip systems.

5.f. Reactor Vessel Water Level—Low Low, Level 2

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some interfaces with the reactor vessel occurs to isolate the potential sources of a break. The isolation of the RWCU System on Level 2 supports actions to ensure that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. The Reactor Vessel Water Level—Low Low, Level 2 Function associated with RWCU isolation is not directly assumed in the FSAR safety analyses because the RWCU System line break is bounded by breaks of larger systems (recirculation and MSL breaks are more limiting).

Reactor Vessel Water Level—Low Low, Level 2 signals are initiated from four level instruments that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6.b. Reactor Vessel Water Level—Low, Level 3 (continued)

In MODES 1 and 2, another isolation (i.e., Reactor Steam Dome Pressure—High) and administrative controls ensure that this flow path remains isolated to prevent unexpected loss of inventory via this flow path.

6.c Manual Initiation

The Manual Initiation push button channels introduce signals to RHR Shutdown Cooling System isolation logic that is redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific FSAR safety analysis that takes credit for this Function. It is retained for overall redundancy and diversity of the isolation function as required by the NRC in the plant licensing basis.

There are two push buttons for the logic, one manual initiation push button per trip system. There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons.

Two channels of the Manual Initiation Function are available and are required to be OPERABLE in MODES 3, 4, and 5, since these are the MODES in which the RHR Shutdown Cooling System Isolation automatic Function are required to be OPERABLE.

Traversing Incore Probe System Isolation

7.a Reactor Vessel Water Level - Low, Level 3

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 3 supports actions to ensure that offsite and control room dose regulatory limits ~~of 10 CFR 100~~ are not exceeded. The Reactor Vessel Water Level - Low, Level 3 Function associated with isolation is implicitly assumed in the FSAR analysis as these leakage paths are assumed to be isolated post LOCA.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7.a Reactor Vessel Water Level - Low, Level 3 (continued)

Reactor Vessel Water Level - Low, Level 3 signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Two channels of Reactor Vessel Water Level - Low, Level 3 Function are available and are required to be OPERABLE to ensure that no single instrument failure can initiate an inadvertent isolation actuation. The isolation function is ensured by the manual shear valve in each penetration.

The Reactor Vessel Water Level - Low, Level 3 Allowable Value was chosen to be the same as the RPS Level 3 scram Allowable Value (LCO 3.3.1.1), since isolation of these valves is not critical to orderly plant shutdown.

7.b. Drywell Pressure - High

and control room

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits ~~of 10 CFR 100~~ are not exceeded. The Drywell Pressure - High Function, associated with isolation of the primary containment, is implicitly assumed in the FSAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

regulatory

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Two channels of Drywell Pressure - High per Function are available and are required to be OPERABLE to ensure that no single instrument failure can initiate an inadvertent actuation. The isolation function is ensured by the manual shear valve in each penetration.

The Allowable Value was selected to be the same as the ECCS Drywell Pressure - High Allowable Value (LCO 3.3.5.1), since this may be indicative of a LOCA inside primary containment.

(continued)

BASES

BACKGROUND (continued)

system initiates isolation of one automatic isolation valve (damper) and starts one SGT subsystem (including its associated reactor building recirculation subsystem) while the other trip system initiates isolation of the other automatic isolation valve in the penetration and starts the other SGT subsystem (including its associated reactor building recirculation subsystem). Each logic closes one of the two valves on each penetration and starts one SGT subsystem, so that operation of either logic isolates the secondary containment and provides for the necessary filtration of fission products.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The isolation signals generated by the secondary containment isolation instrumentation are implicitly assumed in the safety analyses of References 1 and 2 to initiate closure of valves and start the SGT System to limit offsite doses.

and control room

Refer to LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," Applicable Safety Analyses Bases for more detail of the safety analyses.

The secondary containment isolation instrumentation satisfies Criterion 3 of the NRC Policy Statement. (Ref. 7) Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the secondary containment isolation instrumentation is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have the required number of OPERABLE channels with their setpoints set within the specified Allowable Values, as shown in Table 3.3.6.2-1. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. Each channel must also respond within its assumed response time, where appropriate.

Allowable Values are specified for each Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Reactor Vessel Water Level—Low Low, Level 2 (continued)

level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low Low, Level 2 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low Low, Level 2 Allowable Value was chosen to be the same as the High Pressure Coolant Injection/Reactor Core Isolation Cooling (HPCI/RCIC) Reactor Vessel Water Level—Low Low, Level 2 Allowable Value (LCO 3.3.5.1 and LCO 3.3.5.2), since this could indicate that the capability to cool the fuel is being threatened.

The Reactor Vessel Water Level—Low Low, Level 2 Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the Reactor Coolant System (RCS); thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, this Function is not required. In addition, the Function is also required to be OPERABLE during operations with a potential for draining the reactor vessel (OPDRVs) because the capability of isolating potential sources of leakage must be provided to ensure that offsite dose limits are not exceeded if core damage occurs.

and control room

Reactor Vessel Water Level–Low Low, Level 2 will isolate the affected Unit's zone (i.e., Zone I for Unit 1 and Zone II for Unit 2) and Zone III.

2. Drywell Pressure—High

High drywell pressure can indicate a break in the reactor coolant pressure boundary (RCPB). An isolation of the secondary containment and actuation of the SGT System are initiated in order to minimize the potential of an offsite dose release. The isolation on high drywell pressure supports actions to ensure that any offsite releases are within the limits calculated in the safety analysis. However, the Drywell Pressure—High Function associated with

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

<u>3, 4, 5, 6, 7 Refuel Floor High Exhaust Duct, Refuel Floor Wall Exhaust Duct, and Railroad Access Shaft Exhaust Duct Radiation—High</u> (continued)

The Exhaust Radiation—High signals are initiated from radiation detectors that are located on the ventilation exhaust ductwork coming from the refueling floor zones and the Railroad Access Shaft. The signal from each detector is input to an individual monitor whose trip outputs are assigned to an isolation channel. Eight channels of Refuel Floor High Exhaust Duct and Wall Exhaust Duct Radiation—High Function (four from Unit 1 and four from Unit 2) and two channels of Railroad Access Shaft Exhaust Duct Radiation - High Function (both from Unit 1) are available to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding.

The Refuel Floor Exhaust Radiation—High Functions are required to be OPERABLE during CORE ALTERATIONS, OPDRVs, and movement of irradiated fuel assemblies in the secondary containment, because the capability of detecting radiation releases due to fuel failures (due to a fuel handling accident) must be provided to ensure that offsite dose limits are not exceeded. and control room

The Railroad Access Shaft Exhaust Duct Radiation - High Function is only required to be OPERABLE during handling of irradiated fuel within the Railroad Access Shaft, and above the Railroad Access Shaft with the Railroad Access Shaft Equipment Hatch open. This provides the capability of detecting radiation releases due to fuel failures resulting from dropped fuel assemblies which ensures that offsite dose limits are not exceeded. and control room

Refuel Floor High and Wall Exhaust Duct and Railroad Access Shaft Exhaust Duct Radiation - High Functions will isolate Zone III of secondary containment.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The ability of the CREOAS System to maintain the habitability of the main control room is explicitly assumed for certain accidents as discussed in the FSAR safety analyses (Refs. 1 and 2). CREOAS System operation ensures that the radiation exposure of control room personnel, through the duration of any one of the postulated accidents, does not exceed ~~the~~ regulatory limits ~~set by GDC 19 of 10 CFR 50, Appendix A~~.

CREOAS System instrumentation satisfies Criterion 3 of the NRC Policy Statement. (Ref. 5)

The OPERABILITY of the CREOAS System instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.7.1-1. Each Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Values, where appropriate. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

Allowable Values are specified for each CREOAS System Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter reaches the setpoint, the associated device changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3, 4, 5, 6, 7 Refuel Floor High Exhaust Duct, Refuel Floor Wall Exhaust Duct and Railroad Access Shaft Exhaust Duct Radiation—High (continued)

Duct Radiation—High Function (four from Unit 1 and four from Unit 2), and two channels of the Railroad Access Shaft Exhaust Radiation - High Function (both from Unit 1) are available and are required to be OPERABLE when the associated Refuel Floor Exhaust System is in operation to ensure that no single instrument failure can preclude the initiation function.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding. The Refuel Floor Exhaust Duct and Wall Exhaust Duct Radiation—High are required to be OPERABLE during CORE ALTERATIONS, OPDRVs, and movement of irradiated fuel assemblies in the secondary containment, because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded. and control room

The Railroad Access Shaft Exhaust Duct Radiation - High Function is only require d to be OPERABLE during handling of irradiated fuel within the Railroad Access Shaft, and above the Railroad Access Shaft with the Railroad Access Shaft Equipment Hatch open, because the capability of detecting radiation releases due to fuel failures (dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded. and control room

8. Main Control Room Outside Air Intake Radiation—High

The main control room outside air intake radiation monitors measure radiation levels at the control structure outside air intake duct. A high radiation level may pose a threat to main control room personnel; thus, automatically initiating the CREOAS System. The Control Room Air Inlet Radiation—High Function consists of two independent monitors. Two channels of Control Room Air Inlet Radiation—High are available and are required to be OPERABLE to ensure that no single instrument failure can preclude CREOAS System initiation. The Allowable Value was selected to ensure protection of the control room personnel.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Specific Activity

BASES

BACKGROUND

During circulation, the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks into the reactor coolant and activation of corrosion products in the reactor coolant. These radioactive materials in the reactor coolant can plate out in the RCS, and, at times, an accumulation will break away to spike the normal level of radioactivity. The release of coolant during a Design Basis Accident (DBA) could send radioactive materials into the environment.

Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure that in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within ~~the~~ regulatory limits ~~of 10 CFR 100 (Ref. 1)~~.

This LCO contains iodine specific activity limits. The iodine isotopic activities per gram of reactor coolant are expressed in terms of a DOSE EQUIVALENT I-131. The allowable levels are intended to limit the 2 hour radiation dose to an individual at the site boundary to ~~a small fraction of the 10 CFR 100~~ regulatory limits.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving radioactive material in the primary coolant are presented in the FSAR (Ref. 2). The specific activity in the reactor coolant (the source term) is an initial condition for evaluation of the consequences of an accident due to a main steam line break (MSLB) outside containment. No fuel damage is postulated in the MSLB accident, and the release of radioactive material to the environment is assumed to end when the main steam isolation valves (MSIVs) close completely.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is ≤ 4.0 μCi/gm, samples must be analyzed for DOSE EQUIVALENT I-131 at least once every 4 hours. In addition, the specific activity must be restored to the LCO limit within 48 hours. The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with the normal processing systems.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

B.1, B.2.1, B.2.2.1, and B.2.2.2

If the DOSE EQUIVALENT I-131 cannot be restored to ≤ 0.2 μCi/gm within 48 hours, or if at any time it is > 4.0 μCi/gm, it must be determined at least once every 4 hours and all the main steam lines must be isolated within 12 hours. Isolating the main steam lines precludes the possibility of releasing radioactive material to the environment in an amount that ~~is more than a small fraction of the requirements of 10 CFR 100~~ would exceed dose regulatory limits during a postulated MSLB accident.

Alternatively, the plant can be placed in MODE 3 within 12 hours and in MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

(continued)

BASES

ACTIONS

B.1, B.2.1, B.2.2.1, and B.2.2.2 (continued)

The Completion Time of once every 4 hours is the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines in an orderly manner and without challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for placing the unit in MODES 3 and 4 are reasonable, based on operating experience, to achieve the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level.

This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.

REFERENCES

1. ~~10 CFR 100.11, 1973.~~ DELETED

2. FSAR, Section 15.6.4.

3. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

BASES (continued)

APPLICABLE SAFETY ANALYSES

The safety design basis for the primary containment is that it must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage.

Analytical methods and assumptions involving the primary containment are presented in References 1 and 2. The safety analyses assume a nonmechanistic fission product release following a DBA, which forms the basis for determination of offsite doses. The fission product release is, in turn, based on an assumed leakage rate from the primary containment. OPERABILITY of the primary containment ensures that the leakage rate assumed in the safety analyses is not exceeded.

and control room

The maximum allowable leakage rate for the primary containment (L_a) is 1.0% by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_a) of 45 psig.

Primary containment satisfies Criterion 3 of the NRC Policy Statement. (Ref. 6)

LCO

Primary containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$, except prior to each startup after performing a required Primary Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.
Individual leakage rates specified for the primary containment air lock are addressed in LCO 3.6.1.2.

Leakage requirements for MSIVs and Secondary containment bypass are addressed in LCO 3.6.1.3.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.1 (continued) and control room

to meet these SRs must be evaluated against the Type A, B, and C acceptance criteria of the Primary Containment Leakage Rate Testing Program. As left leakage prior to each startup after performing a required leakage test is required to be $< 0.6\ L_a$ for combined Type B and C leakage, and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

As noted in Table B 3.6.1.3-1, an exemption to Appendix J is provided that isolation barriers which remain filled or a water seal remains in the line post-LOCA are tested with water and the leakage is not included in the Type B and C $0.60\ L_a$ test.

SR 3.6.1.1.2

Maintaining the pressure suppression function of primary containment requires limiting the leakage from the drywell to the suppression chamber. Thus, if an event were to occur that pressurized the drywell, the steam would be directed through the downcomers into the suppression pool. This SR measures drywell to suppression chamber leakage to ensure that the leakage paths that would bypass the suppression pool are within allowable limits. The allowable limit is 10% of the acceptable SSES $A/\sqrt{k}$ design value. For SSES, the $A/\sqrt{k}$ design value is .0535 ft^2.

Satisfactory performance of this SR can be achieved by establishing a known differential pressure between the drywell and the suppression chamber and determining the leakage. The leakage test is performed when the 10 CFR 50, Appendix J, Type A test is performed in accordance with the Primary Containment Leakage Rate Testing Program. This testing Frequency was developed considering this test is

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The PCIVs LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory, and establishing the primary containment boundary during major accidents. As part of the primary containment boundary, PCIV OPERABILITY supports leak tightness of primary containment. Therefore, the safety analysis of any event requiring isolation of primary containment is applicable to this LCO.

The DBAs that result in a release of radioactive material within primary containment are a LOCA and a main steam line break (MSLB). In the analysis for each of these accidents, it is assumed that PCIVs are either closed or close within the required isolation times following event initiation. This ensures that potential paths to the environment through PCIVs (including primary containment purge valves) are minimized. ~~Of the events analyzed in Reference 1, the MSLB is the most limiting event due to radiological consequences~~. The closure time of the main steam isolation valves (MSIVs) is a significant variable from a radiological standpoint. The MSIVs are required to close within 3 to 5 seconds since the 5 second closure time is assumed in the analysis. The safety analyses assume that the purge valves were closed at event initiation. Likewise, it is assumed that the primary containment is isolated such that release of fission products to the environment is controlled.

for a MSLB outside primary containment

The DBA analysis assumes that within the required isolation time leakage is terminated, except for the maximum allowable leakage rate, L_a.

The single failure criterion required to be imposed in the conduct of unit safety analyses was considered in the original design of the primary containment purge valves. Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred.

The primary containment purge valves may be unable to close in the environment following a LOCA. Therefore, each of the purge valves is required to remain closed during MODES 1, 2, and 3 except as permitted under Note 2 of SR 3.6.1.3.1. In this case, the single failure criterion remains applicable to the primary containment purge valve

(continued)

BASES

SURVEILLANCE REQUIREMENTS

S R 3.6.1.3.5 (continued)

OPERABILITY. MSIVs may be excluded from this SR since MSIV full closure isolation time is demonstrated by SR 3.6.1.3.7. The isolation time test ensures that the valve will isolate in a time period less than or equal to that assumed in the Final Safety Analyses Report. The isolation time and Frequency of this SR are in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.6

For primary containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B, (Ref. 3), is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between primary containment and the environment), a Frequency of 184 days was established. The acceptance criteria for these valves is defined in the Primary Containment Leakage Rate Testing Program, 5.5.12.

The SR is modified by a Note stating that the primary containment purge valves are only required to meet leakage rate testing requirements in MODES 1, 2, and 3. If a LOCA inside primary containment occurs in these MODES, purge valve leakage must be minimized to ensure offsite radiological release is within limits. At other times when the purge valves are required to be capable of closing (e.g., during handling of irradiated fuel), pressurization concerns are not present and the purge valves are not required to meet any specific leakage criteria.

SR 3.6.1.3.7

Verifying that the isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The isolation time test ensures that the MSIV will isolate in a time period that does not exceed the times assumed in the DBA analyses. This ensures that the calculated radiological consequences of these events remain within ~~10 CFR 100~~ limits.
regulatory

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.1.4 and SR 3.6.4.1.5

The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. To ensure that all fission products are treated, SR 3.6.4.1.4 verifies that the SGT System will rapidly establish and maintain a pressure in the secondary containment that is less than the pressure external to the secondary containment boundary. This is confirmed by demonstrating that one SGT subsystem will draw down the secondary containment to ≥ 0.25 inches of vacuum water gauge in less than or equal to the maximum time allowed. This cannot be accomplished if the secondary containment boundary is not intact. SR 3.6.4.1.5 demonstrates that one SGT subsystem can maintain ≥ 0.25 inches of vacuum water gauge for at least 1 hour at less than or equal to the maximum flow rate permitted for the secondary containment configuration that is operable. The 1 hour test period allows secondary containment to be in thermal equilibrium at steady state conditions. As noted, both SR 3.6.4.1.4 and SR 3.6.4.1.5 acceptance limits are dependent upon the secondary containment configuration when testing is being performed. The acceptance criteria for the SRs based on secondary containment configuration is defined as follows:

SECONDARY CONTAINMENT TEST CONFIGURATION	MAXIMUM DRAWDOWN TIME(SEC) (SR 3.6.4.1.4 ACCEPTANCE CRITERIA)	MAXIMUM FLOW RATE (CFM) (SR 3.6.4.1.5 ACCEPTANCE CRITERIA)
Zones I, II and III.	300 ≤ ~~425~~ Seconds (Zones I, II, and III)	≤ 4000 CFM (From Zones I, II, and III)
Zones II and III.	300 ≤ ~~440~~ Seconds (Zones II and III)	≤ 2960 CFM (From Zones II and III)

Only one of the above listed configurations needs to be tested to confirm secondary containment OPERABILITY.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.3 Control Room Emergency Outside Air Supply (CREO~~O~~AS) System

BASES

BACKGROUND

The CREOAS System provides a protected ~~radiologically controlled~~ environment from which operators can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or smoke from outside the control room envelope. ~~can be safely operated following a Design Basis Accident (DBA).~~ This ~~radiologically controlled~~protected environment is termed the habitability envelope and is comprised of Control Structure floor elevations 697'-0" through 783'-0" including the stairwells as described in FSAR Section 6.4, (Ref. 1)

The safety related function of CREOAS System includes two independent and redundant high efficiency air filtration subsystems for emergency treatment of outside supply air and a control room boundary which limits the inleakage of unfiltered air. Each subsystem consists of an electric heater, a prefilter, an upstream high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section, a downstream HEPA filter, a CREOAS fan, a control structure heating and ventilation fan, a control room floor cooling fan, a computer room floor cooling fan, and the associated ductwork, ~~and~~ dampers, doors, barriers and instrumentation. Prefilters and HEPA filters remove particulate matter, which may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay. With the exception of the CREOAS fan, all other CREOAS subsystem fans operate continuously to maintain the affected compartments environment. These other ventilation fans operate independently of the CREOAS fans and are required to operate to ensure a positive pressure in the control structure is maintained utilizing filtered outside air supplied by the CREOAS fans.

The habitability envelope is protected for normal operation, natural events, and accident conditions. The habitability envelope is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the habitability envelope. The integrity of the habitability envelope must be maintained to limit the inleakage of unfiltered air into the habitability envelope. The habitability envelope and habitability boundary are defined in the Control Room Habitability Program.

Upon receipt of the initiation signal(s) (indicative of conditions that could result in radiation exposure to occupants~~plant operators~~ in the habitability envelope), the CREOAS System automatically switches to the pressurization/filtration mode of operation to prevent infiltration of contaminated air into the habitability envelope. A system of dampers aligns the outside air intake to the CREOAS fan suction and filter train. Outside air is taken in at the normal ventilation intake and passed through

(continued)

one of the charcoal adsorber filter subsystems. The filtered air leaving the CREOAS filtration train is routed to the inlet of the other ventilation fans for distribution.

One of the CREOAS System design requirements is to maintain the habitability envelope~~control room~~ environment for a 30 day continuous occupancy after a DBA without exceeding dose regulatory limits~~5 remwhole body dose or its equivalent to any part of the body~~. A single CREOAS subsystem with an intact control structure habitability envelope will pressurize the habitability envelope (which

(continued)

BACKGROUND (continued)

includes the control room) to greater than or equal to 0.125 inches water gauge to prevent infiltration of air from surrounding buildings. CREOAS System operation in maintaining the habitability envelope environment is discussed in the FSAR, Chapters 6 and 9, (Refs. 1 and 2, respectively).

APPLICABLE SAFETY ANALYSES

The ability of the CREOAS System to maintain the habitability of the control structure habitability envelope is an explicit assumption for the safety analyses presented in the FSAR, Chapters 6 and 15 (Refs. 1 and 3, respectively). The pressurization/filtration mode of the CREOAS System is assumed to operate following a loss of coolant accident and~~,~~ fuel handling accident~~, and control rod drop accident,~~ as discussed in the FSAR, Section 6.4.1 (Ref. 1). The radiological doses to occupants~~plant operators~~ in the habitability envelope as a result of the various DBAs are summarized in Reference 3. No single active failure will cause the loss of outside or recirculated air.

The CREOAS System satisfies Criterion 3 of the NRC Policy Statement. (Ref. 4)

LCO

Two redundant subsystems of the CREOAS System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure or loss of habitability boundary integrity could result in exceeding ~~a dose~~ regulatory limits~~of 5 rem whole body or equivalent~~ to occupants~~plant operators~~ in the habitability envelope in the event of a DBA.

The CREOAS System is considered OPERABLE when the individual components necessary to limit~~control~~ operator exposure are OPERABLE ~~in both subsystems~~. Both subsystems are considered OPERABLE when:

a. Both filter trains each consisting of a CREOAS fan, heater, a HEPA filter, and charcoal adsorber which is not excessively restricting flow is OPERABLE; and

b. Both Control Structure Heating and Ventilation fans, Computer Room Floor Cooling fans, and Control Room Floor Cooling fans are OPERABLE; and

c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

d. Neither Smoke Removal Fan (0V104A/B) is in operation.

(continued)

LCO (continued)

One subsystem is considered OPERABLE when:

a. One filter train consisting of a CREOAS fan, heater, a HEPA filter, and charcoal adsorber which is not excessively restricting flow is OPERABLE; and

b. The 'A' Control Structure Heating and Ventilationfan (0V103A) and the 'A' Computer Room Floor Cooling fan (0V115A) and the 'A' Control Room Floor Cooling fan (0V117A) are OPERABLE

OR

The 'B' Control Structure Heating and Ventilation fan (0V103B) and the 'B' Computer Room Floor Cooling fan (0V115B) and the 'B' Control Room Floor Cooling fan (0V117B) are OPERABLE

(These fans are not dedicated to either CREOAS subsystem. As a result when any one set of fans is not OPERABLE, one arbitrarily determined CREOAS subsystem is not OPERABLE); and

c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

d. Neither Smoke Removal Fan (0V104A/B) is in operation.

~~In addition, the habitability envelope must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors to maintain a positive pressure.~~ Note the habitability envelope can not be maintained with a smoke removal fan (0V104A or 0V104B) in operation. In order for the CREOAS subsystems to be considered OPERABLE, the integrity of the habitability boundary must be maintained such that control room occupant dose from a large radioactive release dose not exceed regulatory limits and the control room occupants are protected from hazardous chemicals and smoke from outside the habitability boundary.

The LCO is modified by a Note allowing the control room habitability envelope boundary to be opened intermittently under administrative controls. This Note only applies to openings in the habitability envelope that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the control room operators. This individual will have a method to rapidly close the opening

(continued)

and to restore the habitability envelope to a condition equivalent to the design condition when a need for control room habitability envelope isolation is indicated.

APPLICABILITY

In MODES 1, 2, and 3, the CREOAS System must be OPERABLE to ensure that the habitability envelope will remain habitable ~~control operator exposure~~ during and following a DBA, since the DBA could lead to a fission product release.

(continued)

APPLICABILITY (continued)

In MODES 4 and 5, the probability and consequences of a DBA are reduced because of the pressure and temperature limitations in these MODES. Therefore, maintaining the CREOAS System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During operations with potential for draining the reactor vessel (OPDRVs);

b. During CORE ALTERATIONS; and

c During movement of irradiated fuel assemblies in the secondary containment

ACTIONS

A.1

With one CREOAS subsystem inoperable for reasons other than an inoperable habitability boundary, the inoperable CREOAS subsystem must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE CREOAS subsystem is adequate to perform the habitability envelope~~its radiation~~ protection function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced CREOAS System capability. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and that the remaining subsystem can provide the required capabilities.

B.1 and B.2

If the unfiltered inleakage of potentially contaminated air past the habitability boundary and into the habitability envelope can result in occupants of the habitability envelope receiving doses greater than regulatory limits or the control room occupants are not protected from hazardous chemicals or smoke from outside the habitability boundary, the habitability boundary is inoperable. However, the control room habitability boundary may be considered OPERABLE, but degraded or nonconforming, when unfiltered air inleakage is greater than assumed in the licensing basis accident analyses if compensatory measures can ensure that the habitability envelope remains habitable for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section 2.7.3, (Ref. 5) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 6). Temporary analytical methods may also be used as compensatory measures (Ref. 7). The Control Room Habitability Program provides limits on use of

(continued)

compensatory measures to maintain the control room habitability boundary operable but degraded or nonconforming when the unfiltered air inleakage is greater than that assumed in the licensing basis analyses. When those limits are exceeded, Condition B must be entered. Actions must be taken to restore an OPERABLE habitability boundary within 24 hours. During the period that the habitability boundary is inoperable, mitigating actions must be implemented to lessen the effect to control room occupants from the potential hazards of a radiological or chemical event or a challenge from smoke exhaust to the habitability envelope. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable habitability boundary) should be preplanned for implementation upon entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. ~~If the control room habitability envelope boundary is inoperable in MODES 1,2, and 3, the CREOAS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE control room habitability envelope boundary within 24 hours. During the period that the control room habitability envelope boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the Condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the control room habitability envelope boundary.~~

ACTIONS (continued)

C.1 and C.2

In MODE 1, 2, or 3, if the inoperable CREOAS subsystem or control room habitability envelope boundary cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1, D.2.1, D.2.2, and D.2.3

The Required Actions of Condition D are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while

(continued)

in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require either an entry into LCO 3.0.3 or a reactor shutdown in accordance with LCO 3.0.3.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, if the inoperable CREOAS subsystem cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CREOAS subsystem may be placed in the pressurization/filtration mode. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the habitability envelope~~control room~~. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

ACTIONS (continued)

E.1

If both CREOAS subsystems are inoperable in MODE 1, 2, or 3, for reasons other than an inoperable control room habitability envelope boundary (i. e., Condition B) the CREOAS System may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

F.1, F.2, and F.3

The Required Actions of Condition F are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require either an entry into LCO

(continued)

3.0.3 or a reactor shutdown in accordance with LCO 3.0.3.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, with two CREOAS subsystems inoperable or if the habitability boundary cannot be restored to OPERABLE status within the required Completion Time, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require pressurization of the habitability envelope. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. If applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that a CREOAS fan in a standby mode starts on demand from the control room and continues to operate with flow through the HEPA filters and charcoal adsorbers. Standby systems should be checked periodically to ensure that they start and function properly. As the environmental and normal operating conditions of this system are not severe, testing each subsystem once every month provides an adequate check on this system. Monthly heater operation dries out any moisture

that has accumulated in the charcoal as a result of humidity in the ambient air. Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Furthermore, the 31 day Frequency is based on the known reliability of the equipment and the availability of two redundant subsystems.

SR 3.7.3.2

This SR verifies that the required CREOAS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations).

(continued)

Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.3.3

This SR verifies that on an actual or simulated initiation signal, each CREOAS subsystem starts and operates. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.7.1.5 overlaps this SR to provide complete testing of the safety function. The 24 month Frequency is consistent with industry practice and other filtration systems SRs.

SR 3.7.3.4

This SR verifies the integrity of the habitability boundary by testing for unfiltered air inleakage past the habitability boundary and into the habitability envelope. The details of the testing are specified in the Control Room Habitability Program.

Unfiltered air inleakage through the habitability boundary and into the habitability envelope greater than the amount assumed in the licensing basis accident analyses results in the control room habitability boundary being inoperable when habitability is not maintained (i.e., accident dose is greater than regulatory limits or the occupants are not protected from hazardous chemical and smoke.) However, the control room habitability boundary may be considered OPERABLE, but degraded or nonconforming, when unfiltered air inleakage is greater than assumed in the licensing basis accident analyses if compensatory measures can ensure that the habitability envelope remains habitable for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section 2.7.3, (Ref. 5) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 6). Temporary analytical methods may also be used as compensatory measures (Ref. 7). The Control Room Habitability Program provides limits on use of compensatory measures to maintain the control room habitability boundary operable but degraded or nonconforming when the unfiltered air inleakage is greater than that assumed in the licensing basis analyses. When those limits are exceeded, Condition B must be entered. ~~This SR verifies the integrity of the habitability envelope and the assumed inleakage rates of potentially contaminated air. The habitability envelope positive pressure, with respect to potentially contaminated adjacent areas (the turbine building), is periodically tested to verify proper function of the CREOAS System and the integrity of the habitability envelope. During the emergency mode of operation, the CREOAS System is designed to slightly pressurize the control structure ≥ 0.125 inches water gauge positive pressure with respect to the outside atmosphere to prevent~~

(continued)

~~unfiltered inleakage. The CREOAS System is designed to maintain this positive pressure at a flow rate of ≤ 5810 cfm to the control structure in the pressurization/filtration mode. The control structure habitability envelope is maintained when the control structure habitability envelope can be pressurized to ≥ 0.125 inches water gauge positive pressure with respect to outside atmosphere. The Frequency of 24 months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration systems SRs.~~

(continued)

REFERENCES

1. FSAR, Chapter 6.

2. FSAR, Chapter 9.

3. FSAR, Chapter 15.

4. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

5. Regulatory Guide 1.196, May, 2003.

6. NEI 99-03 "Control Room Habitability Assessment", March 2003.

7. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2005, "NEI Draft White Paper, use of Generic Letter 91-18 Process and Alternate Source Terms in the Context of Control Room Habitability".

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Condenser Offgas

BASES

BACKGROUND

During unit operation, steam from the low pressure turbine is exhausted directly into the condenser. Air and noncondensible gases are collected in the condenser, then exhausted through the steam jet air ejectors (SJAEs) to the Main Condenser Offgas System. The offgas from the main condenser normally includes radioactive gases.

The Main Condenser Offgas System has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled by the offgas condenser; the water and condensibles are stripped out by the offgas condenser and moisture separator. The radioactivity of the remaining gaseous mixture (i.e., the offgas recombiner effluent) is monitored downstream of the moisture separator prior to entering the holdup line.

APPLICABLE SAFETY ANALYSES

The main condenser offgas radioactivity rate is an initial condition of the Main Condenser Offgas System failure event, discussed in the FSAR, Section 15.7.1 (Ref. 1). The analysis assumes a gross failure in the Main Condenser Offgas System that results in the rupture of the Main Condenser Offgas System pressure boundary. The radioactivity rate of the specified noble gases (Xe-133, Xe-135, Xe-138, Kr-85m, Kr-87, and Kr-88) is controlled to ensure that, during the event, the calculated offsite doses will be well within ~~the~~ limits ~~of 10 CFR 100 (Ref. 2)~~ or the NRC staff approved ~~licensing~~ regulatory basis.

The main condenser offgas limits satisfy Criterion 2 of the NRC Policy Statement. (Ref. 3)

LCO

To ensure compliance with the assumptions of the Main Condenser Offgas System failure event (Ref.1), the fission product release rate should be consistent with a specified noble gas release to the reactor coolant of

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.7 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool design is found in the FSAR, Section 9.1 (Ref. 1). The assumptions of the fuel handling accident are found in the FSAR, Section 15.7.4 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the radiological consequences (calculated ~~whole body and thyroid~~ doses at the exclusion area and low population zone boundaries) are ~~≤ 25% of 10 CFR 100 (Ref. 4) exposure guidelines NUREG-0800 (Ref. 3)~~ within the regulatory limits of 10CFR 50.67 (Ref. 4). A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide ~~1.25~~ 1.183 (Ref. 5).

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. With an assumed minimum water level of 21 ft and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and that offsite and control room doses are maintained within allowable limits (Ref. 2). The consequences of a fuel handling accident over the spent fuel storage pool are no more severe than those of the fuel handling accident over the reactor core, as discussed in the FSAR, Section 15.7.4 (Ref. 2). The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The spent fuel storage pool water level satisfies Criteria 2 and 3 of the NRC Policy Statement (Ref. 6).

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 9.1.
2. FSAR, Section 15.7.4.
3. ~~NUREG-0800, Section 15.7.4, Revision 1, July 1981.~~ Deleted.
4. 10 CFR ~~100.~~ 50.67
5. Regulatory Guide ~~1.25, March 1972.~~ 1.183, July 2000
6. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

B 3.9 REFUELING OPERATIONS

B 3.9.6 Reactor Pressure Vessel (RPV) Water Level

BASES

BACKGROUND

The movement of fuel assemblies or handling of control rods within the RPV requires a minimum water level of 22 ft above the top of the RPV flange. During refueling, this maintains a sufficient water level in the reactor vessel cavity and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to ≤ 25% of 10 CFR ~~100~~50.67 limits, as provided by the guidance of Reference ~~3~~1.

APPLICABLE SAFETY ANALYSES

During movement of fuel assemblies or handling of control rods, the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by Regulatory Guide 1.~~25~~183 (Ref. 1). A decontamination factor of ~~100~~38 ~~(Regulatory Position C.1.g of Ref. 1)~~ is used in the accident analysis for iodine. This relates to the assumption that 99.3% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the water. The fuel pellet to cladding gap is assumed to contain ~~10~~8% of the total fuel rod ~~iodine~~ I-131inventory and 5% of the total fuel rod I-132, I-133, I-134, and I-135 inventory (Ref. 1).

Analysis of the fuel handling accident inside containment is described in Reference 2. With an assumed minimum water level of 21 ft and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and that offsite and control room doses are maintained within allowable limits (Ref. 2).

While the worst case assumptions include the dropping of the irradiated fuel assembly being handled onto the reactor core, the possibility exists of the dropped assembly striking the RPV flange and releasing fission products. Therefore, the minimum depth for water coverage to ensure

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

acceptable radiological consequences is specified from the RPV flange. Since the worst case event results in failed fuel assemblies seated in the core, as well as the dropped assembly, dropping an assembly on the RPV flange will result in reduced releases of fission gases.

RPV water level satisfies Criterion 2 of the NRC Policy Statement (Ref. 5).

LCO

A minimum water level of 22 ft above the top of the RPV flange is required to ensure that the radiological consequences of a postulated fuel handling accident are within acceptable limits, as provided by the guidance of Reference ~~9.~~ 1

APPLICABILITY

LCO 3.9.6 is applicable when moving fuel assemblies or handling control rods (i.e., movement with other than the normal control rod drive) within the RPV. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. Requirements for fuel handling accidents in the spent fuel storage pool are covered by LCO 3.7.7, "Spent Fuel Storage Pool Water Level."

ACTIONS

A.1

If the water level is < 22 ft above the top of the RPV flange, all operations involving movement of fuel assemblies and handling control rods within the RPV shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of fuel movement and handling control rods shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 22 ft above the top of the RPV flange ensures that the design basis for the

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

postulated fuel handling accident analysis during refueling operations is met. Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide ~~1.25, March 23, 1972.~~ 1.183, July 2000
2. FSAR, Section 15.7.4.
3. ~~NUREG-0800, Section 15.7.4.~~ Deleted.
4. 10 CFR ~~100.11.~~ 50.67
5. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

Attachment 8 to PLA-5963

Activities to be Completed Before AST Implementation

Activities to be Completed Before AST Implementation

The following table identifies implementation activities to be taken by PPL Susquehanna as described in this document.

Activities	Due Date/Event
1. As a result of the analysis performed in Calculation EC-RADN-1129, "DBA LOCA Total Control Room Dose", access controls are required in the CRHE to maintain the Control Room operator dose as calculated and to meet the dose acceptance requirements of RG 1.183 and 10 CFR 50.67. Access control requirements in the CRHE result from core spray pipe shine located in the Reactor Building.	Access control requirements shall be completed prior to implementation of the AST License Amendments for Units 1 & 2.
Elevations 698' (Computer Room), 729'-1" (Control Room), and 741'-1" (TSC) are the only areas of the CRHE that require personnel to meet the occupancy requirements of RG 1.183, Section 4.2.6 during a DBA LOCA. Other areas of the CRHE require significantly less occupancy. Core spray piping is located in the adjacent RB near the northeast and southeast corners of the CRHE. A section of an Office (Room C-401) and a section of the Operational Support Center (Room C-402) located on elevation 729'-1" and the Electrical Room (Room C-413) and a section of the NRC Conference Room (Room C-414) located on elevation 741'-1", must have access controls to limit personnel exposure to ≤ 0.738 rem TEDE for the duration of the accident. This is accomplished by designating an area 5' from the CRHE east wall as a limited entry zone on elevations 729'-1" and 741'-1". Due to the location of the computer room on elevation 698', access controls are not required. The Emergency Plan and implementing procedures and station procedures including, but not limited to the following shall be reviewed and revised as necessary. EP-PS-104 Radiation Protection Coordinator Emergency Plan Position Specific Procedure EP-PS-106 Health Physics Specialist	Applicable Emergency Plan and implementing procedures and station procedures shall be completed prior to implementation of the AST License Amendments for Units 1 & 2.

2.	Applicable Sections of the TS and Bases were revised to reflect changes associated with the implementation of the proposed AST. The Control Room Habitability Program was added as Section 5.5.13. There are new manual operator actions associated with the SLC System required as part of this LAR that are not currently considered in the SSES design basis and must be directed by new Emergency Operation procedures that will be written and approved before SSES AST implementation. The operator actions assumed in the proposed DBA LOCA AST dose consequence analyses is the initiation of the SLC system for boron injection to maintain the suppression pool water pH above 7.0, precluding iodine re-evolution. TS Sections 3.1.7, "Standby Liquid Control (SLC) System and 3.3.6.1, "Primary Containment Isolation Instrumentation" and their Bases were also revised to address this change in the SLC system requirements (see Attachments 5 through 7). No hardware changes are necessary to use SLC in this new functional mode. Applicable procedure(s) will be reviewed/revised as necessary to ensure the operation of the SLC System during a DBA LOCA. See #4. Applicable sections of the TS Matrix shall be revised. As a result of the revision to the TS Bases concerning the definition of Dose Equivalent I-131, Plant Chemistry will evaluate revising appropriate software, counting system library (data file), and chemistry (CH) and emergency plan position specific (EP-PS) procedures. This assumes that the limiting values for coolant concentrations of 0.2 and 4.0 uCi/g DE I-131 are not re-evaluated. If Chemistry chooses to re-evaluate the limits, other changes would also be required.	A markup of the TS and Bases impacted by implementation of the AST is provided in Attachments 6 and 7 respectively. Applicable procedure(s) requirements and TS Matrix revisions shall be completed prior to implementation of the AST License Amendments for Units 1 & 2. The necessary software, data file, and procedural changes required to reflect the change in the DE I-131 definition shall be completed prior to implementation of the AST License Amendments for Units 1 & 2.
3.	Nuclear Fuels Engineering Technical Instruction NF-202 shall be modified to incorporate limits on LHGR for burnups exceeding 54 GWD/MTU per Footnote 11 of RG 1.183 and core average burnup of 39 GWd/MTU.	The procedural changes required to reflect compliance with Footnote 11 of RG 1.183 shall be completed prior to implementation of the AST License Amendments for Units 1 & 2.

4. As a result of the new SLC System function of controlling suppression pool pH post-LOCA, the station procedures including, but not limited to the following shall be reviewed and revised as necessary. Any procedures that call for the termination of the SLC System reactivity control measure will be reviewed and revised as necessary to prevent blocking of SLC System injection as a pH control measure. ES-150(250)-002 Boron Injection Via RCIC OP-153(253)-001 Standby Liquid Control System SO-100(200)-007 Daily Surveillance Operating Log SO-100(200)-008 Weekly Surveillance Operating Log EO-000-102 RPV Control EO-000-103 Primary Containment Control EO-000-113 Level/Power Control EO-000-114 RPV Flooding EP-PS-133 Severe Accident Management Coordinator NDAP-QA-0312 Controls of LCO's, TRO's, and Safety Function Determination Program Emergency Plan Damage Support Procedures System 53 Maintenance Rule Basis Document In addition, operator training will be updated to reflect the pH control function of the SLC System as defined in the procedural changes above. The impact of utilizing the sodium pentaborate water solution during a DBA LOCA to maintain suppression pool pH on equipment requiring to operate during the accident shall be evaluated.	The procedural changes required to reflect compliance with Footnote 11 of RG 1.183 shall be completed prior to implementation of the AST License Amendments for Units 1 & 2.
5. Applicable Sections of the FSAR will be reviewed and revised as necessary to reflect changes associated with the implementation of the proposed AST. Sections include, but are not limited to the following. Section 2.3, "Meteorology" Section 6.4, "Habitability Systems" Section 6.5, "Fission Product Removal and Control Systems" Section 9.3.5, "Standby Liquid Control System" Section 9.4, "Air Conditioning, Heating, Cooling and Ventilation Systems" Section 12.3.2.2.5, Control Room Shielding Design Section 15, "Accident Analysis" Section 18.1.69, "Integrity of Systems Outside Containment Likely to Contain Radioactive Material (III.D.1.1)"	Upon issuance of a License Amendment, conforming FSAR changes shall be completed as required by PPL procedures and submitted to the NRC staff in accordance with the regular FSAR update process as required by 10 CFR 50.71.

6. Review and revise as necessary the station procedures including, but not limited to the following to reflect a system leakage quantification limit of 2.5 gpm ESF leakage and 15 gpm CRD leakage. An ESF leak rate of 2.5 gpm is required to support the LOCA analysis, as discussed in Attachment 2, Section 4.3.4. The current value is 5 gpm. NDAP-QA-0412 Leakage Rate Test Program <u>Unit 1</u> SE-149-400 RHR System Leakage Quantification Test SE-150-400 RCIC System Leakage Quantification Test SE-151-400 Core Spray System Leakage Quantification Test SE-152-400 HPCI System Leakage Quantification Test SE-153-400 Standby Liquid Control System Discharge Line Leakage Quantification Test SE-155-400 CRD System Leakage Quantification Test SE-155-401 SDV Vent & Drain Valve Leakage Test SE-159-400 RHR/Core Spray/HPCI/RCIC Component Post-Maintenance Closed System Testing SE-161-400 RWCU System Leakage Quantification Test SE-176-400 PASS Liquid Leakage Quantification Test TP 159-004 Measurement of CRD Header Leakage <u>Unit 2</u> SE-249-400 RHR System Leakage Quantification Test SE-250-400 RCIC System Leakage Quantification Test SE-251-400 Core Spray System Leakage Quantification Test SE-252-400 HPCI System Leakage Quantification Test SE-253-400 Standby Liquid Control System Discharge Line Leakage Quantification Test SE-255-400 CRD System Leakage Quantification Test SE-155-401 SDV Vent & Drain Valve Leakage Test SE-259-400 RHR/Core Spray/HPCI/RCIC Component Post-Maintenance Closed System Testing SE-261-400 RWCU System Leakage Quantification Test SE-276-400 PASS Liquid Leakage Quantification Test TP 259-004 Measurement of CRD Header Leakage	The revision to procedures NDAP-QA-0412, Attachment P, Page 98 of 123 and the applicable Surveillance Engineering procedures shall be completed prior to implementation of the AST License Amendments for Units 1 & 2.

7. The χ/Qs calculated at the CRHE outside air intake are based on a new location, located on the roof of the Unit 2 Reactor Building (at column lines U and 36). A preliminary evaluation of the new location determined that seismic and security concerns were found to be acceptable. A more thorough evaluation of the acceptability of the new CRHE outside air intake location, including the impact of hazardous chemical and smoke on CRHE operators will be conducted. Appropriate drawings, station modification package(s), 10 CFR 50.59s, hazardous chemical and smoke evaluations, and other activities, as deemed appropriate, will be completed. Generate applicable TS changes to allow for the shutdown of the CREOASS to allow for connection of the existing CRHE ductwork to the new air intake.	The documentation and evaluation of the new CRHE outside air intake shall be completed prior to implementation of the AST License Amendments for Units 1 & 2. Appropriate drawings, station modification package(s), 10 CFR 50.59s, hazardous chemical and smoke evaluations, TS changes and other activities, as deemed appropriate, shall be completed prior to implementation of the AST License Amendments for Units 1 & 2.
8. The following assumptions were utilized in the AST analysis, based on projected EPU values: • For the DBA LOCA, the maximum bulk suppression pool water temperature shall not exceed 212 °F. • For the MSLB accident, the mass releases were increased by 20%. • For the DBA LOCA, a 50% reduction of primary containment leakage, secondary containment bypass leakage, and MSIV leakage at 24 hours. These assumptions shall be verified when the information becomes available, but prior to AST implementation.	The evaluations shall be completed prior to implementation of the AST License Amendments for Units 1 & 2.
9. The Emergency Plan and implementing procedures will be reviewed and updated as appropriate to reflect TEDE.	The reviews shall be completed prior to implementation of the AST License Amendments for Units 1 & 2.
10. The Control Room Habitability Program shall be developed.	Development of the Control Room Habitability Program shall be completed prior to implementation of the AST License Amendments for Units 1 & 2.

Attachment 9 to PLA-5963

No Significant Hazards Consideration Determination & Environmental Consideration for the Proposed Changes

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Description of Amendment Request

PPL Susquehanna, LLC (PPL) is proposing to amend the operating license for Susquehanna Steam Electric Station (SSES) Units 1 and 2, by revising the Technical Specifications (TS) and incorporating an alternative source term (AST) methodology into the facility's licensing basis. The proposed license amendment involves a full implementation of an AST methodology by revising the current accident source term and replacing it with an AST, as prescribed in 10 CFR 50.67.

AST analyses were performed using the guidance provided by Regulatory Guide 1.183, "Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000, and Standard Review Plan Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms." The four BWR limiting design basis accidents (DBAs) identified in RG 1.183 considered were the Control Rod Drop Accident, the Refueling Accident, the Loss of Coolant Accident, and the Main Steam Line Break Accident. As a result of the application of a revised accident source term, changes are proposed to the TS which revise the definition of dose equivalent I-131, and the operation of the SLC.

The AST analyses are based on new offsite and CRHE atmospheric dispersion coefficients (χ/Qs) based on site specific meteorological data determined based on Regulatory Guides 1.145 and 1.194.

In addition to revising the SSES licensing basis to adopt the AST, licensing basis changes are proposed and justified to respond to NRC Generic Letter 2003-01, "Control Room Habitability", dated June 12, 2003 (Reference 12.1). These proposed changes are pursuant to the Technical Specification Task Force Improved Standard Technical Specifications Change Traveler TSTF-448, Revision 2 (Reference 12.2).

Basis for No Significant Hazards Determination:

Pursuant to 10 CFR 50.92, SSES has reviewed the proposed change and concludes that the change does not involve a significant hazards consideration, since the proposed change satisfies the criteria in 10 CFR 50.92(c). These criteria require that the operation of the facility in accordance with the proposed amendment will not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. The discussion below addresses each of these criteria and demonstrates that the proposed amendment does not constitute a significant hazard.

1.0 Does the proposed change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

Response: No.

Adoption of the AST and pursuant TS changes, changes to the TS's to address NRC Generic Letter 2003-01 (Reference 12.1) and the changes to the atmospheric dispersion factors, have no impact to the initiation of DBAs. Once the occurrence of an accident has been postulated, the new accident source term and atmospheric dispersion factors are an input to analyses that evaluate the radiological consequences. Some of the proposed changes do affect the design or manner in which the facility is operated following an accident; however, the proposed changes do not involve a revision to the design or manner in which the facility is operated that could increase in the probability of an accident previously evaluated of a DBA discussed in Chapter 15 of the FSAR.

Therefore, the proposed change does not involve an increase in the probability of an accident previously evaluated.

The structures, systems and components affected by the proposed changes act as mitigators to the consequences of accidents. Based on the revised analyses, the proposed changes do revise certain performance requirements; however, the proposed changes do not involve a revision to the parameters or conditions that could contribute to the initiation of a DBA discussed in Chapter 15 of the FSAR.

Plant-specific radiological analyses have been performed using the AST methodology and new atmospheric dispersion factors. Based on the results of these analyses, it has been demonstrated that the CRHE dose consequences of the limiting events considered in the analyses meet the regulatory guidance provided for use with the AST, and the offsite doses are well within acceptable limits. This guidance is presented in 10 CFR 50.67, RG 1.183, and Standard Review Plan Section 15.0.1.

Therefore, the proposed amendment does not result in a significant increase in the consequences of any previously evaluated accident.

2.0 Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

Implementation of AST and the associated proposed TS changes and new atmospheric dispersion factors do not alter or involve any design basis accident initiators. These changes do not affect the design function or mode of operations of structures, systems and components in the facility prior to a postulated accident. Since structures, systems and components are operated essentially no differently after the AST implementation, no new failure modes are created by this proposed change.

Licensing basis changes are proposed and justified to credit use of the SLC System to buffer suppression pool pH to prevent iodine re-evolution following a postulated design basis loss of coolant accident. There are new required manual operator actions associated with the SLC System that are not currently considered in the SSES design basis. Operator training will be updated to reflect the new manual operator actions for the pH control function of the SLC System as defined in the TS Section 3.1.7. These changes are not significant because the operators are already trained for the operation of the SLC System. Procedural changes are mostly limited to the timing of SLC initiation and termination. In addition, no new hardware changes are necessary to use SLC in this new functional mode.

Licensing basis changes are proposed and justified for the operation of the CREOASS to respond to NRC Generic Letter 2003-01 and TSTF-448. No new hardware changes are necessary to implement these changes. Since CREOASS will not be operated differently as a result of these changes, no new failure modes are created by these changes.

Therefore, the proposed license amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3.0 Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The results of the accident analyses revised in support of the proposed change are subject to the acceptance criteria in 10 CFR 50.67. The analyzed events have been carefully selected, and the analyses supporting these changes have been performed using approved methodologies to ensure that analyzed events are bounding and safety margin has not been reduced. The dose consequences of these limiting events are within the acceptance criteria presented in 10 CFR 50.67, RG 1.183, and SRP 15.0.1. Thus, by meeting the applicable regulatory limits for AST, there is no significant reduction in a margin of safety.

Changes to the SLC System to credit use of the Standby Liquid Control (SLC) System to buffer suppression pool pH to prevent iodine re-evolution and the CREOASS to address NRC Generic Letter 2003-01 and TSTF-448 improve the margin of safety.

New offsite and Control Room atmospheric dispersion factors (χ/Qs) based on site specific meteorological data, calculated in accordance with the guidance of RGs 1.145 and 1.194, utilizes more recent data and improved calculational methodologies.

Therefore, because the proposed changes continue to result in dose consequences within the applicable regulatory limits, the changes are considered to not result in a significant reduction in a margin of safety.

<u>Conclusion</u>

On the basis of the above, SSES has determined that operation of the facility in accordance with the proposed change does not involve a significant hazards consideration as defined in 10 CFR 50.92(C), in that it: (1) does not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) does not create the possibility of a new or different kind of accident from any accident previously evaluated; and (3) does not involve a significant reduction in a margin of safety.

ENVIRONMENTAL CONSIDERATION

PPL has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21.

PPL has determined that the proposed change meets the criteria for categorical exclusion as provided for under 10 CFR 51.22(c)(9). 10 CFR 51.22(c)(9) identifies certain licensing and regulatory actions, which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility does not require an environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; or (3) result in a significant increase in individual or cumulative occupational radiation exposure. PPL Susquehanna, LLC has evaluated the proposed change and has determined that the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Accordingly, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with issuance of the amendment. The basis for this determination, using the above criteria, follows:

Basis

As demonstrated in the No Significant Hazards Consideration Evaluation, the proposed amendment does not involve a significant hazards consideration. As a result of the DBA LOCA analysis, the outside air intake of the CRHE shall be relocated. The change in location resulted from the need to lower the existing χ/Qs. An evaluation of the impact of hazardous chemicals and smoke on CRHE operators will be conducted, based on the new outside air intake location. The evaluation will be completed prior to implementation of the AST license amendments for Units 1 & 2. It is not anticipated that the relocation of the CRHE intake will result in an increase in the consequences previously analyzed.

There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite. The proposed change does not involve any unreviewed safety questions concerning the physical alteration of the plant or change in methods governing normal plant operation.

There is no significant increase in individual or cumulative occupational radiation exposure. The proposed change does not involve any unreviewed safety questions concerning the physical alteration of the plant or change in methods governing normal plant operation.

Conclusion

The alternative source term does not affect the design or operation of the facility; rather, once the occurrence of an accident has been postulated, the alternative source term is an input to evaluate the consequences of accidents. The implementation of the alternative source term has been evaluated in revisions to the analyses of the limiting design basis accidents at SSES (control rod drop accident, fuel handling accident, loss of coolant accident, and main steam line break accident). Based upon the results of these analyses it has been demonstrated that, with the requested changes, the dose consequences are within NRC regulatory limits for alternative source term (i.e., 10 CFR 50.67 and 10 CFR 50, Appendix A, General Design Criterion 19).

On the basis of the above, SSES has determined that operation of the facility in accordance with the proposed change does not involve a significant hazards consideration as defined in 10 CFR 51.22(c)(9), in that it does not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; or (3) result in a significant increase in individual or cumulative occupational radiation exposure.

Attachment 10 to PLA-5963

Non-Proprietary Versions of Supporting Calculations

The following non-proprietary Supporting Calculations are included in this section.

1. PPL Calculation EC-ENVR-1057, "Offsite χ/Q Values for the SSES Based on 1999-2003 Meteorological Data", Revision 0.

2. PPL Calculation EC-ENVR-1058, "CRHE Accident Dispersion Factors (χ/Q)", Revision 0.

3. PPL Calculation EC-ENVR-1059, "CRHE Accident Dispersion Factors (χ/Q) RB U2 Intake", Revision 0.

4. PPL Calculation EC-RADN-1125, "CRHE and Offsite Post LOCA Doses – AST", Revision 0.

5. PPL Calculation EC-RADN-1126, "CRHE and Offsite FHA/EHA Doses – AST", Revision 0.

6. PPL Calculation EC-RADN-1127, "Control Rod Drop Accident CRHE and Offsite Doses – AST", Revision 0.

7. PPL Calculation EC-RADN-1128, "Steam Line break Accident CRHE and Offsite Post LOCA Doses – AST", Revision 0.

8. PPL Calculation EC-059-1041, "Suppression Pool pH Post LOCA", Revision 0.

9. PPL Calculation EC-053-1012, "Assessment of SLC System for Suppression Pool pH Control", Revision 0.

10. PPL Calculation EC-RADN-1129, "DBA LOCA Total Control Room Dose", Revision 0.

11. PPL Calculation EC-RADN-1134, "Impact of AST on Current NUREG-0737 Radiological Evaluations that use TID-14844 DBA-LOCA Releases", Revision 0.

12. PPL Calculation EC-RADN-1135, "Justification of AST 60 Isotope RADTRAD Source Term for Direct Shine Calculations", Revision 0.

13. Non-proprietary pages of Calculation EC-FUEL-1615, "AREVA Alternate Source Term (AST) Fission Product Inventory for Atrium-10 Fuel."

Attachment 11 to PLA-5963

CD of Meteorological Data Used to Determine New χ/Qs

Attachment 12 to PLA-5963

References

12.0 REFERENCES

1. NRC Generic Letter 2003-01, "Control Room Habitability", dated June 12, 2003.

2. TSTF-448, Revision 2, BWOG-111, R0, "Technical Specification Task Force – Improved Standard Technical Specifications Change Traveler."

- Framatome-ANP and PPL Susquehanna performed calculations to determine radionuclide source terms for ATRIUM™-10 fuel irradiated in the Susquehanna units. The source terms were generated using the SAS2H/ORIGEN-S computer code system.

- This analysis assumes a core thermal power of 4032 MW_t (102% of 3952 MW_t), which corresponds to an input fuel assembly power of 5.28 MW_t. The fission product source terms calculated are slightly conservative (about 2%) because the uranium mass assumed is higher than expected ATRIUM-10 assembly mass values. The activity results have been selected to be applicable over a wide range of reload fuel conditions including enrichment levels of 4.5 wt% U235.

- Source terms were generated for two different scenarios. Core-wide source terms (full core inventories) were generated at a fuel exposure of 39 GWd/MTU for use in analyzing core-wide releases (LOCA, etc). Assembly source terms (for a single assembly) were generated at a fuel exposure of 58 GWd/MTU for use in analyzing releases from dropping or otherwise damaging a single assembly (Fuel Handling Accident, etc).

- The attached tables contain the results from the SAS2H/ORIGEN-S calculations. Four tables are provided:

  Table 3.1 – Core Activity in Curies following Burnup to 39 GWD/MTU

  Table 3.2 – Core Mass in Grams following Burnup to 39 GWD/MTU

  Table 3.3 – Assembly Activity in Curies following Burnup to 58 GWD/MTU

  Table 3.4 – Assembly Activity in Grams following Burnup to 58 GWD/MTU

  The "a", "l" or "f" in the 2nd, 3rd, or 4th column refer to the library in which the nuclide was found. An "a" refers to actinides; an "l" refers to light elements (i.e. structural materials), and an "f" refers to fission products. The values in the table will be the sum of the ORIGEN results from all 3 libraries.

Table 3.1

Activity (Curies) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide		limit enr	0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
h 3	f	4.25	7.58E+04	7.58E+04	7.58E+04	7.58E+04	7.58E+04	7.58E+04	7.57E+04	7.55E+04	7.47E+04	7.37E+04	7.17E+04	6.40E+04
c 15	l	4.25	1.95E+01	1.47E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
n 16	l	4.25	1.73E+02	1.57E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ne 23	l	4.25	3.46E+01	3.39E+01	9.78E-14	2.79E-28	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
na 24	l	4.25	2.73E+02	2.73E+02	2.66E+02	2.60E+02	1.86E+02	8.79E+01	2.91E+00	4.46E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00
na 25	l	4.25	2.09E+01	2.06E+01	1.70E-08	1.37E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mg 27	l	4.25	1.05E+03	1.05E+03	1.16E+02	1.29E+01	5.61E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
al 28	l	4.25	1.90E+04	1.89E+04	1.77E+00	1.65E-04	4.35E-07	2.56E-07	2.35E-08	2.45E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i128	f	4.25	1.19E+06	1.18E+06	5.17E+05	2.25E+05	1.96E+00	5.29E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
al 29	l	4.25	5.46E+01	5.45E+01	2.30E+00	9.70E-02	5.41E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
si 31	l	4.25	4.08E+02	4.08E+02	3.58E+02	3.13E+02	4.92E+01	7.16E-01	3.88E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ti 51	l	4.25	5.86E+01	5.84E+01	1.58E+00	4.28E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cr 51	l	4.25	5.65E+06	5.65E+06	5.65E+06	5.65E+06	5.61E+06	5.52E+06	5.11E+06	2.67E+06	5.95E+05	6.26E+04	6.08E+02	7.04E-06
v 52	l	4.25	8.56E+04	8.48E+04	3.34E+02	1.30E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
v 53	l	4.25	3.54E+02	3.52E+02	8.71E-04	2.14E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mn 54	l	4.25	4.27E+05	4.27E+05	4.27E+05	4.27E+05	4.27E+05	4.26E+05	4.23E+05	4.00E+05	3.50E+05	2.87E+05	1.90E+05	3.75E+04
cr 55	l	4.25	8.63E+04	8.56E+04	2.25E+02	5.88E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
fe 55	l	4.25	1.90E+06	1.90E+06	1.90E+06	1.90E+06	1.90E+06	1.90E+06	1.90E+06	1.86E+06	1.79E+06	1.68E+06	1.48E+06	8.86E+05
mn 56	l	4.25	1.21E+07	1.21E+07	1.05E+07	9.17E+06	1.40E+06	1.90E+04	7.47E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mn 57	l	4.25	1.36E+03	1.35E+03	8.33E-04	5.07E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
co 58	l	4.25	5.91E+05	5.91E+05	5.91E+05	5.91E+05	5.90E+05	5.86E+05	5.68E+05	4.41E+05	2.45E+05	1.02E+05	1.67E+04	1.32E+01
fe 59	l	4.25	1.28E+05	1.28E+05	1.28E+05	1.28E+05	1.28E+05	1.27E+05	1.21E+05	8.02E+04	3.16E+04	7.79E+03	4.35E+02	4.97E-03
ni 59	l	4.25	2.58E+02	2.58E+02	2.58E+02	2.58E+02	2.58E+02	2.58E+02	2.58E+02	2.58E+02	2.58E+02	2.58E+02	2.58E+02	2.58E+02
co 60	l	4.25	3.19E+05	3.19E+05	3.19E+05	3.19E+05	3.19E+05	3.19E+05	3.19E+05	3.16E+05	3.09E+05	2.99E+05	2.80E+05	2.15E+05
co 60m	l	4.25	5.27E+05	5.26E+05	7.24E+04	9.93E+03	8.33E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
co 61	l	4.25	7.11E+03	7.11E+03	5.76E+03	4.67E+03	2.47E+02	2.97E-01	2.17E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
co 62	l	4.25	5.66E+01	5.62E+01	5.41E-05	5.14E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ni 63	l	4.25	3.55E+04	3.55E+04	3.55E+04	3.55E+04	3.55E+04	3.55E+04	3.55E+04	3.55E+04	3.54E+04	3.54E+04	3.53E+04	3.48E+04
cu 64	l	4.25	2.65E+02	2.65E+02	2.58E+02	2.51E+02	1.72E+02	7.17E+01	1.41E+00	2.28E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ni 65	l	4.25	5.57E+04	5.57E+04	4.86E+04	4.23E+04	6.17E+03	7.57E+01	1.89E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cu 66	l	4.25	3.48E+02	3.48E+02	6.30E+00	4.91E-01	3.58E-01	2.93E-01	1.17E-01	4.26E-05	4.92E-13	6.10E-25	0.00E+00	0.00E+00
cu 73	f	4.25	2.80E+03	2.48E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zn 73	f	4.25	4.88E+03	4.82E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 73	f	4.25	4.97E+03	4.97E+03	4.64E+03	4.32E+03	1.59E+03	1.63E+02	5.65E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 73m	f	4.25	4.91E+03	4.91E+03	4.58E+03	4.26E+03	1.57E+03	1.60E+02	5.58E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.1

Activity (Curies) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide		limit enr	0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
cu 74	f	4.5	3.48E+03	1.32E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zn 74	f	4.5	1.18E+04	1.18E+04	2.70E-02	6.10E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 74	f	4.5	3.73E+03	3.73E+03	3.45E+02	2.66E+01	7.15E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cu 75	f	4.5	4.40E+03	2.10E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zn 75	f	4.5	2.75E+04	2.59E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 75	f	4.5	3.37E+04	3.36E+04	1.81E+00	9.02E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 75	f	4.5	3.39E+04	3.39E+04	2.71E+04	2.11E+04	6.26E+02	2.02E-01	3.96E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cu 76	f	4.5	3.56E+03	2.58E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zn 76	f	4.5	6.04E+04	5.34E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 76	f	4.5	9.02E+04	8.94E+04	2.48E-12	5.23E-29	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 76	f	4.25	2.56E+03	2.56E+03	2.53E+03	2.50E+03	2.08E+03	1.36E+03	2.05E+02	1.49E-05	5.05E-22	0.00E+00	0.00E+00	0.00E+00
cu 77	f	4.5	2.08E+03	2.14E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zn 77	f	4.5	7.37E+04	5.30E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 77	f	4.5	1.88E+05	1.82E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 77	f	4.5	7.31E+04	7.31E+04	7.10E+04	6.88E+04	4.48E+04	1.68E+04	2.02E+02	4.83E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 77m	f	4.5	1.93E+05	1.93E+05	1.49E-05	8.48E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 77	f	4.5	2.26E+05	2.26E+05	2.25E+05	2.23E+05	2.04E+05	1.60E+05	4.61E+04	6.72E-01	4.61E-12	8.71E-29	0.00E+00	0.00E+00
zn 78	f	4.5	9.47E+04	5.92E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 78	f	4.5	4.25E+05	3.80E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 78	f	4.5	7.22E+05	7.22E+05	5.70E+05	4.50E+05	1.65E+04	8.56E+00	1.43E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 78	f	4.5	7.32E+05	7.32E+05	7.13E+05	6.70E+05	8.25E+04	1.25E+02	1.39E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zn 79	f	4.5	4.45E+04	2.22E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 79	f	4.5	4.19E+05	3.39E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 79	f	4.5	1.24E+06	1.21E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 79	f	4.5	1.34E+06	1.34E+06	1.38E+05	1.38E+04	1.28E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 79m	f	4.5	1.34E+06	1.34E+06	2.37E+05	2.41E+04	2.24E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zn 80	f	4.25	1.67E+04	4.61E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 80	f	4.5	3.39E+05	2.26E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 80	f	4.5	3.03E+06	2.96E+06	1.31E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 80	f	4.5	3.53E+06	3.51E+06	2.71E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zn 81	f	4.25	3.77E+03	1.31E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 81	f	4.5	2.13E+05	1.21E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 81	f	4.5	3.21E+06	2.94E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 81	f	4.5	5.12E+06	5.07E+06	3.26E-10	1.73E-26	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 81	f	4.5	5.39E+06	5.39E+06	2.02E+06	8.02E+05	1.71E+03	1.54E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.1

Activity (Curies) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide		limit enr	0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
se 81m	f	4.5	3.87E+05	3.87E+05	2.70E+05	1.88E+05	1.16E+03	1.04E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 82	f	4.5	1.34E+05	4.20E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 82	f	4.5	3.08E+06	2.66E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 82	f	4.5	5.03E+06	4.96E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 82m	f	4.5	1.96E+06	1.86E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 82	f	4.25	3.81E+05	3.81E+05	3.78E+05	3.75E+05	3.27E+05	2.38E+05	5.81E+04	2.77E-01	1.46E-13	0.00E+00	0.00E+00	0.00E+00
br 82m	f	4.25	3.29E+05	3.29E+05	1.11E+04	3.72E+02	8.86E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 83	f	4.5	1.60E+04	1.70E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 83	f	4.5	1.38E+06	9.63E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 83	f	4.5	8.02E+06	7.64E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 83	f	4.5	6.10E+06	6.10E+06	2.41E+06	9.47E+05	2.03E+00	2.23E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 83m	f	4.5	6.45E+06	6.43E+06	1.46E-01	2.70E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 83	f	4.5	1.29E+07	1.29E+07	1.18E+07	1.04E+07	1.40E+06	1.38E+04	1.28E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 83m	f	4.5	1.31E+07	1.31E+07	1.29E+07	1.26E+07	3.64E+06	5.27E+04	5.39E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 84	f	4.5	1.28E+05	1.09E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 84	f	4.5	8.86E+05	5.02E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 84	f	4.5	5.91E+06	5.29E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 84	f	4.5	2.34E+07	2.33E+07	3.55E+04	5.34E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 84	f	4.5	2.40E+07	2.40E+07	1.38E+07	7.20E+06	7.62E+02	6.22E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 84m	f	4.5	6.43E+05	6.42E+05	2.01E+04	6.27E+02	5.32E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 85	f	4.5	1.53E+05	9.47E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 85	f	4.5	3.40E+06	2.43E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 85	f	4.5	1.10E+07	1.08E+07	9.02E-11	7.15E-28	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 85m	f	4.5	9.70E+06	9.32E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 85	f	4.5	2.67E+07	2.67E+07	2.17E+04	1.54E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 85	f	4.5	1.48E+06	1.48E+06	1.48E+06	1.48E+06	1.48E+06	1.48E+06	1.48E+06	1.47E+06	1.46E+06	1.44E+06	1.39E+06	1.22E+06
kr 85m	f	4.5	2.68E+07	2.68E+07	2.51E+07	2.32E+07	7.87E+06	6.62E+05	9.63E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 86	f	4.5	3.20E+04	1.92E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 86	f	4.5	1.89E+06	8.86E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 86	f	4.5	2.64E+07	2.52E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 86	f	4.5	3.25E+07	3.25E+07	6.28E-03	9.17E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 86m	f	4.5	6.21E+06	5.33E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 86	f	4.25	2.17E+05	2.17E+05	2.16E+05	2.16E+05	2.14E+05	2.09E+05	1.86E+05	7.10E+04	7.62E+03	2.68E+02	2.73E-01	4.31E-13
rb 86m	f	4.25	1.79E+04	1.76E+04	2.36E-05	3.09E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 87	f	4.5	2.70E+05	1.51E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.1

Activity (Curies) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide		limit enr	0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
as 87	f	4.5	1.08E+06	1.25E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 87	f	4.5	1.54E+07	1.36E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 87	f	4.5	4.24E+07	4.20E+07	8.25E-03	1.52E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 87	f	4.5	5.37E+07	5.37E+07	4.13E+07	3.15E+07	6.94E+05	1.13E+02	1.03E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 88	f	4.5	4.79E+05	2.78E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 88	f	4.5	8.17E+06	5.18E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 88	f	4.5	4.09E+07	3.95E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 88	f	4.5	7.45E+07	7.45E+07	6.60E+07	5.84E+07	1.05E+07	2.12E+05	4.93E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 88	f	4.5	7.64E+07	7.64E+07	7.16E+07	6.46E+07	1.18E+07	2.38E+05	5.51E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr 89	l	4.25	5.52E+04	5.52E+04	5.50E+04	5.48E+04	5.15E+04	4.47E+04	2.37E+04	9.55E+01	2.84E-04	1.46E-12	1.74E-29	0.00E+00
as 89	f	4.25	8.56E+03	2.77E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 89	f	4.5	2.86E+06	5.25E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 89	f	4.5	2.83E+07	2.44E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 89	f	4.5	9.24E+07	9.24E+07	1.32E+05	1.86E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 89	f	4.5	9.93E+07	9.93E+07	3.16E+07	8.02E+06	3.87E-02	3.76E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 89	l f	4.5	1.03E+08	1.03E+08	1.03E+08	1.03E+08	1.03E+08	1.02E+08	9.78E+07	6.85E+07	3.01E+07	8.79E+06	6.91E+05	3.09E+01
y 89m	l f	4.25	1.68E+05	1.63E+05	6.43E+04	6.40E+04	6.07E+04	5.39E+04	3.25E+04	6.46E+03	2.80E+03	8.18E+02	6.43E+01	2.87E-03
se 90	f	4.5	6.30E+05	1.24E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 90	f	4.5	1.54E+07	1.08E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 90	f	4.5	9.93E+07	9.70E+07	1.73E-09	2.92E-26	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 90	f	4.5	9.17E+07	9.17E+07	4.70E+04	1.23E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 90m	f	4.5	2.91E+07	2.90E+07	2.44E+05	1.94E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 90	f	4.5	1.31E+07	1.31E+07	1.31E+07	1.31E+07	1.31E+07	1.31E+07	1.31E+07	1.30E+07	1.30E+07	1.29E+07	1.27E+07	1.21E+07
y 90	l f	4.5	1.36E+07	1.36E+07	1.36E+07	1.36E+07	1.35E+07	1.34E+07	1.32E+07	1.30E+07	1.30E+07	1.29E+07	1.27E+07	1.21E+07
zr 90m	f	4.25	8.86E+03	3.77E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 91	f	4.5	6.31E+04	4.82E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 91	f	4.5	5.03E+06	1.59E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 91	f	4.5	6.79E+07	6.28E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 91	f	4.5	1.21E+08	1.21E+08	7.04E-02	3.68E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 91	l f	4.5	1.31E+08	1.31E+08	1.26E+08	1.21E+08	7.30E+07	2.28E+07	1.21E+05	2.23E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 91	l f	4.5	1.34E+08	1.34E+08	1.34E+08	1.34E+08	1.34E+08	1.34E+08	1.29E+08	9.48E+07	4.65E+07	1.60E+07	1.78E+06	3.11E+02
y 91m	f	4.5	7.57E+07	7.57E+07	7.53E+07	7.40E+07	4.64E+07	1.44E+07	7.64E+04	1.41E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 92	f	4.25	5.17E+03	8.33E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 92	f	4.5	8.56E+05	1.28E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 92	f	4.5	3.61E+07	2.49E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.1

Activity (Curies) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide		limit enr	0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
rb 92	f	4.5	1.08E+08	9.63E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 92	f	4.5	1.39E+08	1.39E+08	1.22E+08	1.08E+08	1.80E+07	2.99E+05	3.01E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 92	l f	4.5	1.40E+08	1.40E+08	1.39E+08	1.37E+08	6.53E+07	4.42E+06	4.04E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr 93	l	4.25	2.97E+01	2.97E+01	2.97E+01	2.97E+01	2.97E+01	2.97E+01	2.97E+01	2.97E+01	2.97E+01	2.97E+01	2.97E+01	2.97E+01
br 93	f	4.25	2.60E+05	5.06E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 93	f	4.5	1.22E+07	7.14E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 93	f	4.5	8.94E+07	8.02E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 93	l f	4.5	1.57E+08	1.57E+08	9.63E+06	5.84E+05	5.42E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 93	l f	4.5	1.07E+08	1.07E+08	1.04E+08	1.01E+08	6.23E+07	2.08E+07	1.48E+05	3.74E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 94	f	4.25	1.23E+04	2.33E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 94	f	4.5	5.64E+06	2.07E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 94	f	4.5	4.63E+07	3.61E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 94	f	4.5	1.57E+08	1.56E+08	9.93E+00	6.13E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 94	l f	4.5	1.70E+08	1.70E+08	5.95E+07	1.96E+07	3.39E+00	1.19E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 95	f	4.25	5.39E+05	2.21E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 95	f	4.5	2.25E+07	3.91E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 95	f	4.5	1.41E+08	1.38E+08	3.67E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 95	f	4.5	1.76E+08	1.76E+08	2.52E+07	3.48E+06	3.16E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr 95	l f	4.5	1.92E+08	1.92E+08	1.92E+08	1.92E+08	1.91E+08	1.89E+08	1.84E+08	1.39E+08	7.23E+07	2.73E+07	3.67E+06	1.35E+03
nb 95	l f	4.5	1.92E+08	1.92E+08	1.92E+08	1.92E+08	1.92E+08	1.92E+08	1.92E+08	1.79E+08	1.21E+08	5.36E+07	7.92E+06	2.98E+03
nb 95m	l f	4.5	2.13E+06	2.13E+06	2.13E+06	2.13E+06	2.13E+06	2.12E+06	2.10E+06	1.63E+06	8.50E+05	3.21E+05	4.32E+04	1.59E+01
kr 96	f	4.25	9.17E+04	8.56E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 96	f	4.5	5.62E+06	1.87E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 96	f	4.5	1.03E+08	5.43E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 96	f	4.5	1.70E+08	1.60E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb 96	l f	4.25	3.12E+05	3.12E+05	3.07E+05	3.02E+05	2.45E+05	1.53E+05	1.80E+04	1.63E-04	4.44E-23	0.00E+00	0.00E+00	0.00E+00
rb 97	f	4.5	1.54E+06	2.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 97	f	4.5	5.23E+07	1.02E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 97	f	4.5	1.40E+08	1.18E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr 97	l f	4.5	1.90E+08	1.90E+08	1.86E+08	1.82E+08	1.36E+08	7.09E+07	3.70E+06	2.84E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb 97	l f	4.5	1.91E+08	1.91E+08	1.90E+08	1.88E+08	1.46E+08	7.13E+07	3.72E+06	3.06E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb 97m	l f	4.5	1.80E+08	1.80E+08	1.76E+08	1.73E+08	1.30E+08	6.73E+07	3.51E+06	2.70E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 98	f	4.25	1.54E+05	3.64E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 98	f	4.5	2.13E+07	7.33E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 98	f	4.5	1.04E+08	4.30E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.1

Activity (Curies) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide		limit enr	0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
zr 98	f	4.5	1.84E+08	1.82E+08	4.22E-10	9.24E-28	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb 98	f	4.5	1.86E+08	1.86E+08	4.65E-10	1.01E-27	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb 98m	f	4.25	1.47E+06	1.47E+06	9.78E+05	6.52E+05	2.24E+03	5.20E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 99	f	4.25	4.46E+03	3.44E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 99	f	4.5	8.17E+06	6.33E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 99	f	4.5	6.59E+07	4.21E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr 99	f	4.5	1.82E+08	1.45E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb 99	f	4.5	1.18E+08	1.17E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb 99m	f	4.5	8.10E+07	8.10E+07	2.76E+04	9.24E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo 99	I f	4.25	2.02E+08	2.02E+08	2.01E+08	2.00E+08	1.86E+08	1.57E+08	7.37E+07	1.05E+05	2.79E-02	3.84E-12	0.00E+00	0.00E+00
tc 99	f	4.5	2.21E+03	2.21E+03	2.21E+03	2.21E+03	2.21E+03	2.21E+03	2.21E+03	2.22E+03	2.22E+03	2.22E+03	2.22E+03	2.22E+03
tc 99m	f	4.5	1.79E+08	1.79E+08	1.79E+08	1.79E+08	1.73E+08	1.51E+08	7.14E+07	1.01E+05	2.70E-02	3.72E-12	0.00E+00	0.00E+00
sr100	f	4.25	1.08E+06	3.47E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y100	f	4.25	2.13E+07	8.40E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr100	f	4.5	1.79E+08	1.63E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb100	f	4.5	1.94E+08	1.86E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb100m	f	4.25	1.60E+07	1.27E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc100	I f	4.25	5.05E+07	4.83E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr101	f	4.25	1.60E+05	4.45E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y101	f	4.5	1.05E+07	2.64E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr101	f	4.5	1.04E+08	7.52E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb101	f	4.25	1.74E+08	1.66E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo101	I f	4.25	1.83E+08	1.83E+08	4.45E+07	1.07E+07	2.35E-02	3.77E-22	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc101	I f	4.25	1.83E+08	1.83E+08	1.06E+08	4.00E+07	4.18E-01	1.18E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr102	f	4.5	2.66E+04	2.37E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y102	f	4.5	4.78E+06	2.21E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr102	f	4.5	7.20E+07	5.73E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb102	f	4.25	1.47E+08	1.12E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo102	f	4.25	1.74E+08	1.74E+08	2.77E+07	4.40E+06	2.85E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc102	f	4.25	1.74E+08	1.74E+08	2.80E+07	4.43E+06	2.87E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc102m	f	4.25	1.77E+05	1.76E+05	1.49E+03	1.25E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y103	f	4.5	1.41E+06	9.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr103	f	4.25	2.67E+07	1.57E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb103	f	4.25	1.09E+08	7.62E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo103	f	4.25	1.69E+08	1.68E+08	1.61E+00	1.51E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.1

Activity (Curies) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide		limit enr	0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
tc103	f	4.25	1.71E+08	1.71E+08	8.10E+00	7.64E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru103	f	4.25	1.72E+08	1.72E+08	1.72E+08	1.71E+08	1.70E+08	1.69E+08	1.60E+08	1.01E+08	3.50E+07	7.15E+06	2.71E+05	6.80E-01
rh103m	f	4.25	1.71E+08	1.71E+08	1.71E+08	1.71E+08	1.70E+08	1.68E+08	1.60E+08	1.01E+08	3.50E+07	7.14E+06	2.71E+05	6.79E-01
y104	f	4.25	4.13E+04	1.86E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr104	f	4.25	8.02E+06	6.10E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb104	f	4.25	5.13E+07	4.53E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo104	f	4.25	1.34E+08	1.34E+08	1.31E-01	1.21E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc104	f	4.25	1.41E+08	1.41E+08	4.80E+07	1.54E+07	1.89E+00	3.06E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh104	f	4.25	1.01E+08	9.93E+07	7.30E+04	6.06E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh104m	f	4.25	7.38E+06	7.36E+06	6.13E+04	5.08E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr105	f	4.5	2.19E+06	5.35E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb105	f	4.25	2.07E+07	1.66E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo105	f	4.25	9.86E+07	9.70E+07	6.09E-08	3.62E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc105	f	4.25	1.17E+08	1.17E+08	8.17E+06	5.28E+05	1.22E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru105	f	4.25	1.19E+08	1.19E+08	1.13E+08	1.05E+08	3.51E+07	2.89E+06	3.78E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh105	f	4.25	1.11E+08	1.11E+08	1.12E+08	1.12E+08	1.05E+08	8.02E+07	1.96E+07	9.55E+01	5.26E-11	1.74E-29	0.00E+00	0.00E+00
rh105m	f	4.25	3.38E+07	3.38E+07	3.22E+07	2.99E+07	1.00E+07	8.25E+05	1.08E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr106	f	4.25	1.12E+05	5.17E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb106	f	4.25	4.20E+06	2.13E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo106	f	4.25	5.62E+07	5.20E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc106	f	4.25	8.48E+07	8.40E+07	9.09E-08	8.02E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru106	f	4.25	6.85E+07	6.85E+07	6.85E+07	6.85E+07	6.85E+07	6.84E+07	6.80E+07	6.48E+07	5.79E+07	4.90E+07	3.47E+07	8.86E+06
rh106	f	4.25	7.38E+07	7.37E+07	6.85E+07	6.85E+07	6.85E+07	6.84E+07	6.80E+07	6.48E+07	5.79E+07	4.90E+07	3.47E+07	8.86E+06
rh106m	f	4.25	2.39E+06	2.39E+06	2.04E+06	1.74E+06	1.85E+05	1.11E+03	1.10E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb107	f	4.25	7.72E+05	3.11E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo107	f	4.25	2.33E+07	1.92E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc107	f	4.25	5.83E+07	5.71E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru107	f	4.25	6.97E+07	6.96E+07	2.98E+05	1.16E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh107	f	4.25	6.98E+07	6.98E+07	3.29E+07	1.26E+07	1.88E+01	9.09E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd107m	f	4.25	5.43E+05	5.26E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb108	f	4.25	2.41E+04	1.38E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo108	f	4.25	3.54E+06	2.23E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc108	f	4.25	2.07E+07	1.84E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru108	f	4.25	4.49E+07	4.48E+07	4.70E+05	4.86E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh108	f	4.25	4.56E+07	4.55E+07	5.00E+05	5.17E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.1

Activity (Curies) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide		limit enr	0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
rh108m	f	4.25	6.83E+05	6.82E+05	2.14E+04	6.67E+02	5.65E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo109	f	4.25	3.69E+05	2.25E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc109	f	4.25	6.89E+06	4.31E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru109	f	4.25	3.00E+07	2.95E+07	1.01E-08	3.28E-24	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh109	f	4.25	3.48E+07	3.47E+07	1.80E+01	3.03E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh109m	f	4.25	1.73E+07	1.73E+07	7.72E-04	1.12E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd109	f	4.25	4.14E+07	4.14E+07	4.04E+07	3.94E+07	2.77E+07	1.23E+07	3.22E+05	6.26E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd109m	f	4.25	2.22E+05	2.22E+05	2.64E+03	3.13E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag109m	f	4.25	4.14E+07	4.14E+07	4.04E+07	3.94E+07	2.77E+07	1.23E+07	3.22E+05	7.79E-02	7.12E-02	6.22E-02	4.71E-02	1.57E-02
mo110	f	4.25	3.80E+04	2.96E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc110	f	4.25	1.11E+06	5.00E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru110	f	4.25	9.78E+06	9.40E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh110	f	4.25	1.56E+06	1.25E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh110m	f	4.25	1.14E+07	1.13E+07	2.19E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag110	f	4.25	1.87E+07	1.82E+07	6.23E+03	6.23E+03	6.23E+03	6.22E+03	6.17E+03	5.74E+03	4.86E+03	3.78E+03	2.26E+03	2.98E+02
ag110m	f	4.25	4.58E+05	4.58E+05	4.58E+05	4.58E+05	4.58E+05	4.57E+05	4.53E+05	4.22E+05	3.57E+05	2.78E+05	1.67E+05	2.19E+04
mo111	f	4.25	3.11E+03	7.01E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc111	f	4.25	2.11E+05	1.49E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru111	f	4.25	3.27E+06	2.18E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh111	f	4.25	6.24E+06	6.03E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd111	f	4.25	6.66E+06	6.66E+06	2.88E+06	1.29E+06	8.10E+04	1.08E+04	1.24E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd111m	f	4.25	2.83E+05	2.83E+05	2.65E+05	2.49E+05	1.03E+05	1.38E+04	1.57E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag111	f	4.25	6.72E+06	6.72E+06	6.71E+06	6.70E+06	6.52E+06	6.13E+06	4.65E+06	4.13E+05	1.56E+03	3.59E-01	1.18E-08	0.00E+00
ag111m	f	4.25	6.71E+06	6.71E+06	3.04E+06	1.39E+06	1.01E+05	1.34E+04	1.54E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd111m	f	4.25	2.50E+04	2.50E+04	1.63E+04	1.06E+04	2.66E+01	3.01E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc112	f	4.25	3.21E+04	6.59E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru112	f	4.25	1.05E+06	8.71E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh112	f	4.25	2.47E+06	1.90E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd112	f	4.25	3.02E+06	3.02E+06	2.97E+06	2.92E+06	2.32E+06	1.37E+06	1.28E+05	1.51E-04	3.82E-25	0.00E+00	0.00E+00	0.00E+00
ag112	f	4.25	3.03E+06	3.03E+06	3.03E+06	3.02E+06	2.64E+06	1.60E+06	1.51E+05	1.78E-04	4.49E-25	0.00E+00	0.00E+00	0.00E+00
in113m	l	4.25	3.47E+05	3.47E+05	3.47E+05	3.47E+05	3.46E+05	3.45E+05	3.39E+05	2.90E+05	2.02E+05	1.18E+05	3.85E+04	4.73E+02
sn113	l	4.25	3.47E+05	3.47E+05	3.47E+05	3.47E+05	3.46E+05	3.45E+05	3.38E+05	2.90E+05	2.02E+05	1.18E+05	3.84E+04	4.72E+02
sn113m	l	4.25	1.13E+05	1.13E+05	4.27E+04	1.62E+04	2.00E-02	6.26E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc113	f	4.25	6.77E+03	2.34E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru113	f	4.25	3.25E+05	2.59E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.1

Activity (Curies) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide		limit enr	0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
rh113	f	4.25	1.18E+06	7.02E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd113	f	4.25	1.74E+06	1.73E+06	2.64E+00	3.91E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag113	f	4.25	1.69E+06	1.69E+06	1.60E+06	1.50E+06	6.05E+05	7.64E+04	7.05E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag113m	f	4.25	3.38E+05	3.38E+05	1.86E+00	2.77E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd113m	f	4.25	4.33E+03	4.33E+03	4.33E+03	4.33E+03	4.33E+03	4.33E+03	4.33E+03	4.32E+03	4.29E+03	4.23E+03	4.13E+03	3.74E+03
in114	l	4.25	5.63E+04	5.62E+04	3.54E+04	3.54E+04	3.53E+04	3.50E+04	3.35E+04	2.33E+04	1.01E+04	2.86E+03	2.13E+02	7.72E-03
in114m	l	4.25	3.71E+04	3.71E+04	3.71E+04	3.71E+04	3.69E+04	3.65E+04	3.51E+04	2.44E+04	1.05E+04	2.98E+03	2.23E+02	8.10E-03
ru114	f	4.25	1.08E+05	9.93E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh114	f	4.25	5.75E+05	4.17E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd114	f	4.25	1.32E+06	1.31E+06	2.75E+02	5.65E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag114	f	4.25	1.37E+06	1.36E+06	2.83E+02	5.83E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru115	f	4.25	2.32E+04	1.05E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh115	f	4.25	2.30E+05	2.13E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd115	f	4.25	7.95E+05	7.79E+05	4.74E-09	2.58E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag115	f	4.25	6.26E+05	6.26E+05	2.28E+05	8.10E+04	3.85E-02	1.37E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag115m	f	4.25	2.62E+05	2.60E+05	2.43E-09	1.32E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd115	l f	4.25	9.02E+05	9.02E+05	8.94E+05	8.94E+05	8.17E+05	6.62E+05	2.61E+05	8.02E+01	6.26E-07	4.32E-19	0.00E+00	0.00E+00
cd115m	f	4.25	4.20E+04	4.20E+04	4.19E+04	4.19E+04	4.18E+04	4.13E+04	3.95E+04	2.64E+04	1.04E+04	2.56E+03	1.44E+02	1.68E-03
in115m	f	4.25	9.02E+05	9.02E+05	9.02E+05	9.02E+05	8.63E+05	7.21E+05	2.84E+05	9.02E+01	1.15E+00	2.83E-01	1.59E-02	1.86E-07
ru116	f	4.25	7.14E+03	4.74E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh116	f	4.25	1.08E+05	5.51E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd116	f	4.25	8.40E+05	8.02E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag116	f	4.25	9.63E+05	9.63E+05	4.42E+02	1.89E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag116m	f	4.25	1.21E+05	1.14E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in116	l f	4.25	1.61E+05	1.54E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in116m	f	4.25	6.06E+05	6.06E+05	4.13E+05	2.81E+05	1.30E+03	5.99E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh117	f	4.25	4.20E+04	2.39E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd117	f	4.25	5.69E+05	4.99E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag117	f	4.25	5.07E+05	5.05E+05	1.91E-02	6.85E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag117m	f	4.25	5.07E+05	4.78E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd117	f	4.25	8.86E+05	8.86E+05	7.79E+05	6.76E+05	9.63E+04	1.12E+03	2.22E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd117m	f	4.25	2.03E+05	2.03E+05	1.83E+05	1.66E+05	3.90E+04	1.44E+03	5.13E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in117	l f	4.25	6.62E+05	6.62E+05	6.56E+05	6.38E+05	2.02E+05	4.12E+03	6.63E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in117m	f	4.25	8.10E+05	8.10E+05	8.02E+05	7.79E+05	2.34E+05	4.13E+03	2.74E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn117m	l f	4.25	2.49E+06	2.49E+06	2.49E+06	2.48E+06	2.44E+06	2.36E+06	2.03E+06	5.39E+05	2.53E+04	2.58E+02	2.04E-02	1.38E-18

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Table 3.1

Activity (Curies) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide		limit enr	0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
rh118	f	4.25	9.93E+03	1.18E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd118	f	4.25	2.73E+05	2.19E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag118	f	4.25	4.68E+05	4.31E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag118m	f	4.25	3.32E+05	2.70E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd118	f	4.25	8.17E+05	8.17E+05	5.43E+05	3.59E+05	1.10E+03	1.98E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in118	l f	4.25	8.18E+05	8.18E+05	5.44E+05	3.60E+05	1.10E+03	1.98E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh119	f	4.25	3.15E+03	7.11E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd119	f	4.25	1.28E+05	8.63E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag119	f	4.25	5.10E+05	3.95E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd119	f	4.25	5.65E+05	5.64E+05	2.51E+02	1.10E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd119m	f	4.25	2.80E+05	2.78E+05	2.22E+01	1.73E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in119	l f	4.25	3.64E+05	3.64E+05	5.97E+03	1.73E+03	1.63E-04	1.44E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in119m	f	4.25	5.12E+05	5.12E+05	1.89E+05	5.97E+04	5.65E-03	4.98E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn119m	l f	4.25	2.46E+06	2.46E+06	2.46E+06	2.46E+06	2.46E+06	2.45E+06	2.43E+06	2.29E+06	1.99E+06	1.60E+06	1.04E+06	1.84E+05
pd120	f	4.25	7.52E+04	6.30E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag120	f	4.25	3.57E+05	2.28E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd120	f	4.25	8.02E+05	7.95E+05	1.76E-05	3.77E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in120	f	4.25	8.17E+05	8.10E+05	1.88E-05	4.02E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in120m	f	4.25	1.44E+04	1.42E+04	2.70E-08	5.00E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd121	f	4.25	3.06E+04	1.05E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag121	f	4.25	2.52E+05	1.16E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd121	f	4.25	7.95E+05	7.64E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in121	f	4.25	7.72E+04	7.51E+04	5.40E+01	2.54E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in121m	f	4.25	8.10E+05	8.10E+05	4.06E+03	1.90E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn121	l f	4.25	1.54E+06	1.54E+06	1.52E+06	1.51E+06	1.26E+06	8.35E+05	1.32E+05	4.17E+02	4.16E+02	4.16E+02	4.12E+02	4.03E+02
sn121m	l	4.25	1.92E+02	1.92E+02	1.92E+02	1.92E+02	1.92E+02	1.92E+02	1.92E+02	1.92E+02	1.91E+02	1.91E+02	1.89E+02	1.85E+02
pd122	f	4.25	9.93E+03	6.09E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag122	f	4.25	1.30E+05	3.61E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd122	f	4.25	8.48E+05	7.55E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in122	f	4.25	9.47E+05	8.94E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in122m	f	4.25	9.86E+04	9.24E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb122	l f	4.25	2.13E+05	2.13E+05	2.12E+05	2.12E+05	1.96E+05	1.65E+05	7.65E+04	9.66E+01	1.98E-05	1.83E-15	0.00E+00	0.00E+00
sb122m	l f	4.25	1.28E+04	1.27E+04	9.16E+01	6.53E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te123m	l	4.25	9.70E+02	9.70E+02	9.70E+02	9.70E+02	9.70E+02	9.63E+02	9.47E+02	8.17E+02	5.76E+02	3.42E+02	1.17E+02	1.70E+00
ag123	f	4.25	5.14E+04	9.09E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

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Table 3.1

Activity (Curies) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide		limit enr	0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
cd123	f	4.25	5.51E+05	5.11E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in123	f	4.25	7.07E+05	6.75E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in123m	f	4.25	1.93E+05	1.92E+05	1.03E-06	4.71E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn123	l f	4.25	1.36E+05	1.36E+05	1.36E+05	1.36E+05	1.36E+05	1.35E+05	1.33E+05	1.16E+05	8.40E+04	5.18E+04	1.92E+04	3.80E+02
sn123m	l f	4.25	9.25E+05	9.25E+05	5.53E+05	3.29E+05	2.31E+02	1.42E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag124	f	4.25	3.90E+04	2.76E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd124	f	4.25	8.33E+05	3.91E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in124	f	4.25	1.57E+06	1.38E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb124	l f	4.25	9.72E+04	9.72E+04	9.72E+04	9.72E+04	9.72E+04	9.64E+04	9.31E+04	6.89E+04	3.45E+04	1.22E+04	1.45E+03	3.22E-01
sb124m	l f	4.25	1.93E+03	1.91E+03	2.88E-03	4.28E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag125	f	4.25	1.65E+04	2.08E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd125	f	4.25	4.98E+05	3.20E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in125	f	4.25	8.86E+05	7.27E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in125m	f	4.25	6.84E+05	6.52E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn125	l f	4.25	5.57E+05	5.57E+05	5.56E+05	5.55E+05	5.44E+05	5.18E+05	4.18E+05	6.45E+04	8.61E+02	1.33E+00	2.19E-06	0.00E+00
sn125m	l f	4.25	2.07E+06	2.07E+06	2.35E+05	2.65E+04	1.39E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb125	l f	4.25	1.37E+06	1.37E+06	1.37E+06	1.37E+06	1.37E+06	1.37E+06	1.37E+06	1.35E+06	1.29E+06	1.21E+06	1.07E+06	6.43E+05
te125m	l f	4.25	3.03E+05	3.03E+05	3.03E+05	3.03E+05	3.03E+05	3.03E+05	3.04E+05	3.06E+05	3.04E+05	2.93E+05	2.61E+05	1.57E+05
ag126	f	4.25	7.72E+03	5.97E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd126	f	4.25	6.10E+05	1.55E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in126	f	4.25	2.12E+06	1.31E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb126	l f	4.25	4.64E+04	4.64E+04	4.64E+04	4.64E+04	4.56E+04	4.39E+04	3.71E+04	8.68E+03	3.15E+02	1.36E+01	1.16E+01	1.16E+01
sb126m	l f	4.25	5.45E+04	5.45E+04	1.83E+04	6.18E+03	8.33E+01	8.33E+01	8.33E+01	8.33E+01	8.33E+01	8.33E+01	8.33E+01	8.33E+01
ag127	f	4.25	4.29E+03	8.17E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd127	f	4.25	4.74E+05	1.41E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in127	f	4.25	1.93E+06	1.11E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in127m	f	4.25	1.93E+06	1.62E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn127	f	4.25	3.78E+06	3.78E+06	3.21E+06	2.72E+06	2.70E+05	1.38E+03	6.56E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn127m	f	4.25	5.10E+06	5.10E+06	3.35E+04	2.17E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb127	f	4.25	9.40E+06	9.40E+06	9.40E+06	9.32E+06	8.94E+06	7.95E+06	4.61E+06	4.28E+04	8.71E-01	7.95E-08	2.60E-22	0.00E+00
te127	l f	4.25	9.32E+06	9.32E+06	9.32E+06	9.32E+06	9.24E+06	8.63E+06	5.78E+06	1.38E+06	9.09E+05	5.13E+05	1.58E+05	1.52E+03
te127m	f	4.25	1.59E+06	1.59E+06	1.59E+06	1.59E+06	1.59E+06	1.59E+06	1.57E+06	1.36E+06	9.32E+05	5.25E+05	1.61E+05	1.55E+03
ag128	f	4.25	1.94E+03	1.22E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd128	f	4.25	4.12E+05	2.13E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in128	f	4.25	3.32E+06	1.69E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

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Table 3.1

Activity (Curies) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide		limit enr	0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
sn128	f	4.25	1.51E+07	1.51E+07	1.07E+07	7.50E+06	5.44E+04	7.01E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb128	f	4.25	1.61E+06	1.61E+06	1.57E+06	1.53E+06	9.17E+05	2.67E+05	1.05E+03	1.50E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb128m	f	4.25	1.60E+07	1.60E+07	1.26E+07	9.09E+06	6.61E+04	8.48E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd129	f	4.25	2.02E+05	1.98E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in129	f	4.25	3.81E+06	1.22E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn129	f	4.25	1.35E+07	1.34E+07	8.94E+02	1.35E-01	1.03E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn129m	f	4.5	1.29E+07	1.29E+07	5.80E+05	2.60E+04	3.51E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb129	f	4.25	3.47E+07	3.47E+07	3.25E+07	3.00E+07	9.93E+06	8.02E+05	9.47E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te129	f	4.25	3.29E+07	3.29E+07	3.26E+07	3.19E+07	1.53E+07	5.10E+06	3.95E+06	2.31E+06	6.71E+05	1.05E+05	2.29E+03	6.54E-04
te129m	f	4.25	6.66E+06	6.66E+06	6.66E+06	6.66E+06	6.64E+06	6.56E+06	6.17E+06	3.61E+06	1.05E+06	1.63E+05	3.58E+03	1.02E-03
xe129m	f	4.25	5.16E+03	5.16E+03	5.15E+03	5.14E+03	5.03E+03	4.78E+03	3.77E+03	4.97E+02	4.62E+00	4.15E-03	2.21E-09	0.00E+00
cd130	f	4.25	7.41E+04	1.73E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in130	f	4.5	2.74E+06	3.38E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn130	f	4.5	3.61E+07	3.59E+07	1.34E+05	5.02E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb130	f	4.25	1.15E+07	1.15E+07	6.82E+06	4.03E+06	2.54E+03	1.22E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb130m	f	4.5	4.82E+07	4.81E+07	3.50E+06	1.35E+05	1.16E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i130	f	4.25	2.64E+06	2.64E+06	2.58E+06	2.51E+06	1.70E+06	6.91E+05	1.22E+04	7.79E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i130m	f	4.25	1.41E+06	1.41E+06	1.40E+05	1.38E+04	1.24E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd131	f	4.25	1.15E+04	1.65E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in131	f	4.5	1.17E+06	9.02E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn131	f	4.5	3.07E+07	3.02E+07	3.94E-07	4.99E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb131	f	4.5	8.40E+07	8.40E+07	3.45E+07	1.40E+07	4.45E+01	1.21E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te131	f	4.5	9.09E+07	9.09E+07	6.98E+07	4.42E+07	4.03E+06	2.79E+06	5.28E+05	2.90E-01	1.02E-15	0.00E+00	0.00E+00	0.00E+00
te131m	f	4.25	2.15E+07	2.15E+07	2.12E+07	2.11E+07	1.80E+07	1.24E+07	2.35E+06	1.28E+00	4.56E-15	0.00E+00	0.00E+00	0.00E+00
i131	f	4.5	1.07E+08	1.07E+08	1.07E+08	1.07E+08	1.05E+08	1.00E+08	7.87E+07	8.40E+06	4.76E+04	2.03E+01	2.35E-06	0.00E+00
xe131m	f	4.25	1.47E+06	1.47E+06	1.47E+06	1.47E+06	1.47E+06	1.45E+06	1.38E+06	5.05E+05	2.00E+04	1.11E+02	2.29E-03	7.54E-22
in132	f	4.25	3.07E+05	7.38E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn132	f	4.25	2.45E+07	2.41E+07	7.01E-07	1.97E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb132	f	4.25	5.08E+07	5.07E+07	3.60E+05	2.54E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb132m	f	4.5	4.81E+07	4.80E+07	3.32E+04	1.97E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te132	f	4.5	1.54E+08	1.54E+08	1.54E+08	1.53E+08	1.44E+08	1.25E+08	6.59E+07	2.61E+05	7.46E-01	3.61E-09	2.76E-26	0.00E+00
i132	f	4.25	1.57E+08	1.57E+08	1.57E+08	1.56E+08	1.48E+08	1.28E+08	6.78E+07	2.69E+05	7.72E-01	3.71E-09	2.84E-26	0.00E+00
cs132	f	4.25	4.35E+03	4.35E+03	4.35E+03	4.33E+03	4.20E+03	3.91E+03	2.83E+03	1.76E+02	2.87E-01	1.89E-05	4.67E-14	0.00E+00
in133	f	4.25	9.63E+03	1.91E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn133	f	4.25	6.62E+06	4.09E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

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Table 3.1

Activity (Curies) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide		limit enr	0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
sb133	f	4.5	6.96E+07	6.94E+07	1.70E+04	4.15E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te133	f	4.5	1.20E+08	1.20E+08	3.66E+07	1.46E+07	5.53E+04	3.36E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te133m	f	4.5	9.86E+07	9.86E+07	6.82E+07	4.69E+07	2.44E+05	1.49E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i133	f	4.5	2.22E+08	2.22E+08	2.20E+08	2.18E+08	1.74E+08	1.02E+08	9.24E+06	8.63E-03	1.25E-23	0.00E+00	0.00E+00	0.00E+00
i133m	f	4.25	1.70E+07	1.64E+07	6.98E+06	4.80E+06	2.51E+04	1.52E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe133	f	4.5	2.12E+08	2.12E+08	2.12E+08	2.12E+08	2.12E+08	2.06E+08	1.51E+08	4.99E+06	1.80E+03	1.22E-02	2.82E-13	0.00E+00
xe133m	f	4.25	6.99E+06	6.99E+06	6.99E+06	6.98E+06	6.86E+06	6.29E+06	3.00E+06	8.48E+02	4.78E-06	2.03E-18	0.00E+00	0.00E+00
sn134	f	4.25	1.10E+06	5.66E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb134	f	4.5	1.26E+07	5.92E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb134m	f	4.5	9.32E+06	8.71E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te134	f	4.5	1.95E+08	1.95E+08	1.18E+08	7.21E+07	6.81E+04	8.33E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i134	f	4.5	2.45E+08	2.45E+08	2.15E+08	1.75E+08	1.53E+06	5.71E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i134m	f	4.25	2.17E+07	2.16E+07	7.72E+04	2.76E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe134m	f	4.25	5.90E+06	9.93E+05	1.79E+03	6.36E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs134	f	4.25	2.30E+07	2.30E+07	2.30E+07	2.30E+07	2.30E+07	2.30E+07	2.29E+07	2.24E+07	2.12E+07	1.95E+07	1.64E+07	8.40E+06
cs134m	f	4.25	4.81E+06	4.81E+06	4.27E+06	3.79E+06	7.16E+05	1.59E+04	5.69E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn135	f	4.25	9.40E+04	1.78E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb135	f	4.25	5.84E+06	3.90E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te135	f	4.5	1.06E+08	1.02E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i135	f	4.5	2.11E+08	2.11E+08	2.00E+08	1.90E+08	9.09E+07	1.68E+07	8.40E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe135	f	4.5	7.03E+07	7.03E+07	7.56E+07	8.02E+07	1.01E+08	5.62E+07	4.00E+05	1.18E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe135m	f	4.25	4.63E+07	4.63E+07	3.58E+07	3.18E+07	1.48E+07	2.74E+06	1.38E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs135m	f	4.25	4.65E+06	4.65E+06	3.14E+06	2.12E+06	8.71E+03	3.08E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba135m	f	4.25	4.19E+04	4.19E+04	4.13E+04	4.09E+04	3.45E+04	2.35E+04	4.12E+03	1.17E-03	9.17E-19	0.00E+00	0.00E+00	0.00E+00
sn136	f	4.25	8.10E+03	3.07E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb136	f	4.25	9.02E+05	3.89E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te136	f	4.5	4.70E+07	4.52E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i136	f	4.5	9.63E+07	9.55E+07	3.46E+01	1.10E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i136m	f	4.5	4.71E+07	4.65E+07	1.32E-04	3.67E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs136	f	4.5	7.34E+06	7.34E+06	7.33E+06	7.33E+06	7.21E+06	6.96E+06	5.94E+06	1.51E+06	6.41E+04	5.60E+02	3.24E-02	6.31E-19
ba136m	f	4.5	8.40E+05	8.25E+05	8.17E+05	8.17E+05	8.10E+05	7.79E+05	6.66E+05	1.70E+05	7.18E+03	6.27E+01	3.63E-03	7.07E-20
sb137	f	4.5	8.40E+05	1.97E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te137	f	4.5	1.57E+07	1.29E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i137	f	4.5	1.03E+08	1.01E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe137	f	4.5	2.02E+08	2.02E+08	9.24E+05	3.97E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

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Table 3.1

Activity (Curies) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide		limit enr	0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
cs137	f	4.5	1.73E+07	1.73E+07	1.73E+07	1.73E+07	1.73E+07	1.73E+07	1.73E+07	1.73E+07	1.73E+07	1.72E+07	1.70E+07	1.62E+07
ba137m	f	4.5	1.65E+07	1.65E+07	1.64E+07	1.64E+07	1.64E+07	1.64E+07	1.64E+07	1.63E+07	1.63E+07	1.62E+07	1.60E+07	1.53E+07
sb138	f	4.25	1.21E+04	2.22E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te138	f	4.25	3.81E+06	2.32E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i138	f	4.5	5.23E+07	4.73E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe138	f	4.5	1.89E+08	1.89E+08	4.32E+07	9.86E+06	1.03E-02	3.09E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs138	f	4.5	2.05E+08	2.05E+08	1.51E+08	8.94E+07	1.15E+04	1.21E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs138m	f	4.25	8.79E+06	8.71E+06	6.91E+03	5.43E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te139	f	4.25	5.87E+05	1.77E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i139	f	4.5	2.49E+07	1.85E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe139	f	4.5	1.40E+08	1.38E+08	3.13E-06	6.85E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs139	f	4.5	1.90E+08	1.90E+08	2.13E+07	2.26E+06	5.20E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba139	f	4.5	1.95E+08	1.95E+08	1.70E+08	1.34E+08	4.32E+06	1.66E+03	7.15E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te140	f	4.25	8.40E+04	3.87E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i140	f	4.5	6.58E+06	2.96E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe140	f	4.5	9.70E+07	9.24E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs140	f	4.5	1.70E+08	1.70E+08	6.16E-01	1.91E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba140	f	4.5	1.96E+08	1.96E+08	1.96E+08	1.96E+08	1.93E+08	1.86E+08	1.57E+08	3.84E+07	1.47E+06	1.11E+04	4.69E-01	2.69E-18
la140	f	4.5	2.09E+08	2.09E+08	2.09E+08	2.09E+08	2.08E+08	2.03E+08	1.78E+08	4.42E+07	1.70E+06	1.28E+04	5.40E-01	3.10E-18
pr140	f	4.25	3.19E+03	3.19E+03	6.93E+00	1.50E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i141	f	4.25	8.48E+05	1.87E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe141	f	4.5	3.71E+07	2.49E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs141	f	4.5	1.30E+08	1.28E+08	2.51E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba141	f	4.5	1.76E+08	1.76E+08	5.76E+07	1.84E+07	2.21E+00	3.34E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la141	f	4.5	1.78E+08	1.78E+08	1.72E+08	1.60E+08	4.70E+07	2.77E+06	8.17E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce141	f	4.5	1.80E+08	1.80E+08	1.80E+08	1.80E+08	1.79E+08	1.76E+08	1.66E+08	9.55E+07	2.64E+07	3.88E+06	7.46E+04	1.28E-02
i142	f	4.25	2.58E+05	8.17E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe142	f	4.5	1.46E+07	8.25E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs142	f	4.5	7.54E+07	5.38E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba142	f	4.5	1.68E+08	1.68E+08	2.37E+07	3.32E+06	3.93E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la142	f	4.5	1.74E+08	1.74E+08	1.54E+08	1.24E+08	5.10E+06	3.42E+03	1.82E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr142	f	4.25	7.33E+06	7.33E+06	7.20E+06	7.07E+06	5.49E+06	3.07E+06	2.26E+05	3.38E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i143	f	4.25	2.61E+03	4.61E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe143	f	4.25	2.19E+06	1.06E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs143	f	4.5	3.80E+07	2.63E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.1

Activity (Curies) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide		limit enr	0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
ba143	f	4.5	1.44E+08	1.38E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la143	f	4.5	1.65E+08	1.65E+08	3.86E+07	8.86E+06	1.02E-02	3.71E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce143	f	4.5	1.67E+08	1.67E+08	1.66E+08	1.64E+08	1.42E+08	1.02E+08	2.24E+07	4.55E+01	3.32E-12	0.00E+00	0.00E+00	0.00E+00
pr143	f	4.5	1.61E+08	1.61E+08	1.61E+08	1.61E+08	1.61E+08	1.60E+08	1.44E+08	3.90E+07	1.82E+06	1.83E+04	1.41E+00	8.71E-17
xe144	f	4.25	4.58E+05	2.44E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs144	f	4.5	1.12E+07	5.79E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba144	f	4.5	1.12E+08	1.05E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la144	f	4.5	1.46E+08	1.45E+08	1.08E-05	6.01E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce144	f	4.5	1.51E+08	1.51E+08	1.51E+08	1.51E+08	1.51E+08	1.51E+08	1.50E+08	1.41E+08	1.21E+08	9.78E+07	6.22E+07	1.05E+07
pr144	f	4.5	1.52E+08	1.52E+08	1.51E+08	1.51E+08	1.51E+08	1.51E+08	1.50E+08	1.41E+08	1.21E+08	9.78E+07	6.22E+07	1.05E+07
pr144m	f	4.5	2.12E+06	2.12E+06	2.12E+06	2.12E+06	2.12E+06	2.11E+06	2.09E+06	1.97E+06	1.70E+06	1.37E+06	8.71E+05	1.47E+05
xe145	f	4.25	4.57E+04	2.11E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs145	f	4.25	2.70E+06	8.63E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba145	f	4.5	4.97E+07	4.25E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la145	f	4.5	1.02E+08	1.00E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce145	f	4.5	1.13E+08	1.13E+08	1.31E+05	1.31E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr145	f	4.5	1.13E+08	1.13E+08	1.08E+08	1.02E+08	4.53E+07	7.11E+06	1.70E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe146	f	4.25	4.03E+03	1.18E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs146	f	4.5	5.78E+05	7.79E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba146	f	4.5	2.54E+07	1.86E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la146	f	4.5	6.59E+07	6.13E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce146	f	4.5	9.02E+07	9.02E+07	1.95E+07	4.19E+06	1.86E-03	7.87E-25	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr146	f	4.5	9.09E+07	9.09E+07	6.25E+07	3.16E+07	2.15E+02	2.31E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs147	f	4.25	1.28E+04	3.58E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba147	f	4.25	4.26E+06	1.58E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la147	f	4.5	2.87E+07	2.48E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce147	f	4.5	6.81E+07	6.76E+07	1.75E-02	4.29E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr147	f	4.5	7.17E+07	7.17E+07	1.67E+07	3.61E+06	1.83E-03	1.03E-24	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd147	f	4.5	7.24E+07	7.24E+07	7.23E+07	7.22E+07	7.09E+07	6.80E+07	5.62E+07	1.09E+07	2.47E+05	8.40E+02	7.02E-03	6.59E-23
pm147	f	4.5	2.51E+07	2.51E+07	2.51E+07	2.51E+07	2.51E+07	2.51E+07	2.52E+07	2.52E+07	2.43E+07	2.28E+07	1.99E+07	1.18E+07
cs148	f	4.25	2.33E+03	7.95E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba148	f	4.25	8.33E+05	2.66E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la148	f	4.25	9.09E+06	4.90E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce148	f	4.5	4.82E+07	4.77E+07	1.02E-02	2.15E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr148	f	4.5	5.57E+07	5.56E+07	9.40E+03	9.86E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.1

Activity (Curies) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide		limit enr	0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
pm148	f	4.25	2.02E+07	2.02E+07	2.02E+07	2.01E+07	1.94E+07	1.78E+07	1.21E+07	5.38E+05	4.55E+04	1.00E+04	4.46E+02	2.11E-03
pm148m	f	4.5	3.88E+06	3.88E+06	3.88E+06	3.87E+06	3.86E+06	3.81E+06	3.63E+06	2.35E+06	8.56E+05	1.89E+05	8.40E+03	4.00E-02
ba149	f	4.25	9.17E+04	3.38E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la149	f	4.25	2.59E+06	1.96E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce149	f	4.5	2.51E+07	2.22E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr149	f	4.25	3.85E+07	3.84E+07	4.00E+03	4.03E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd149	f	4.25	4.16E+07	4.16E+07	3.47E+07	2.84E+07	1.70E+06	2.75E+03	7.49E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm149	f	4.25	6.46E+07	6.46E+07	6.44E+07	6.42E+07	5.94E+07	4.82E+07	1.89E+07	5.45E+03	3.72E-05	2.09E-17	0.00E+00	0.00E+00
ba150	f	4.25	8.25E+03	4.00E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la150	f	4.25	4.38E+05	1.44E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce150	f	4.25	1.14E+07	9.63E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr150	f	4.25	2.43E+07	2.28E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm150	f	4.25	5.53E+05	5.53E+05	4.87E+05	4.27E+05	6.99E+04	1.12E+03	9.09E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la151	f	4.25	7.27E+04	2.77E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce151	f	4.25	3.28E+06	1.68E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr151	f	4.25	1.28E+07	1.25E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd151	f	4.25	2.15E+07	2.15E+07	4.10E+06	7.72E+05	5.29E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm151	f	4.25	2.17E+07	2.17E+07	2.15E+07	2.13E+07	1.80E+07	1.21E+07	2.10E+06	5.07E-01	2.73E-16	0.00E+00	0.00E+00	0.00E+00
sm151	f	4.5	6.80E+04	6.80E+04	6.80E+04	6.80E+04	6.81E+04	6.83E+04	6.87E+04	6.87E+04	6.86E+04	6.85E+04	6.82E+04	6.72E+04
ce152	f	4.25	4.26E+05	3.89E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr152	f	4.25	4.35E+06	3.96E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd152	f	4.25	1.44E+07	1.44E+07	2.33E+06	3.76E+05	3.06E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm152	f	4.25	1.50E+07	1.50E+07	3.60E+06	5.88E+05	4.77E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm152m	f	4.25	5.37E+05	5.36E+05	3.38E+04	2.13E+03	3.28E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
eu152m	f	4.5	2.48E+04	2.48E+04	2.38E+04	2.30E+04	1.37E+04	4.15E+03	1.96E+01	1.37E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce153	f	4.25	1.47E+05	9.17E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr153	f	4.25	1.82E+06	1.57E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd153	f	4.25	8.25E+06	8.17E+06	7.95E-02	7.39E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm153	f	4.25	9.93E+06	9.93E+06	2.59E+05	5.50E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sm153	l f	4.25	5.31E+07	5.31E+07	5.27E+07	5.23E+07	4.71E+07	3.71E+07	1.26E+07	1.10E+03	4.73E-07	4.21E-21	0.00E+00	0.00E+00
gd153	l f	4.25	8.13E+05	8.13E+05	8.13E+05	8.13E+05	8.13E+05	8.13E+05	8.05E+05	7.49E+05	6.30E+05	4.87E+05	2.86E+05	3.52E+04
ce154	f	4.25	1.51E+04	1.06E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr154	f	4.25	3.58E+05	1.92E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd154	f	4.25	4.19E+06	4.12E+06	1.20E-07	3.37E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm154	f	4.25	5.25E+06	5.24E+06	4.45E+01	2.49E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.1

Activity (Curies) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide		limit enr	0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
pm154m	f	4.25	1.06E+06	1.05E+06	4.54E+02	1.93E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
eu154	l f	4.25	1.06E+06	1.06E+06	1.06E+06	1.06E+06	1.06E+06	1.06E+06	1.06E+06	1.06E+06	1.04E+06	1.02E+06	9.77E+05	8.29E+05
gd155m	l	4.25	1.60E+04	2.96E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr155	f	4.25	7.72E+04	4.19E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd155	f	4.25	1.47E+06	1.41E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm155	f	4.25	3.32E+06	3.29E+06	2.18E-05	1.12E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sm155	l f	4.25	4.10E+06	4.10E+06	1.67E+06	6.58E+05	1.41E+00	1.54E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
eu155	l f	4.25	4.34E+05	4.34E+05	4.34E+05	4.34E+05	4.34E+05	4.34E+05	4.34E+05	4.29E+05	4.18E+05	4.04E+05	3.75E+05	2.78E+05
pr156	f	4.25	1.12E+04	1.84E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd156	f	4.25	5.38E+05	5.20E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm156	f	4.25	1.70E+06	1.64E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sm156	f	4.25	2.57E+06	2.57E+06	2.48E+06	2.38E+06	1.42E+06	4.38E+05	2.16E+03	2.25E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00
eu156	l f	4.25	2.74E+07	2.74E+07	2.74E+07	2.73E+07	2.70E+07	2.62E+07	2.29E+07	6.98E+06	4.51E+05	7.41E+03	1.58E+00	5.20E-15
nd157	f	4.25	1.36E+05	1.02E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm157	f	4.25	7.87E+05	7.79E+05	1.10E-03	1.52E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sm157	f	4.25	1.62E+06	1.62E+06	1.32E+05	1.00E+04	2.15E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
eu157	f	4.25	2.63E+06	2.63E+06	2.59E+06	2.53E+06	1.83E+06	8.86E+05	3.31E+04	1.40E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd158	f	4.25	2.23E+04	1.73E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm158	f	4.25	2.15E+05	1.83E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sm158	f	4.25	8.48E+05	8.48E+05	1.95E+04	4.48E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
eu158	f	4.25	9.47E+05	9.47E+05	6.74E+05	4.30E+05	7.57E+02	3.83E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd159	f	4.25	1.96E+03	6.62E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm159	f	4.25	5.00E+04	3.99E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sm159	f	4.25	3.52E+05	3.51E+05	1.60E+02	7.20E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
eu159	f	4.25	4.81E+05	4.81E+05	1.72E+05	5.45E+04	5.64E-03	6.10E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
gd159	l f	4.25	2.74E+07	2.74E+07	2.69E+07	2.65E+07	2.04E+07	1.12E+07	7.61E+05	5.77E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm160	f	4.25	5.54E+03	2.18E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sm160	f	4.25	1.05E+05	1.04E+05	3.61E-03	1.23E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
eu160	f	4.25	1.93E+05	1.92E+05	9.17E-03	3.12E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tb160	l f	4.25	7.20E+06	7.20E+06	7.19E+06	7.19E+06	7.17E+06	7.13E+06	6.92E+06	5.40E+06	3.03E+06	1.28E+06	2.17E+05	1.97E+02
sm161	f	4.25	2.28E+04	1.99E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
eu161	f	4.25	7.44E+04	7.35E+04	1.01E-08	1.30E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
gd161	l f	4.25	3.27E+06	3.25E+06	1.12E+04	3.81E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tb161	l f	4.25	4.33E+06	4.33E+06	4.32E+06	4.31E+06	4.19E+06	3.92E+06	2.90E+06	2.13E+05	5.14E+02	6.09E-02	5.04E-10	0.00E+00
sm162	f	4.25	3.19E+03	2.80E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.1

Activity (Curies) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide		limit enr	0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
eu162	f	4.25	1.84E+04	1.83E+04	8.56E+00	3.92E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
gd162	l f	4.25	4.22E+04	4.21E+04	4.28E+03	3.60E+02	3.20E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tb162	l f	4.25	4.29E+04	4.29E+04	1.34E+04	1.79E+03	4.06E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
eu163	f	4.25	3.67E+03	3.36E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
gd163	f	4.25	1.47E+04	1.46E+04	2.18E-02	3.12E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tb163	f	4.25	1.68E+04	1.68E+04	6.24E+03	2.15E+03	7.05E-04	1.07E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
gd164	f	4.25	4.36E+03	4.36E+03	1.67E+03	6.41E+02	9.47E-04	4.45E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tb164	f	4.25	6.03E+03	6.02E+03	1.94E+03	7.44E+02	1.10E-03	5.16E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tb165	f	4.25	2.05E+03	2.04E+03	1.33E-01	6.94E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
dy165	l f	4.25	5.38E+05	5.38E+05	4.66E+05	4.02E+05	5.03E+04	4.34E+02	2.25E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
dy165m	l f	4.25	3.46E+05	3.43E+05	3.05E-01	1.48E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
dy166	l	4.25	2.09E+03	2.09E+03	2.09E+03	2.08E+03	1.96E+03	1.71E+03	9.24E+02	4.63E+00	2.26E-05	2.44E-13	1.74E-29	0.00E+00
ho166	l f	4.25	8.58E+04	8.58E+04	8.41E+04	8.33E+04	6.99E+04	4.69E+04	8.43E+03	8.10E+00	3.95E-05	4.26E-13	1.74E-29	0.00E+00
er167m	l	4.25	1.14E+03	8.40E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
hf175	l	4.25	3.62E+03	3.62E+03	3.62E+03	3.62E+03	3.61E+03	3.58E+03	3.48E+03	2.69E+03	1.49E+03	6.10E+02	9.78E+01	7.05E-02
lu176m	l	4.25	1.64E+03	1.64E+03	1.50E+03	1.36E+03	3.58E+02	1.70E+01	1.86E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
lu177	l	4.25	6.84E+02	6.84E+02	6.82E+02	6.81E+02	6.60E+02	6.17E+02	4.52E+02	3.12E+01	3.46E-01	1.93E-01	8.63E-02	3.72E-03
hf178m	l	4.25	3.46E+02	2.90E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
hf179m	l	4.5	3.98E+05	3.84E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
hf180m	l	4.25	1.12E+04	1.12E+04	1.05E+04	9.86E+03	4.10E+03	5.45E+02	6.24E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
hf181	l	4.25	2.38E+05	2.38E+05	2.38E+05	2.38E+05	2.37E+05	2.35E+05	2.23E+05	1.46E+05	5.47E+04	1.25E+04	6.07E+02	3.93E-03
w181	l	4.25	1.42E+03	1.42E+03	1.42E+03	1.42E+03	1.42E+03	1.41E+03	1.39E+03	1.20E+03	8.48E+02	5.09E+02	1.76E+02	2.70E+00
ta182	l	4.25	2.59E+04	2.59E+04	2.59E+04	2.59E+04	2.59E+04	2.57E+04	2.53E+04	2.16E+04	1.51E+04	8.79E+03	2.87E+03	3.51E+01
ta182m	l	4.25	4.18E+01	4.18E+01	1.12E+01	3.03E+00	3.13E-08	1.75E-26	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ta183	l	4.25	5.46E+04	5.46E+04	5.45E+04	5.43E+04	5.23E+04	4.77E+04	3.17E+04	9.32E+02	2.69E-01	1.32E-06	1.57E-17	0.00E+00
w183m	l	4.25	6.40E+04	6.29E+04	5.45E+04	5.43E+04	5.23E+04	4.77E+04	3.17E+04	9.32E+02	2.69E-01	1.32E-06	1.57E-17	0.00E+00
w185	l	4.25	3.52E+04	3.52E+04	3.52E+04	3.52E+04	3.51E+04	3.49E+04	3.39E+04	2.67E+04	1.54E+04	6.69E+03	1.21E+03	1.43E+00
w185m	l	4.25	7.55E+01	7.49E+01	2.96E-04	1.15E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
re186	l	4.25	2.58E+04	2.58E+04	2.57E+04	2.56E+04	2.43E+04	2.15E+04	1.24E+04	1.05E+02	1.73E-03	1.16E-10	1.99E-25	0.00E+00
w187	l	4.25	4.42E+05	4.42E+05	4.36E+05	4.30E+05	3.51E+05	2.21E+05	2.74E+04	3.77E-04	2.75E-22	0.00E+00	0.00E+00	0.00E+00
w188	l	4.25	1.96E+03	1.96E+03	1.96E+03	1.96E+03	1.96E+03	1.94E+03	1.89E+03	1.45E+03	8.02E+02	3.25E+02	5.12E+01	3.49E-02
re188	l	4.25	1.73E+05	1.73E+05	1.72E+05	1.69E+05	1.28E+05	6.75E+04	5.37E+03	1.47E+03	8.10E+02	3.29E+02	5.17E+01	3.52E-02
re188m	l	4.25	1.68E+05	1.67E+05	5.49E+04	1.80E+04	2.86E-03	8.33E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
os191	l	4.25	7.72E+01	7.72E+01	7.64E+01	7.64E+01	7.63E+01	7.49E+01	6.59E+01	2.05E+01	1.38E+00	2.40E-02	5.75E-06	3.06E-20
os191m	l	4.25	5.82E+01	5.82E+01	5.67E+01	5.52E+01	3.81E+01	1.63E+01	3.62E-01	1.66E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00

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Table 3.1

Activity (Curies) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide		limit enr	0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
ir192	l	4.25	3.01E+01	3.01E+01	3.01E+01	3.01E+01	3.00E+01	2.98E+01	2.90E+01	2.27E+01	1.29E+01	5.55E+00	9.78E-01	1.06E-03
np236m	a	4.25	4.84E+02	4.84E+02	4.77E+02	4.70E+02	3.78E+02	2.31E+02	2.51E+01	1.13E-07	6.12E-27	0.00E+00	0.00E+00	0.00E+00
u237	a	4.5	1.00E+08	1.00E+08	1.00E+08	9.93E+07	9.70E+07	9.02E+07	6.64E+07	4.60E+06	1.02E+04	4.48E+02	4.37E+02	3.97E+02
pu237	a	4.25	6.71E+02	6.71E+02	6.70E+02	6.70E+02	6.67E+02	6.60E+02	6.30E+02	4.23E+02	1.69E+02	4.23E+01	2.47E+00	3.35E-05
np238	a	4.25	4.70E+07	4.70E+07	4.67E+07	4.64E+07	4.22E+07	3.39E+07	1.27E+07	2.55E+03	7.44E+00	7.43E+00	7.41E+00	7.34E+00
pu238	a	4.25	4.56E+05	4.56E+05	4.56E+05	4.56E+05	4.56E+05	4.57E+05	4.58E+05	4.63E+05	4.69E+05	4.75E+05	4.82E+05	4.81E+05
u239	a	4.25	2.12E+09	2.12E+09	8.79E+08	3.61E+08	1.47E+03	7.16E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
np239	a	4.25	2.12E+09	2.12E+09	2.12E+09	2.11E+09	1.93E+09	1.59E+09	6.59E+08	3.16E+05	3.13E+03	3.13E+03	3.13E+03	3.13E+03
pu239	a	4.5	4.83E+04	4.83E+04	4.83E+04	4.83E+04	4.83E+04	4.84E+04	4.87E+04	4.88E+04	4.88E+04	4.88E+04	4.88E+04	4.88E+04
np240	a	4.25	3.93E+06	3.93E+06	2.81E+06	2.01E+06	1.82E+04	3.90E-01	6.20E-16	7.40E-16	1.01E-15	1.41E-15	2.25E-15	5.53E-15
pu240	a	4.25	7.79E+04	7.79E+04	7.79E+04	7.79E+04	7.79E+04	7.79E+04	7.79E+04	7.79E+04	7.79E+04	7.79E+04	7.79E+04	7.79E+04
pu241	a	4.25	1.92E+07	1.92E+07	1.92E+07	1.92E+07	1.92E+07	1.92E+07	1.92E+07	1.91E+07	1.89E+07	1.87E+07	1.83E+07	1.66E+07
am241	a	4.5	2.54E+04	2.54E+04	2.54E+04	2.54E+04	2.54E+04	2.55E+04	2.57E+04	2.79E+04	3.29E+04	4.02E+04	5.51E+04	1.11E+05
am242m	a	4.5	1.66E+03	1.66E+03	1.66E+03	1.66E+03	1.66E+03	1.66E+03	1.66E+03	1.66E+03	1.65E+03	1.65E+03	1.65E+03	1.63E+03
am242	a	4.25	1.14E+07	1.14E+07	1.12E+07	1.09E+07	8.10E+06	4.04E+06	1.81E+05	1.65E+03	1.64E+03	1.64E+03	1.64E+03	1.62E+03
cm242	a	4.25	6.67E+06	6.67E+06	6.67E+06	6.67E+06	6.67E+06	6.66E+06	6.59E+06	5.91E+06	4.58E+06	3.12E+06	1.42E+06	6.47E+04
pu243	a	4.25	4.19E+07	4.19E+07	3.90E+07	3.64E+07	1.37E+07	1.46E+06	6.17E+01	3.93E-05	3.93E-05	3.93E-05	3.93E-05	3.93E-05
am243	a	4.25	3.12E+03	3.12E+03	3.12E+03	3.12E+03	3.12E+03	3.13E+03	3.13E+03	3.13E+03	3.13E+03	3.13E+03	3.13E+03	3.13E+03
cm243	a	4.25	2.87E+03	2.87E+03	2.87E+03	2.87E+03	2.87E+03	2.87E+03	2.87E+03	2.87E+03	2.86E+03	2.84E+03	2.80E+03	2.67E+03
am244	a	4.25	1.38E+07	1.38E+07	1.33E+07	1.29E+07	7.95E+06	2.66E+06	1.90E+04	4.79E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cm244	a	4.25	3.90E+05	3.90E+05	3.90E+05	3.90E+05	3.90E+05	3.90E+05	3.90E+05	3.89E+05	3.87E+05	3.83E+05	3.76E+05	3.48E+05

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Mass (grams) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Isotopic Mass (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
h 1	4.5	l	3.73E+05	3.73E+05	3.73E+05	3.73E+05	3.73E+05	3.73E+05	3.73E+05	3.73E+05	3.73E+05	3.73E+05	3.73E+05	3.73E+05
h 2	4.25	l	5.00E+02	5.00E+02	5.00E+02	5.00E+02	5.00E+02	5.00E+02	5.00E+02	5.00E+02	5.00E+02	5.00E+02	5.00E+02	5.00E+02
h 3	4.25	f	7.87E+00	7.87E+00	7.87E+00	7.87E+00	7.87E+00	7.87E+00	7.87E+00	7.79E+00	7.72E+00	7.62E+00	7.41E+00	6.62E+00
he 4	4.25	a l	6.88E+01	6.88E+01	6.88E+01	6.88E+01	6.89E+01	6.90E+01	6.95E+01	7.35E+01	8.14E+01	9.10E+01	1.03E+02	1.26E+02
o 16	4.5	l	6.10E+04	6.10E+04	6.10E+04	6.10E+04	6.10E+04	6.10E+04	6.10E+04	6.10E+04	6.10E+04	6.10E+04	6.10E+04	6.10E+04
o 17	4.5	l	2.47E+01	2.47E+01	2.47E+01	2.47E+01	2.47E+01	2.47E+01	2.47E+01	2.47E+01	2.47E+01	2.47E+01	2.47E+01	2.47E+01
o 18	4.5	l	1.41E+02	1.41E+02	1.41E+02	1.41E+02	1.41E+02	1.41E+02	1.41E+02	1.41E+02	1.41E+02	1.41E+02	1.41E+02	1.41E+02
mg 24	4.5	l	1.19E+03	1.19E+03	1.19E+03	1.19E+03	1.19E+03	1.19E+03	1.19E+03	1.19E+03	1.19E+03	1.19E+03	1.19E+03	1.19E+03
mg 25	4.5	l	1.57E+02	1.57E+02	1.57E+02	1.57E+02	1.57E+02	1.57E+02	1.57E+02	1.57E+02	1.57E+02	1.57E+02	1.57E+02	1.57E+02
mg 26	4.5	l	1.80E+02	1.80E+02	1.80E+02	1.80E+02	1.80E+02	1.80E+02	1.80E+02	1.80E+02	1.80E+02	1.80E+02	1.80E+02	1.80E+02
al 27	4.5	l	4.58E+03	4.58E+03	4.58E+03	4.58E+03	4.58E+03	4.58E+03	4.58E+03	4.58E+03	4.58E+03	4.58E+03	4.58E+03	4.58E+03
si 28	4.5	l	6.67E+03	6.67E+03	6.67E+03	6.67E+03	6.67E+03	6.67E+03	6.67E+03	6.67E+03	6.67E+03	6.67E+03	6.67E+03	6.67E+03
si 29	4.5	l	3.52E+02	3.52E+02	3.52E+02	3.52E+02	3.52E+02	3.52E+02	3.52E+02	3.52E+02	3.52E+02	3.52E+02	3.52E+02	3.52E+02
si 30	4.5	l	2.40E+02	2.40E+02	2.40E+02	2.40E+02	2.40E+02	2.40E+02	2.40E+02	2.40E+02	2.40E+02	2.40E+02	2.40E+02	2.40E+02
v 50	4.25	l	1.21E+01	1.21E+01	1.21E+01	1.21E+01	1.21E+01	1.21E+01	1.21E+01	1.21E+01	1.21E+01	1.21E+01	1.21E+01	1.21E+01
cr 50	4.5	l	4.86E+04	4.86E+04	4.86E+04	4.86E+04	4.86E+04	4.86E+04	4.86E+04	4.86E+04	4.86E+04	4.86E+04	4.86E+04	4.86E+04
v 51	4.25	l	1.61E+03	1.61E+03	1.61E+03	1.61E+03	1.61E+03	1.61E+03	1.62E+03	1.64E+03	1.67E+03	1.67E+03	1.67E+03	1.67E+03
cr 51	4.25	l	6.11E+01	6.11E+01	6.11E+01	6.11E+01	6.07E+01	5.97E+01	5.53E+01	2.89E+01	6.43E+00	6.78E-01	6.58E-03	7.62E-11
cr 52	4.5	l	1.00E+06	1.00E+06	1.00E+06	1.00E+06	1.00E+06	1.00E+06	1.00E+06	1.00E+06	1.00E+06	1.00E+06	1.00E+06	1.00E+06
cr 53	4.5	l	1.14E+05	1.14E+05	1.14E+05	1.14E+05	1.14E+05	1.14E+05	1.14E+05	1.14E+05	1.14E+05	1.14E+05	1.14E+05	1.14E+05
cr 54	4.25	l	3.40E+04	3.40E+04	3.40E+04	3.40E+04	3.40E+04	3.40E+04	3.40E+04	3.40E+04	3.40E+04	3.40E+04	3.40E+04	3.40E+04
mn 54	4.25	l	5.52E+01	5.52E+01	5.52E+01	5.52E+01	5.51E+01	5.50E+01	5.46E+01	5.16E+01	4.52E+01	3.70E+01	2.45E+01	4.84E+00
fe 54	4.5	l	2.39E+05	2.39E+05	2.39E+05	2.39E+05	2.39E+05	2.39E+05	2.39E+05	2.39E+05	2.39E+05	2.39E+05	2.39E+05	2.39E+05
mn 55	4.5	l	9.63E+04	9.63E+04	9.63E+04	9.63E+04	9.63E+04	9.63E+04	9.63E+04	9.63E+04	9.63E+04	9.63E+04	9.63E+04	9.63E+04
fe 55	4.25	l	8.02E+02	8.02E+02	8.02E+02	8.02E+02	8.02E+02	8.02E+02	7.95E+02	7.87E+02	7.52E+02	7.06E+02	6.21E+02	3.74E+02
fe 56	4.5	l	3.92E+06	3.92E+06	3.92E+06	3.92E+06	3.92E+06	3.92E+06	3.92E+06	3.92E+06	3.92E+06	3.92E+06	3.92E+06	3.92E+06
fe 57	4.25	l	1.16E+05	1.16E+05	1.16E+05	1.16E+05	1.16E+05	1.16E+05	1.16E+05	1.16E+05	1.16E+05	1.16E+05	1.16E+05	1.16E+05
fe 58	4.25	l	1.37E+04	1.37E+04	1.37E+04	1.37E+04	1.37E+04	1.37E+04	1.37E+04	1.37E+04	1.37E+04	1.37E+04	1.37E+04	1.37E+04
co 58	4.25	l	1.86E+01	1.86E+01	1.86E+01	1.86E+01	1.86E+01	1.84E+01	1.79E+01	1.39E+01	7.72E+00	3.21E+00	5.24E-01	4.16E-04
ni 58	4.5	l	3.81E+05	3.81E+05	3.81E+05	3.81E+05	3.81E+05	3.81E+05	3.81E+05	3.81E+05	3.81E+05	3.81E+05	3.81E+05	3.81E+05
co 59	4.5	l	2.36E+03	2.36E+03	2.36E+03	2.36E+03	2.36E+03	2.36E+03	2.36E+03	2.36E+03	2.36E+03	2.36E+03	2.36E+03	2.36E+03
ni 59	4.25	l	3.19E+03	3.19E+03	3.19E+03	3.19E+03	3.19E+03	3.19E+03	3.19E+03	3.19E+03	3.19E+03	3.19E+03	3.19E+03	3.19E+03
co 60	4.25	l	2.82E+02	2.82E+02	2.82E+02	2.82E+02	2.82E+02	2.82E+02	2.81E+02	2.79E+02	2.73E+02	2.64E+02	2.48E+02	1.90E+02
ni 60	4.5	l	1.52E+05	1.52E+05	1.52E+05	1.52E+05	1.52E+05	1.52E+05	1.52E+05	1.52E+05	1.52E+05	1.52E+05	1.52E+05	1.52E+05

Mass (grams) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Isotopic Mass (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
ni 61	4.25	l	7.60E+03	7.60E+03	7.60E+03	7.60E+03	7.60E+03	7.60E+03	7.60E+03	7.60E+03	7.60E+03	7.60E+03	7.60E+03	7.60E+03
ni 62	4.5	l	2.11E+04	2.11E+04	2.11E+04	2.11E+04	2.11E+04	2.11E+04	2.11E+04	2.11E+04	2.11E+04	2.11E+04	2.11E+04	2.11E+04
ni 63	4.25	l	6.26E+02	6.26E+02	6.26E+02	6.26E+02	6.26E+02	6.26E+02	6.26E+02	6.26E+02	6.26E+02	6.25E+02	6.23E+02	6.14E+02
cu 63	4.25	l	7.47E+00	7.47E+00	7.47E+00	7.47E+00	7.48E+00	7.49E+00	7.53E+00	7.87E+00	8.56E+00	9.63E+00	1.18E+01	2.04E+01
ni 64	4.5	l	5.65E+03	5.65E+03	5.65E+03	5.65E+03	5.65E+03	5.65E+03	5.65E+03	5.65E+03	5.65E+03	5.65E+03	5.65E+03	5.65E+03
cu 65	4.25	l	2.09E+01	2.09E+01	2.09E+01	2.09E+01	2.09E+01	2.09E+01	2.09E+01	2.09E+01	2.09E+01	2.09E+01	2.09E+01	2.09E+01
ge 72	4.25	f	7.64E-01	7.64E-01	7.64E-01	7.64E-01	7.64E-01	7.64E-01	7.64E-01	7.64E-01	7.64E-01	7.64E-01	7.64E-01	7.64E-01
ge 73	4.25	f	2.33E+00	2.33E+00	2.33E+00	2.33E+00	2.33E+00	2.33E+00	2.33E+00	2.33E+00	2.33E+00	2.33E+00	2.33E+00	2.33E+00
ge 74	4.25	f	2.06E+00	2.06E+00	2.06E+00	2.06E+00	2.06E+00	2.06E+00	2.06E+00	2.06E+00	2.06E+00	2.06E+00	2.06E+00	2.06E+00
as 75	4.5	f	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01
ge 76	4.5	f	5.84E+01	5.84E+01	5.84E+01	5.84E+01	5.84E+01	5.84E+01	5.84E+01	5.84E+01	5.84E+01	5.84E+01	5.84E+01	5.84E+01
se 76	4.25	f	6.06E-01	6.06E-01	6.06E-01	6.06E-01	6.06E-01	6.07E-01	6.07E-01	6.07E-01	6.07E-01	6.07E-01	6.07E-01	6.07E-01
se 77	4.5	f	1.31E+02	1.31E+02	1.31E+02	1.31E+02	1.31E+02	1.31E+02	1.31E+02	1.31E+02	1.31E+02	1.31E+02	1.31E+02	1.31E+02
se 78	4.5	f	4.19E+02	4.19E+02	4.19E+02	4.19E+02	4.19E+02	4.19E+02	4.19E+02	4.19E+02	4.19E+02	4.19E+02	4.19E+02	4.19E+02
se 79	4.5	f	8.02E+02	8.02E+02	8.02E+02	8.02E+02	8.02E+02	8.02E+02	8.02E+02	8.02E+02	8.02E+02	8.02E+02	8.02E+02	8.02E+02
se 80	4.5	f	2.25E+03	2.25E+03	2.25E+03	2.25E+03	2.25E+03	2.25E+03	2.25E+03	2.25E+03	2.25E+03	2.25E+03	2.25E+03	2.25E+03
br 81	4.5	f	3.31E+03	3.31E+03	3.31E+03	3.31E+03	3.31E+03	3.31E+03	3.31E+03	3.31E+03	3.31E+03	3.31E+03	3.31E+03	3.31E+03
se 82	4.5	f	5.45E+03	5.45E+03	5.45E+03	5.45E+03	5.45E+03	5.45E+03	5.45E+03	5.45E+03	5.45E+03	5.45E+03	5.45E+03	5.45E+03
kr 82	4.25	f	1.04E+02	1.04E+02	1.04E+02	1.04E+02	1.04E+02	1.05E+02	1.05E+02	1.05E+02	1.05E+02	1.05E+02	1.05E+02	1.05E+02
w183	4.5	l	1.21E+03	1.21E+03	1.21E+03	1.21E+03	1.21E+03	1.21E+03	1.21E+03	1.21E+03	1.22E+03	1.22E+03	1.22E+03	1.22E+03
br 83	4.5	f	8.17E-01	8.17E-01	7.47E-01	6.60E-01	8.86E-02	8.71E-04	8.17E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 83	4.5	f	7.10E+03	7.10E+03	7.10E+03	7.10E+03	7.10E+03	7.10E+03	7.10E+03	7.10E+03	7.10E+03	7.10E+03	7.10E+03	7.10E+03
kr 83m	4.5	f	6.31E-01	6.31E-01	6.26E-01	6.10E-01	1.76E-01	2.55E-03	2.61E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
w184	4.25	l	2.15E+03	2.15E+03	2.15E+03	2.15E+03	2.15E+03	2.15E+03	2.15E+03	2.15E+03	2.15E+03	2.15E+03	2.15E+03	2.15E+03
kr 84	4.5	f	1.93E+04	1.93E+04	1.93E+04	1.93E+04	1.93E+04	1.93E+04	1.93E+04	1.93E+04	1.93E+04	1.93E+04	1.93E+04	1.93E+04
kr 85	4.5	f	3.78E+03	3.78E+03	3.78E+03	3.78E+03	3.78E+03	3.78E+03	3.77E+03	3.76E+03	3.72E+03	3.66E+03	3.54E+03	3.11E+03
kr 85m	4.5	f	3.25E+00	3.25E+00	3.05E+00	2.82E+00	9.55E-01	8.02E-02	1.17E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 85	4.5	f	1.60E+04	1.60E+04	1.60E+04	1.60E+04	1.60E+04	1.60E+04	1.60E+04	1.60E+04	1.60E+04	1.60E+04	1.62E+04	1.66E+04
w186	4.5	l	1.12E+03	1.12E+03	1.12E+03	1.12E+03	1.12E+03	1.12E+03	1.12E+03	1.12E+03	1.12E+03	1.12E+03	1.12E+03	1.12E+03
kr 86	4.5	f	3.16E+04	3.16E+04	3.16E+04	3.16E+04	3.16E+04	3.16E+04	3.16E+04	3.16E+04	3.16E+04	3.16E+04	3.16E+04	3.16E+04
rb 86	4.25	f	2.66E+00	2.66E+00	2.66E+00	2.65E+00	2.63E+00	2.56E+00	2.29E+00	8.71E-01	9.32E-02	3.29E-03	3.35E-06	5.29E-18
sr 86	4.25	f	6.17E+01	6.17E+01	6.17E+01	6.17E+01	6.17E+01	6.18E+01	6.21E+01	6.35E+01	6.43E+01	6.44E+01	6.44E+01	6.44E+01
kr 87	4.5	f	1.89E+00	1.89E+00	1.46E+00	1.11E+00	2.44E-02	4.00E-06	3.64E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 87	4.5	f	4.13E+04	4.13E+04	4.13E+04	4.13E+04	4.13E+04	4.13E+04	4.13E+04	4.13E+04	4.13E+04	4.13E+04	4.13E+04	4.13E+04

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Mass (grams) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Isotopic Mass (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr			0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
kr 88	4.5		f	5.94E+00	5.94E+00	5.26E+00	4.65E+00	8.40E-01	1.70E-02	3.93E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 88	4.5		f	6.36E-01	6.36E-01	5.95E-01	5.37E-01	9.86E-02	1.98E-03	4.58E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 88	4.5	l	f	5.92E+04	5.92E+04	5.92E+04	5.92E+04	5.92E+04	5.92E+04	5.92E+04	5.92E+04	5.92E+04	5.92E+04	5.92E+04	5.92E+04
rb 89	4.5		f	7.17E-01	7.17E-01	2.27E-01	5.78E-02	2.78E-10	2.70E-29	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 89	4.5		f	3.56E+03	3.56E+03	3.56E+03	3.55E+03	3.54E+03	3.51E+03	3.37E+03	2.36E+03	1.04E+03	3.02E+02	2.38E+01	1.06E-03
y 89	4.5	l	f	7.57E+04	7.57E+04	7.57E+04	7.57E+04	7.57E+04	7.58E+04	7.60E+04	7.72E+04	7.80E+04	7.87E+04	7.95E+04	7.95E+04
sr 90	4.5		f	9.24E+04	9.24E+04	9.24E+04	9.24E+04	9.24E+04	9.24E+04	9.24E+04	9.24E+04	9.17E+04	9.09E+04	9.02E+04	8.56E+04
y 90	4.5		f	2.47E+01	2.47E+01	2.47E+01	2.47E+01	2.46E+01	2.45E+01	2.42E+01	2.39E+01	2.38E+01	2.37E+01	2.34E+01	2.23E+01
zr 90	4.5	l	f	2.96E+07	2.96E+07	2.96E+07	2.96E+07	2.96E+07	2.96E+07	2.96E+07	2.96E+07	2.96E+07	2.96E+07	2.97E+07	2.97E+07
sr 91	4.5		f	3.61E+01	3.61E+01	3.48E+01	3.36E+01	2.02E+01	6.30E+00	3.32E-02	6.16E-22	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 91	4.5		f	5.47E+03	5.47E+03	5.47E+03	5.47E+03	5.46E+03	5.43E+03	5.25E+03	3.86E+03	1.89E+03	6.52E+02	7.27E+01	1.27E-02
y 91m	4.5		f	1.82E+00	1.82E+00	1.81E+00	1.78E+00	1.12E+00	3.48E-01	1.84E-03	3.41E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr 91	4.5	l	f	6.58E+06	6.58E+06	6.58E+06	6.58E+06	6.58E+06	6.58E+06	6.58E+06	6.58E+06	6.59E+06	6.59E+06	6.59E+06	6.59E+06
mo 92	4.5	l		4.32E+02	4.32E+02	4.32E+02	4.32E+02	4.32E+02	4.32E+02	4.32E+02	4.32E+02	4.32E+02	4.32E+02	4.32E+02	4.32E+02
sr 92	4.5		f	1.11E+01	1.11E+01	9.70E+00	8.56E+00	1.43E+00	2.38E-02	2.40E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 92	4.5		f	1.45E+01	1.45E+01	1.44E+01	1.42E+01	6.78E+00	4.59E-01	4.20E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr 92	4.5	l	f	1.02E+07	1.02E+07	1.02E+07	1.02E+07	1.02E+07	1.02E+07	1.02E+07	1.02E+07	1.02E+07	1.02E+07	1.02E+07	1.02E+07
sr 93	4.5		f	5.76E-01	5.76E-01	3.53E-02	2.14E-03	1.99E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 93	4.5		f	3.19E+01	3.19E+01	3.12E+01	3.02E+01	1.86E+01	6.23E+00	4.45E-02	1.12E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr 93	4.5	l	f	9.10E+04	9.10E+04	9.10E+04	9.10E+04	9.10E+04	9.10E+04	9.10E+04	9.10E+04	9.10E+04	9.10E+04	9.10E+04	9.10E+04
mo 94	4.5	l		2.78E+02	2.78E+02	2.78E+02	2.78E+02	2.78E+02	2.78E+02	2.78E+02	2.78E+02	2.78E+02	2.78E+02	2.78E+02	2.78E+02
y 94	4.5		f	1.58E+00	1.58E+00	5.56E-01	1.83E-01	3.17E-08	1.12E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr 94	4.5	l	f	1.06E+07	1.06E+07	1.06E+07	1.06E+07	1.06E+07	1.06E+07	1.06E+07	1.06E+07	1.06E+07	1.06E+07	1.06E+07	1.06E+07
y 95	4.5		f	9.40E-01	9.40E-01	1.34E-01	1.84E-02	1.67E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr 95	4.5	l	f	8.90E+03	8.90E+03	8.90E+03	8.90E+03	8.90E+03	8.82E+03	8.58E+03	6.45E+03	3.37E+03	1.27E+03	1.71E+02	6.28E-02
nb 95	4.5	l	f	4.89E+03	4.89E+03	4.89E+03	4.89E+03	4.89E+03	4.89E+03	4.89E+03	4.54E+03	3.07E+03	1.36E+03	2.01E+02	7.55E-02
nb 95m	4.5		f	5.33E+00	5.33E+00	5.33E+00	5.33E+00	5.33E+00	5.33E+00	5.26E+00	4.07E+00	2.13E+00	8.02E-01	1.08E-01	3.97E-05
mo 95	4.5	l	f	1.18E+05	1.18E+05	1.18E+05	1.18E+05	1.18E+05	1.18E+05	1.19E+05	1.21E+05	1.25E+05	1.29E+05	1.32E+05	1.32E+05
zr 96	4.5	l	f	1.84E+06	1.84E+06	1.84E+06	1.84E+06	1.84E+06	1.84E+06	1.84E+06	1.84E+06	1.84E+06	1.84E+06	1.84E+06	1.84E+06
mo 96	4.25	l	f	7.10E+03	7.10E+03	7.10E+03	7.10E+03	7.10E+03	7.10E+03	7.10E+03	7.10E+03	7.10E+03	7.10E+03	7.10E+03	7.10E+03
zr 97	4.25	l	f	9.92E+01	9.92E+01	9.73E+01	9.50E+01	7.15E+01	3.71E+01	1.94E+00	1.48E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb 97	4.5		f	6.52E+00	6.52E+00	6.49E+00	6.43E+00	5.00E+00	2.44E+00	1.27E-01	1.05E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo 97	4.25	l	f	1.35E+05	1.35E+05	1.35E+05	1.35E+05	1.35E+05	1.35E+05	1.35E+05	1.35E+05	1.35E+05	1.35E+05	1.35E+05	1.35E+05
mo 98	4.5	l	f	1.36E+05	1.36E+05	1.36E+05	1.36E+05	1.36E+05	1.36E+05	1.36E+05	1.36E+05	1.36E+05	1.36E+05	1.36E+05	1.36E+05

Mass (grams) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Isotopic Mass (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
tc 98	4.25	f	1.05E+00	1.05E+00	1.05E+00	1.05E+00	1.05E+00	1.05E+00	1.05E+00	1.05E+00	1.05E+00	1.05E+00	1.05E+00	1.05E+00
mo 99	4.5	f	4.20E+02	4.20E+02	4.19E+02	4.16E+02	3.87E+02	3.27E+02	1.54E+02	2.17E-01	5.80E-08	8.02E-18	4.05E-38	0.00E+00
tc 99	4.5	f	1.29E+05	1.29E+05	1.29E+05	1.29E+05	1.29E+05	1.29E+05	1.29E+05	1.29E+05	1.29E+05	1.29E+05	1.29E+05	1.29E+05
tc 99m	4.5	f	3.40E+01	3.40E+01	3.39E+01	3.39E+01	3.29E+01	2.87E+01	1.35E+01	1.92E-02	5.12E-09	7.07E-19	0.00E+00	0.00E+00
ru 99	4.5	f	5.25E+00	5.25E+00	5.25E+00	5.25E+00	5.25E+00	5.25E+00	5.26E+00	5.29E+00	5.36E+00	5.47E+00	5.68E+00	6.53E+00
mo100	4.5	l f	1.52E+05	1.52E+05	1.52E+05	1.52E+05	1.52E+05	1.52E+05	1.52E+05	1.52E+05	1.52E+05	1.52E+05	1.52E+05	1.52E+05
ru100	4.25	f	1.73E+04	1.73E+04	1.73E+04	1.73E+04	1.73E+04	1.73E+04	1.73E+04	1.73E+04	1.73E+04	1.73E+04	1.73E+04	1.73E+04
mo101	4.5	f	1.44E+00	1.44E+00	3.49E-01	8.40E-02	1.84E-10	2.96E-30	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc101	4.5	f	1.40E+00	1.40E+00	8.10E-01	3.05E-01	3.19E-09	9.02E-29	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru101	4.5	f	1.24E+05	1.24E+05	1.24E+05	1.24E+05	1.24E+05	1.24E+05	1.24E+05	1.24E+05	1.24E+05	1.24E+05	1.24E+05	1.24E+05
mo102	4.25	f	1.07E+00	1.06E+00	1.70E-01	2.70E-02	1.75E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru102	4.25	f	1.25E+05	1.25E+05	1.25E+05	1.25E+05	1.25E+05	1.25E+05	1.25E+05	1.25E+05	1.25E+05	1.25E+05	1.25E+05	1.25E+05
ru103	4.25	f	5.31E+03	5.31E+03	5.31E+03	5.31E+03	5.28E+03	5.22E+03	4.95E+03	3.12E+03	1.08E+03	2.22E+02	8.40E+00	2.11E-05
rh103	4.25	f	6.86E+04	6.86E+04	6.86E+04	6.86E+04	6.86E+04	6.87E+04	6.90E+04	7.08E+04	7.28E+04	7.37E+04	7.39E+04	7.39E+04
rh103m	4.25	f	5.26E+00	5.26E+00	5.26E+00	5.26E+00	5.23E+00	5.17E+00	4.90E+00	3.10E+00	1.08E+00	2.19E-01	8.33E-03	2.09E-08
tc104	4.25	f	1.43E+00	1.43E+00	4.85E-01	1.56E-01	1.92E-08	3.10E-24	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru104	4.25	f	8.33E+04	8.33E+04	8.33E+04	8.33E+04	8.33E+04	8.33E+04	8.33E+04	8.33E+04	8.33E+04	8.33E+04	8.33E+04	8.33E+04
pd104	4.25	f	3.50E+04	3.50E+04	3.50E+04	3.50E+04	3.50E+04	3.50E+04	3.50E+04	3.50E+04	3.50E+04	3.50E+04	3.50E+04	3.50E+04
ru105	4.25	f	1.76E+01	1.76E+01	1.68E+01	1.56E+01	5.23E+00	4.29E-01	5.62E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh105	4.25	f	1.31E+02	1.31E+02	1.31E+02	1.31E+02	1.25E+02	9.47E+01	2.32E+01	1.13E-04	6.23E-17	2.55E-35	0.00E+00	0.00E+00
pd105	4.25	f	5.91E+04	5.91E+04	5.91E+04	5.91E+04	5.91E+04	5.91E+04	5.91E+04	5.92E+04	5.92E+04	5.92E+04	5.92E+04	5.92E+04
ru106	4.25	f	2.06E+04	2.06E+04	2.06E+04	2.06E+04	2.06E+04	2.06E+04	2.05E+04	1.96E+04	1.75E+04	1.47E+04	1.05E+04	2.67E+03
pd106	4.25	f	3.24E+04	3.24E+04	3.24E+04	3.24E+04	3.25E+04	3.25E+04	3.25E+04	3.35E+04	3.56E+04	3.84E+04	4.26E+04	5.04E+04
rh107	4.25	f	8.63E-01	8.63E-01	4.05E-01	1.56E-01	2.32E-07	1.12E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd107	4.25	f	3.25E+04	3.25E+04	3.25E+04	3.25E+04	3.25E+04	3.25E+04	3.25E+04	3.25E+04	3.25E+04	3.25E+04	3.25E+04	3.25E+04
pd108	4.25	f	2.11E+04	2.11E+04	2.11E+04	2.11E+04	2.11E+04	2.11E+04	2.11E+04	2.11E+04	2.11E+04	2.11E+04	2.11E+04	2.11E+04
pd109	4.25	f	1.97E+01	1.97E+01	1.93E+01	1.88E+01	1.31E+01	5.87E+00	1.54E-01	2.99E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag109	4.25	f	1.28E+04	1.28E+04	1.28E+04	1.28E+04	1.28E+04	1.28E+04	1.28E+04	1.28E+04	1.28E+04	1.28E+04	1.28E+04	1.28E+04
pd110	4.25	f	6.33E+03	6.33E+03	6.33E+03	6.33E+03	6.33E+03	6.33E+03	6.33E+03	6.33E+03	6.33E+03	6.33E+03	6.33E+03	6.33E+03
ag110m	4.25	f	9.63E+01	9.63E+01	9.63E+01	9.63E+01	9.63E+01	9.63E+01	9.55E+01	8.86E+01	7.51E+01	5.85E+01	3.50E+01	4.61E+00
cd110	4.25	f	5.66E+03	5.66E+03	5.66E+03	5.66E+03	5.66E+03	5.66E+03	5.66E+03	5.67E+03	5.68E+03	5.70E+03	5.72E+03	5.75E+03
ag111	4.25	f	4.25E+01	4.25E+01	4.25E+01	4.24E+01	4.13E+01	3.89E+01	2.94E+01	2.61E+00	9.86E-03	2.28E-06	7.44E-14	0.00E+00
cd111	4.25	f	3.28E+03	3.28E+03	3.28E+03	3.28E+03	3.28E+03	3.28E+03	3.29E+03	3.32E+03	3.32E+03	3.32E+03	3.32E+03	3.32E+03
sn112	4.5	l	8.02E+03	8.02E+03	8.02E+03	8.02E+03	8.02E+03	8.02E+03	8.02E+03	8.02E+03	8.02E+03	8.02E+03	8.02E+03	8.02E+03

NFE-B-NA-169 REV 1

Mass (grams) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Isotopic Mass (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr			0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
pd112	4.25		f	2.27E+00	2.27E+00	2.23E+00	2.19E+00	1.74E+00	1.03E+00	9.63E-02	1.14E-10	2.87E-31	0.00E+00	0.00E+00	0.00E+00
cd112	4.25		f	1.72E+03	1.72E+03	1.72E+03	1.72E+03	1.72E+03	1.73E+03	1.73E+03	1.73E+03	1.73E+03	1.73E+03	1.73E+03	1.73E+03
sn113	4.25	l		3.45E+01	3.45E+01	3.45E+01	3.45E+01	3.45E+01	3.43E+01	3.37E+01	2.88E+01	2.01E+01	1.17E+01	3.83E+00	4.71E-02
cd113	4.5		f	1.45E+01	1.45E+01	1.45E+01	1.45E+01	1.47E+01	1.48E+01	1.48E+01	1.48E+01	1.48E+01	1.48E+01	1.48E+01	1.48E+01
cd113m	4.25		f	1.93E+01	1.93E+01	1.93E+01	1.93E+01	1.93E+01	1.93E+01	1.93E+01	1.93E+01	1.91E+01	1.89E+01	1.84E+01	1.67E+01
in113	4.25	l	f	2.17E+02	2.17E+02	2.17E+02	2.17E+02	2.17E+02	2.18E+02	2.18E+02	2.23E+02	2.32E+02	2.40E+02	2.49E+02	2.54E+02
sn114	4.5	l		5.62E+03	5.62E+03	5.62E+03	5.62E+03	5.62E+03	5.62E+03	5.62E+03	5.62E+03	5.62E+03	5.62E+03	5.62E+03	5.62E+03
cd114	4.25		f	1.83E+03	1.83E+03	1.83E+03	1.83E+03	1.83E+03	1.83E+03	1.83E+03	1.83E+03	1.83E+03	1.83E+03	1.83E+03	1.83E+03
cd115	4.25		f	1.76E+00	1.76E+00	1.76E+00	1.75E+00	1.60E+00	1.30E+00	5.11E-01	1.57E-04	1.23E-12	8.48E-25	0.00E+00	0.00E+00
cd115m	4.25		f	1.65E+00	1.65E+00	1.65E+00	1.65E+00	1.64E+00	1.62E+00	1.55E+00	1.03E+00	4.07E-01	1.01E-01	5.65E-03	6.62E-08
in115	4.25		f	2.22E+02	2.22E+02	2.22E+02	2.22E+02	2.22E+02	2.22E+02	2.24E+02	2.25E+02	2.25E+02	2.25E+02	2.25E+02	2.25E+02
sn115	4.5	l	f	2.98E+03	2.98E+03	2.98E+03	2.98E+03	2.98E+03	2.98E+03	2.98E+03	2.98E+03	2.98E+03	2.98E+03	2.98E+03	2.98E+03
cd116	4.25		f	7.72E+02	7.72E+02	7.72E+02	7.72E+02	7.72E+02	7.72E+02	7.72E+02	7.72E+02	7.72E+02	7.72E+02	7.72E+02	7.72E+02
sn116	4.25	l	f	1.26E+05	1.26E+05	1.26E+05	1.26E+05	1.26E+05	1.26E+05	1.26E+05	1.26E+05	1.26E+05	1.26E+05	1.26E+05	1.26E+05
sn117m	4.25	l		3.01E+01	3.01E+01	3.01E+01	3.01E+01	2.96E+01	2.87E+01	2.46E+01	6.53E+00	3.07E-01	3.12E-03	2.48E-07	1.67E-23
sn117	4.25	l	f	6.80E+04	6.80E+04	6.80E+04	6.80E+04	6.80E+04	6.80E+04	6.80E+04	6.81E+04	6.81E+04	6.81E+04	6.81E+04	6.81E+04
sn118	4.25	l	f	2.13E+05	2.13E+05	2.13E+05	2.13E+05	2.13E+05	2.13E+05	2.13E+05	2.13E+05	2.13E+05	2.13E+05	2.13E+05	2.13E+05
sn119	4.25	l	f	7.70E+04	7.70E+04	7.70E+04	7.70E+04	7.70E+04	7.70E+04	7.70E+04	7.70E+04	7.70E+04	7.70E+04	7.78E+04	7.78E+04
sn119m	4.25	l	f	6.55E+02	6.55E+02	6.55E+02	6.55E+02	6.55E+02	6.54E+02	6.49E+02	6.11E+02	5.30E+02	4.28E+02	2.76E+02	4.91E+01
sn120	4.25	l	f	2.89E+05	2.89E+05	2.89E+05	2.89E+05	2.89E+05	2.89E+05	2.89E+05	2.89E+05	2.89E+05	2.89E+05	2.89E+05	2.89E+05
sn121	4.25		f	9.09E-01	9.09E-01	9.02E-01	8.94E-01	7.44E-01	4.94E-01	7.87E-02	2.80E-04	2.80E-04	2.79E-04	2.77E-04	2.70E-04
sn121m	4.25		f	6.45E+00	6.45E+00	6.45E+00	6.45E+00	6.45E+00	6.44E+00	6.44E+00	6.44E+00	6.43E+00	6.40E+00	6.36E+00	6.20E+00
sb121	4.25	l	f	1.07E+03	1.07E+03	1.07E+03	1.07E+03	1.07E+03	1.07E+03	1.07E+03	1.07E+03	1.07E+03	1.07E+03	1.07E+03	1.07E+03
sn122	4.25	l	f	4.23E+04	4.23E+04	4.23E+04	4.23E+04	4.23E+04	4.23E+04	4.23E+04	4.23E+04	4.23E+04	4.23E+04	4.23E+04	4.23E+04
te122	4.25	l	f	7.76E+01	7.76E+01	7.77E+01	7.77E+01	7.77E+01	7.78E+01	7.79E+01	7.82E+01	7.82E+01	7.82E+01	7.82E+01	7.82E+01
sn123	4.25	l	f	1.66E+01	1.66E+01	1.66E+01	1.66E+01	1.66E+01	1.64E+01	1.62E+01	1.41E+01	1.02E+01	6.30E+00	2.33E+00	4.63E-02
sb123	4.25	l	f	7.42E+02	7.42E+02	7.42E+02	7.42E+02	7.42E+02	7.42E+02	7.43E+02	7.44E+02	7.49E+02	7.52E+02	7.56E+02	7.59E+02
sn124	4.25	l	f	5.23E+04	5.23E+04	5.23E+04	5.23E+04	5.23E+04	5.23E+04	5.23E+04	5.23E+04	5.23E+04	5.23E+04	5.23E+04	5.23E+04
sb124	4.25		f	5.24E+00	5.24E+00	5.24E+00	5.24E+00	5.23E+00	5.19E+00	5.01E+00	3.71E+00	1.86E+00	6.60E-01	7.79E-02	1.73E-05
te124	4.25		f	3.35E+01	3.35E+01	3.35E+01	3.35E+01	3.35E+01	3.35E+01	3.38E+01	3.51E+01	3.69E+01	3.81E+01	3.87E+01	3.87E+01
sn125	4.25		f	5.01E+00	5.01E+00	5.00E+00	5.00E+00	4.90E+00	4.67E+00	3.76E+00	5.80E-01	7.79E-03	1.20E-05	1.97E-11	0.00E+00
sb125	4.25	l	f	1.31E+03	1.31E+03	1.31E+03	1.31E+03	1.31E+03	1.31E+03	1.31E+03	1.29E+03	1.23E+03	1.16E+03	1.02E+03	6.13E+02
te125	4.25	l	f	6.13E+02	6.13E+02	6.13E+02	6.13E+02	6.13E+02	6.14E+02	6.17E+02	6.40E+02	6.93E+02	7.69E+02	9.10E+02	1.32E+03
te125m	4.25		f	1.38E+01	1.38E+01	1.38E+01	1.38E+01	1.38E+01	1.38E+01	1.38E+01	1.39E+01	1.38E+01	1.33E+01	1.18E+01	7.14E+00

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Mass (grams) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Isotopic Mass (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
sn126	4.25	f	2.92E+03	2.92E+03	2.92E+03	2.92E+03	2.92E+03	2.92E+03	2.92E+03	2.92E+03	2.92E+03	2.92E+03	2.92E+03	2.92E+03
te126	4.25	f	5.35E+01	5.35E+01	5.35E+01	5.35E+01	5.35E+01	5.35E+01	5.36E+01	5.39E+01	5.40E+01	5.40E+01	5.40E+01	5.41E+01
sb127	4.25	f	3.51E+01	3.51E+01	3.51E+01	3.50E+01	3.34E+01	2.96E+01	1.73E+01	1.60E-01	3.25E-06	2.98E-13	9.70E-28	0.00E+00
te127	4.25	f	3.53E+00	3.53E+00	3.53E+00	3.53E+00	3.49E+00	3.28E+00	2.19E+00	5.20E-01	3.45E-01	1.95E-01	5.99E-02	5.75E-04
te127m	4.25	f	1.68E+02	1.68E+02	1.68E+02	1.68E+02	1.68E+02	1.68E+02	1.67E+02	1.44E+02	9.86E+01	5.56E+01	1.71E+01	1.64E-01
i127	4.25	f	6.35E+03	6.35E+03	6.35E+03	6.35E+03	6.35E+03	6.36E+03	6.37E+03	6.41E+03	6.46E+03	6.50E+03	6.54E+03	6.56E+03
sn128	4.25	f	6.10E-01	6.10E-01	4.29E-01	3.02E-01	2.19E-03	2.82E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb128	4.25	f	5.93E-01	5.93E-01	5.78E-01	5.62E-01	3.36E-01	9.78E-02	3.86E-04	5.51E-25	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te128	4.25	f	1.42E+04	1.42E+04	1.42E+04	1.42E+04	1.42E+04	1.42E+04	1.42E+04	1.42E+04	1.42E+04	1.42E+04	1.42E+04	1.42E+04
xe128	4.25	f	4.10E+02	4.10E+02	4.10E+02	4.10E+02	4.10E+02	4.10E+02	4.10E+02	4.10E+02	4.10E+02	4.10E+02	4.10E+02	4.10E+02
sb129	4.25	f	6.28E+00	6.28E+00	5.88E+00	5.43E+00	1.80E+00	1.45E-01	1.72E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te129	4.25	f	1.57E+00	1.57E+00	1.56E+00	1.52E+00	7.28E-01	2.43E-01	1.89E-01	1.10E-01	3.20E-02	5.00E-03	1.09E-04	3.12E-11
te129m	4.25	f	2.21E+02	2.21E+02	2.21E+02	2.21E+02	2.20E+02	2.18E+02	2.05E+02	1.20E+02	3.47E+01	5.42E+00	1.18E-01	3.38E-08
i129	4.25	f	2.84E+04	2.84E+04	2.84E+04	2.84E+04	2.84E+04	2.84E+04	2.84E+04	2.86E+04	2.87E+04	2.87E+04	2.87E+04	2.87E+04
xe129	4.25	f	2.45E+00	2.45E+00	2.45E+00	2.45E+00	2.45E+00	2.45E+00	2.46E+00	2.49E+00	2.50E+00	2.50E+00	2.50E+00	2.50E+00
te130	4.25	f	5.86E+04	5.86E+04	5.86E+04	5.86E+04	5.86E+04	5.86E+04	5.86E+04	5.86E+04	5.86E+04	5.86E+04	5.86E+04	5.86E+04
i130	4.25	f	1.35E+00	1.35E+00	1.32E+00	1.28E+00	8.71E-01	3.54E-01	6.25E-03	3.97E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe130	4.25	f	1.13E+03	1.13E+03	1.13E+03	1.13E+03	1.13E+03	1.13E+03	1.13E+03	1.13E+03	1.13E+03	1.13E+03	1.13E+03	1.13E+03
sb131	4.5	f	1.35E+00	1.35E+00	5.53E-01	2.24E-01	7.13E-07	1.94E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te131	4.25	f	1.58E+00	1.58E+00	1.21E+00	7.72E-01	7.02E-02	4.85E-02	9.17E-03	5.03E-09	1.79E-23	0.00E+00	0.00E+00	0.00E+00
te131m	4.25	f	2.69E+01	2.69E+01	2.67E+01	2.64E+01	2.25E+01	1.55E+01	2.94E+00	1.61E-06	5.72E-21	0.00E+00	0.00E+00	0.00E+00
i131	4.5	f	8.63E+02	8.63E+02	8.63E+02	8.63E+02	8.48E+02	8.02E+02	6.33E+02	6.77E+01	3.84E-01	1.63E-04	1.90E-11	0.00E+00
xe131	4.5	f	6.74E+04	6.74E+04	6.74E+04	6.74E+04	6.74E+04	6.75E+04	6.76E+04	6.82E+04	6.83E+04	6.83E+04	6.83E+04	6.83E+04
xe131m	4.25	f	1.76E+01	1.76E+01	1.76E+01	1.76E+01	1.75E+01	1.73E+01	1.63E+01	6.02E+00	2.38E-01	1.32E-03	2.74E-08	9.02E-27
te132	4.5	f	5.07E+02	5.07E+02	5.06E+02	5.03E+02	4.73E+02	4.10E+02	2.17E+02	8.56E-01	2.45E-06	1.18E-14	9.09E-32	0.00E+00
i132	4.25	f	1.51E+01	1.51E+01	1.51E+01	1.51E+01	1.42E+01	1.24E+01	6.52E+00	2.58E-02	7.39E-08	3.58E-16	2.74E-33	0.00E+00
xe132	4.25	f	1.76E+05	1.76E+05	1.76E+05	1.76E+05	1.76E+05	1.76E+05	1.76E+05	1.77E+05	1.77E+05	1.77E+05	1.77E+05	1.77E+05
te133	4.5	f	1.06E+00	1.05E+00	3.23E-01	1.29E-01	4.89E-04	2.97E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te133m	4.5	f	3.87E+00	3.87E+00	2.67E+00	1.83E+00	9.63E-03	5.83E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i133	4.5	f	1.96E+02	1.96E+02	1.94E+02	1.92E+02	1.54E+02	9.02E+01	8.17E+00	7.62E-09	1.10E-29	0.00E+00	0.00E+00	0.00E+00
xe133	4.5	f	1.13E+03	1.13E+03	1.13E+03	1.13E+03	1.13E+03	1.10E+03	8.10E+02	2.67E+01	9.55E-03	6.51E-08	1.51E-18	0.00E+00
xe133m	4.5	f	1.56E+01	1.56E+01	1.56E+01	1.56E+01	1.53E+01	1.41E+01	6.70E+00	1.89E-03	1.07E-11	4.53E-24	0.00E+00	0.00E+00
cs133	4.5	f	1.87E+05	1.87E+05	1.87E+05	1.87E+05	1.87E+05	1.87E+05	1.87E+05	1.88E+05	1.88E+05	1.88E+05	1.88E+05	1.88E+05
te134	4.5	f	5.81E+00	5.81E+00	3.53E+00	2.15E+00	2.02E-03	2.48E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Mass (grams) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Isotopic Mass (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
i134	4.5	f	9.17E+00	9.17E+00	8.10E+00	6.56E+00	5.73E-02	2.14E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe134	4.5	f	2.48E+05	2.48E+05	2.48E+05	2.48E+05	2.48E+05	2.48E+05	2.48E+05	2.48E+05	2.48E+05	2.48E+05	2.48E+05	2.48E+05
cs134	4.25	f	1.78E+04	1.78E+04	1.78E+04	1.78E+04	1.78E+04	1.77E+04	1.77E+04	1.73E+04	1.63E+04	1.51E+04	1.27E+04	6.48E+03
cs134m	4.25	f	5.99E-01	5.99E-01	5.32E-01	4.72E-01	8.94E-02	1.97E-03	7.08E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba134	4.25	f	8.17E+03	8.17E+03	8.17E+03	8.17E+03	8.17E+03	8.17E+03	8.17E+03	8.63E+03	9.55E+03	1.08E+04	1.32E+04	1.94E+04
i135	4.5	f	5.97E+01	5.97E+01	5.67E+01	5.38E+01	2.57E+01	4.74E+00	2.38E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe135	4.5	f	2.77E+01	2.77E+01	2.97E+01	3.16E+01	3.97E+01	2.21E+01	1.57E-01	4.64E-22	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs135	4.5	f	8.25E+04	8.25E+04	8.25E+04	8.25E+04	8.25E+04	8.25E+04	8.25E+04	8.25E+04	8.25E+04	8.25E+04	8.25E+04	8.25E+04
ba135	4.25	f	6.19E+01	6.19E+01	6.19E+01	6.19E+01	6.19E+01	6.20E+01	6.20E+01	6.20E+01	6.20E+01	6.20E+01	6.20E+01	6.20E+01
xe136	4.25	f	3.49E+05	3.49E+05	3.49E+05	3.49E+05	3.49E+05	3.49E+05	3.49E+05	3.49E+05	3.49E+05	3.49E+05	3.49E+05	3.49E+05
cs136	4.5	f	1.01E+02	1.01E+02	1.00E+02	1.00E+02	9.86E+01	9.55E+01	8.17E+01	2.07E+01	8.79E-01	7.64E-03	4.44E-07	8.63E-24
ba136	4.5	f	3.57E+03	3.57E+03	3.57E+03	3.57E+03	3.57E+03	3.57E+03	3.58E+03	3.64E+03	3.67E+03	3.67E+03	3.67E+03	3.67E+03
xe137	4.5	f	5.62E-01	5.60E-01	2.57E-03	1.11E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs137	4.5	f	1.99E+05	1.99E+05	1.99E+05	1.99E+05	1.99E+05	1.99E+05	1.99E+05	1.99E+05	1.99E+05	1.97E+05	1.95E+05	1.86E+05
ba137	4.5	f	8.63E+03	8.63E+03	8.63E+03	8.63E+03	8.63E+03	8.63E+03	8.63E+03	9.02E+03	9.70E+03	1.08E+04	1.31E+04	2.20E+04
xe138	4.5	f	1.95E+00	1.95E+00	4.46E-01	1.02E-01	1.07E-10	3.19E-31	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs138	4.5	f	4.83E+00	4.83E+00	3.57E+00	2.10E+00	2.70E-04	2.87E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba138	4.5	f	2.11E+05	2.11E+05	2.11E+05	2.11E+05	2.11E+05	2.11E+05	2.11E+05	2.11E+05	2.11E+05	2.11E+05	2.11E+05	2.11E+05
la138	4.25	f	1.38E+00	1.38E+00	1.38E+00	1.38E+00	1.38E+00	1.38E+00	1.38E+00	1.38E+00	1.38E+00	1.38E+00	1.38E+00	1.38E+00
cs139	4.5	f	1.31E+00	1.31E+00	1.46E-01	1.55E-02	3.56E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba139	4.5	f	1.22E+01	1.22E+01	1.06E+01	8.40E+00	2.70E-01	1.04E-04	4.47E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la139	4.5	f	1.99E+05	1.99E+05	1.99E+05	1.99E+05	1.99E+05	1.99E+05	1.99E+05	1.99E+05	1.99E+05	1.99E+05	1.99E+05	1.99E+05
ba140	4.5	f	2.67E+03	2.67E+03	2.67E+03	2.67E+03	2.63E+03	2.54E+03	2.15E+03	5.24E+02	2.01E+01	1.51E-01	6.41E-06	3.68E-23
la140	4.5	f	3.77E+02	3.77E+02	3.77E+02	3.77E+02	3.74E+02	3.65E+02	3.21E+02	7.95E+01	3.05E+00	2.29E-02	9.70E-07	5.58E-24
ce140	4.5	f	2.11E+05	2.11E+05	2.11E+05	2.11E+05	2.11E+05	2.11E+05	2.12E+05	2.13E+05	2.14E+05	2.14E+05	2.14E+05	2.14E+05
ba141	4.5	f	2.42E+00	2.42E+00	7.87E-01	2.53E-01	3.03E-08	4.58E-24	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la141	4.5	f	3.14E+01	3.14E+01	3.03E+01	2.83E+01	8.25E+00	4.89E-01	1.44E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce141	4.5	f	6.30E+03	6.30E+03	6.30E+03	6.30E+03	6.27E+03	6.20E+03	5.81E+03	3.34E+03	9.32E+02	1.36E+02	2.62E+00	4.48E-07
pr141	4.5	f	1.77E+05	1.77E+05	1.77E+05	1.77E+05	1.77E+05	1.77E+05	1.78E+05	1.80E+05	1.83E+05	1.83E+05	1.83E+05	1.83E+05
ba142	4.5	f	1.34E+00	1.34E+00	1.89E-01	2.66E-02	3.15E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la142	4.5	f	1.20E+01	1.20E+01	1.05E+01	8.56E+00	3.51E-01	2.35E-04	1.25E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce142	4.5	f	1.86E+05	1.86E+05	1.86E+05	1.86E+05	1.86E+05	1.86E+05	1.86E+05	1.86E+05	1.86E+05	1.86E+05	1.86E+05	1.86E+05
pr142	4.25	f	6.36E+00	6.36E+00	6.24E+00	6.13E+00	4.75E+00	2.66E+00	1.96E-01	2.93E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd142	4.25	f	3.17E+03	3.17E+03	3.17E+03	3.17E+03	3.17E+03	3.17E+03	3.17E+03	3.17E+03	3.17E+03	3.17E+03	3.17E+03	3.17E+03

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Mass (grams) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Isotopic Mass (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
la143	4.5	f	1.78E+00	1.78E+00	4.15E-01	9.55E-02	1.09E-10	3.98E-31	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce143	4.5	f	2.51E+02	2.51E+02	2.50E+02	2.48E+02	2.14E+02	1.53E+02	3.37E+01	6.84E-05	5.00E-18	9.86E-38	0.00E+00	0.00E+00
pr143	4.5	f	2.40E+03	2.40E+03	2.40E+03	2.40E+03	2.39E+03	2.38E+03	2.15E+03	5.78E+02	2.70E+01	2.71E-01	2.10E-05	1.29E-21
nd143	4.5	f	1.36E+05	1.36E+05	1.36E+05	1.36E+05	1.36E+05	1.37E+05	1.37E+05	1.38E+05	1.39E+05	1.39E+05	1.39E+05	1.39E+05
ce144	4.5	f	4.75E+04	4.75E+04	4.75E+04	4.75E+04	4.74E+04	4.74E+04	4.71E+04	4.42E+04	3.81E+04	3.06E+04	1.96E+04	3.31E+03
pr144	4.5	f	2.01E+00	2.01E+00	2.00E+00	2.00E+00	2.00E+00	1.99E+00	1.98E+00	1.86E+00	1.60E+00	1.29E+00	8.25E-01	1.39E-01
nd144	4.25	f	1.67E+05	1.67E+05	1.67E+05	1.67E+05	1.67E+05	1.67E+05	1.67E+05	1.70E+05	1.76E+05	1.83E+05	1.95E+05	2.11E+05
pr145	4.5	f	3.14E+01	3.14E+01	2.99E+01	2.82E+01	1.25E+01	1.96E+00	4.69E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd145	4.5	f	1.12E+05	1.12E+05	1.12E+05	1.12E+05	1.12E+05	1.12E+05	1.12E+05	1.12E+05	1.12E+05	1.12E+05	1.12E+05	1.12E+05
ce146	4.5	f	9.47E-01	9.47E-01	2.05E-01	4.39E-02	1.96E-11	8.25E-33	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr146	4.5	f	1.70E+00	1.70E+00	1.17E+00	5.91E-01	4.03E-06	4.35E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd146	4.5	f	1.13E+05	1.13E+05	1.13E+05	1.13E+05	1.13E+05	1.13E+05	1.13E+05	1.13E+05	1.13E+05	1.13E+05	1.13E+05	1.13E+05
pm146	4.25	f	1.18E+00	1.18E+00	1.18E+00	1.18E+00	1.18E+00	1.18E+00	1.18E+00	1.16E+00	1.14E+00	1.11E+00	1.04E+00	8.10E-01
sm146	4.25	f	1.09E+00	1.09E+00	1.09E+00	1.09E+00	1.09E+00	1.09E+00	1.09E+00	1.10E+00	1.11E+00	1.12E+00	1.14E+00	1.22E+00
pr147	4.5	f	7.62E-01	7.62E-01	1.77E-01	3.84E-02	1.94E-11	1.09E-32	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd147	4.5	f	8.94E+02	8.94E+02	8.94E+02	8.94E+02	8.79E+02	8.40E+02	6.95E+02	1.34E+02	3.05E+00	1.04E-02	8.71E-08	8.17E-28
pm147	4.5	f	2.70E+04	2.70E+04	2.70E+04	2.70E+04	2.70E+04	2.70E+04	2.71E+04	2.72E+04	2.62E+04	2.45E+04	2.15E+04	1.27E+04
sm147	4.5	f	1.41E+04	1.41E+04	1.41E+04	1.41E+04	1.41E+04	1.41E+04	1.41E+04	1.47E+04	1.58E+04	1.74E+04	2.06E+04	2.93E+04
nd148	4.5	f	5.94E+04	5.94E+04	5.94E+04	5.94E+04	5.94E+04	5.94E+04	5.94E+04	5.94E+04	5.94E+04	5.94E+04	5.94E+04	5.94E+04
pm148	4.25	f	1.23E+02	1.23E+02	1.23E+02	1.22E+02	1.18E+02	1.08E+02	7.39E+01	3.28E+00	2.77E-01	6.08E-02	2.71E-03	1.28E-08
pm148m	4.5	f	1.82E+02	1.82E+02	1.81E+02	1.81E+02	1.80E+02	1.79E+02	1.70E+02	1.10E+02	4.00E+01	8.86E+00	3.95E-01	1.86E-06
sm148	4.25	f	2.08E+04	2.08E+04	2.08E+04	2.08E+04	2.08E+04	2.08E+04	2.09E+04	2.09E+04	2.10E+04	2.11E+04	2.11E+04	2.11E+04
nd149	4.25	f	3.41E+00	3.41E+00	2.84E+00	2.33E+00	1.40E-01	2.25E-04	6.14E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm149	4.25	f	1.63E+02	1.63E+02	1.63E+02	1.62E+02	1.50E+02	1.21E+02	4.75E+01	1.38E-02	9.40E-11	5.29E-23	0.00E+00	0.00E+00
sm149	4.5	f	3.66E+02	3.66E+02	3.67E+02	3.67E+02	3.82E+02	4.10E+02	4.84E+02	5.30E+02	5.30E+02	5.30E+02	5.30E+02	5.30E+02
nd150	4.25	f	2.83E+04	2.83E+04	2.83E+04	2.83E+04	2.83E+04	2.83E+04	2.83E+04	2.83E+04	2.83E+04	2.83E+04	2.83E+04	2.83E+04
sm150	4.25	f	5.21E+04	5.21E+04	5.21E+04	5.21E+04	5.21E+04	5.21E+04	5.21E+04	5.21E+04	5.21E+04	5.21E+04	5.21E+04	5.21E+04
pm151	4.25	f	2.96E+01	2.96E+01	2.95E+01	2.92E+01	2.46E+01	1.67E+01	2.87E+00	6.93E-07	3.73E-22	0.00E+00	0.00E+00	0.00E+00
sm151	4.5	f	2.58E+03	2.58E+03	2.58E+03	2.58E+03	2.59E+03	2.60E+03	2.61E+03	2.61E+03	2.61E+03	2.61E+03	2.59E+03	2.55E+03
eu151	4.5	f	3.89E+00	3.89E+00	3.89E+00	3.90E+00	3.91E+00	3.94E+00	4.11E+00	5.54E+00	8.86E+00	1.38E+01	2.39E+01	6.36E+01
sm152	4.25	f	2.06E+04	2.06E+04	2.06E+04	2.06E+04	2.06E+04	2.06E+04	2.06E+04	2.06E+04	2.06E+04	2.06E+04	2.06E+04	2.06E+04
eu152	4.5	f	8.79E+00	8.79E+00	8.79E+00	8.79E+00	8.79E+00	8.79E+00	8.79E+00	8.71E+00	8.63E+00	8.56E+00	8.33E+00	7.51E+00
gd152	4.5	f	2.15E+03	2.15E+03	2.15E+03	2.15E+03	2.15E+03	2.15E+03	2.15E+03	2.15E+03	2.15E+03	2.15E+03	2.15E+03	2.15E+03
sm153	4.25	f	1.20E+02	1.20E+02	1.19E+02	1.18E+02	1.06E+02	8.40E+01	2.85E+01	2.48E-03	1.07E-12	9.47E-27	0.00E+00	0.00E+00

Mass (grams) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Isotopic Mass (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
eu153	4.25	l f	1.90E+04	1.90E+04	1.90E+04	1.90E+04	1.90E+04	1.90E+04	1.90E+04	1.91E+04	1.92E+04	1.92E+04	1.93E+04	1.93E+04
gd153	4.25	l f	2.31E+02	2.31E+02	2.31E+02	2.31E+02	2.31E+02	2.30E+02	2.28E+02	2.12E+02	1.79E+02	1.38E+02	8.13E+01	9.97E+00
sm154	4.25	f	5.58E+03	5.58E+03	5.58E+03	5.58E+03	5.58E+03	5.58E+03	5.58E+03	5.58E+03	5.58E+03	5.58E+03	5.58E+03	5.58E+03
eu154	4.25	l f	3.92E+03	3.92E+03	3.92E+03	3.92E+03	3.92E+03	3.92E+03	3.92E+03	3.89E+03	3.84E+03	3.77E+03	3.61E+03	3.07E+03
gd154	4.5	l f	2.53E+04	2.53E+04	2.53E+04	2.53E+04	2.53E+04	2.54E+04	2.54E+04	2.54E+04	2.54E+04	2.55E+04	2.56E+04	2.62E+04
eu155	4.25	l f	8.81E+02	8.81E+02	8.81E+02	8.81E+02	8.81E+02	8.81E+02	8.81E+02	8.73E+02	8.49E+02	8.18E+02	7.60E+02	5.65E+02
gd155	4.5	l f	1.55E+02	1.55E+02	1.55E+02	1.55E+02	1.55E+02	1.56E+02	1.57E+02	1.66E+02	1.86E+02	2.16E+02	2.74E+02	4.65E+02
sm156	4.25	f	1.20E+00	1.20E+00	1.16E+00	1.12E+00	6.66E-01	2.05E-01	1.02E-03	1.05E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00
eu156	4.25	l f	4.96E+02	4.96E+02	4.96E+02	4.96E+02	4.89E+02	4.75E+02	4.15E+02	1.27E+02	8.17E+00	1.35E-01	2.86E-05	9.41E-20
gd156	4.5	l f	5.40E+05	5.40E+05	5.40E+05	5.40E+05	5.40E+05	5.40E+05	5.40E+05	5.40E+05	5.40E+05	5.40E+05	5.40E+05	5.40E+05
eu157	4.25	f	2.00E+00	2.00E+00	1.96E+00	1.93E+00	1.40E+00	6.72E-01	2.51E-02	1.06E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00
gd157	4.5	l f	1.51E+02	1.51E+02	1.51E+02	1.51E+02	1.52E+02	1.52E+02	1.53E+02	1.53E+02	1.53E+02	1.53E+02	1.53E+02	1.53E+02
gd158	4.25	l f	6.55E+05	6.55E+05	6.55E+05	6.55E+05	6.55E+05	6.55E+05	6.55E+05	6.55E+05	6.55E+05	6.55E+05	6.55E+05	6.55E+05
gd159	4.25	l f	2.58E+01	2.58E+01	2.54E+01	2.49E+01	1.91E+01	1.05E+01	7.16E-01	5.44E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tb159	4.25	l f	2.09E+04	2.09E+04	2.09E+04	2.09E+04	2.09E+04	2.09E+04	2.09E+04	2.09E+04	2.09E+04	2.09E+04	2.09E+04	2.09E+04
gd160	4.25	l f	3.49E+05	3.49E+05	3.49E+05	3.49E+05	3.49E+05	3.49E+05	3.49E+05	3.49E+05	3.49E+05	3.49E+05	3.49E+05	3.49E+05
tb160	4.25	l f	6.37E+02	6.37E+02	6.37E+02	6.37E+02	6.35E+02	6.31E+02	6.13E+02	4.78E+02	2.69E+02	1.13E+02	1.92E+01	1.74E-02
dy160	4.25	l f	2.34E+03	2.34E+03	2.34E+03	2.34E+03	2.34E+03	2.35E+03	2.37E+03	2.50E+03	2.71E+03	2.86E+03	2.96E+03	2.98E+03
tb161	4.25	l f	3.68E+01	3.68E+01	3.68E+01	3.67E+01	3.57E+01	3.33E+01	2.47E+01	1.81E+00	4.37E-03	5.18E-07	4.29E-15	0.00E+00
dy161	4.25	l f	2.52E+03	2.52E+03	2.52E+03	2.52E+03	2.52E+03	2.52E+03	2.53E+03	2.56E+03	2.56E+03	2.56E+03	2.56E+03	2.56E+03
dy162	4.25	l f	1.32E+03	1.32E+03	1.32E+03	1.32E+03	1.32E+03	1.32E+03	1.32E+03	1.32E+03	1.32E+03	1.32E+03	1.32E+03	1.32E+03
dy163	4.25	l f	7.64E+02	7.64E+02	7.64E+02	7.64E+02	7.64E+02	7.64E+02	7.64E+02	7.64E+02	7.64E+02	7.64E+02	7.64E+02	7.64E+02
dy164	4.25	l f	1.30E+02	1.30E+02	1.30E+02	1.30E+02	1.30E+02	1.30E+02	1.30E+02	1.30E+02	1.30E+02	1.30E+02	1.30E+02	1.30E+02
ho165	4.25	l f	1.24E+02	1.24E+02	1.24E+02	1.24E+02	1.24E+02	1.24E+02	1.24E+02	1.24E+02	1.24E+02	1.24E+02	1.24E+02	1.24E+02
er166	4.25	l f	1.77E+01	1.77E+01	1.77E+01	1.77E+01	1.77E+01	1.77E+01	1.77E+01	1.77E+01	1.77E+01	1.77E+01	1.77E+01	1.77E+01
hf176	4.5	l	1.18E+02	1.18E+02	1.18E+02	1.18E+02	1.18E+02	1.18E+02	1.18E+02	1.18E+02	1.18E+02	1.18E+02	1.18E+02	1.18E+02
hf177	4.5	l	1.28E+01	1.28E+01	1.28E+01	1.28E+01	1.28E+01	1.28E+01	1.28E+01	1.28E+01	1.28E+01	1.28E+01	1.28E+01	1.28E+01
hf178	4.5	l	2.73E+02	2.73E+02	2.73E+02	2.73E+02	2.73E+02	2.73E+02	2.73E+02	2.73E+02	2.73E+02	2.73E+02	2.73E+02	2.73E+02
hf179	4.5	l	1.94E+03	1.94E+03	1.94E+03	1.94E+03	1.94E+03	1.94E+03	1.94E+03	1.94E+03	1.94E+03	1.94E+03	1.94E+03	1.94E+03
hf180	4.25	l	3.61E+03	3.61E+03	3.61E+03	3.61E+03	3.61E+03	3.61E+03	3.61E+03	3.61E+03	3.61E+03	3.61E+03	3.61E+03	3.61E+03
hf181	4.25	l	1.40E+01	1.40E+01	1.40E+01	1.40E+01	1.39E+01	1.38E+01	1.31E+01	8.56E+00	3.21E+00	7.37E-01	3.56E-02	2.31E-07
ta181	4.25	l	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.60E+02	1.60E+02	1.65E+02	1.70E+02	1.73E+02	1.73E+02	1.73E+02
w182	4.5	l	9.93E+02	9.93E+02	9.93E+02	9.93E+02	9.93E+02	9.93E+02	9.93E+02	9.93E+02	9.93E+02	9.93E+02	9.93E+02	9.93E+02
re185	4.25	l	1.93E+01	1.93E+01	1.93E+01	1.93E+01	1.93E+01	1.93E+01	1.94E+01	2.02E+01	2.13E+01	2.23E+01	2.28E+01	2.30E+01

Mass (grams) per Full Core of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 39 GWd/MTU
Maximum Isotopic Mass (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
os186	4.25	l	1.58E+01	1.58E+01	1.58E+01	1.58E+01	1.58E+01	1.59E+01	1.59E+01	1.60E+01	1.60E+01	1.60E+01	1.60E+01	1.60E+01
re187	4.25	l	5.51E+02	5.51E+02	5.51E+02	5.51E+02	5.52E+02	5.52E+02	5.52E+02	5.52E+02	5.52E+02	5.52E+02	5.52E+02	5.52E+02
os188	4.25	l	1.10E+02	1.10E+02	1.10E+02	1.10E+02	1.10E+02	1.10E+02	1.11E+02	1.11E+02	1.11E+02	1.11E+02	1.11E+02	1.11E+02
u234	4.5	a	4.21E+03	4.21E+03	4.21E+03	4.21E+03	4.21E+03	4.21E+03	4.21E+03	4.22E+03	4.26E+03	4.31E+03	4.42E+03	4.84E+03
u235	4.5	a	1.84E+06	1.84E+06	1.84E+06	1.84E+06	1.84E+06	1.84E+06	1.84E+06	1.84E+06	1.84E+06	1.84E+06	1.84E+06	1.84E+06
u236	4.5	a	7.72E+05	7.72E+05	7.72E+05	7.72E+05	7.72E+05	7.72E+05	7.72E+05	7.72E+05	7.72E+05	7.72E+05	7.72E+05	7.72E+05
u237	4.5	a	1.22E+03	1.22E+03	1.22E+03	1.22E+03	1.18E+03	1.11E+03	8.10E+02	5.63E+01	1.25E-01	5.49E-03	5.35E-03	4.86E-03
np237	4.5	a	7.36E+04	7.36E+04	7.36E+04	7.36E+04	7.36E+04	7.37E+04	7.40E+04	7.48E+04	7.49E+04	7.49E+04	7.49E+04	7.49E+04
u238	4.5	a	1.28E+08	1.28E+08	1.28E+08	1.28E+08	1.28E+08	1.28E+08	1.28E+08	1.28E+08	1.28E+08	1.28E+08	1.28E+08	1.28E+08
np238	4.25	a	1.81E+02	1.81E+02	1.80E+02	1.79E+02	1.63E+02	1.31E+02	4.90E+01	9.86E-03	2.87E-05	2.87E-05	2.86E-05	2.83E-05
pu238	4.25	a	2.66E+04	2.66E+04	2.66E+04	2.66E+04	2.67E+04	2.67E+04	2.67E+04	2.70E+04	2.74E+04	2.77E+04	2.81E+04	2.81E+04
u239	4.25	a	6.33E+01	6.33E+01	2.61E+01	1.08E+01	4.41E-05	2.13E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
np239	4.25	a	9.17E+03	9.17E+03	9.09E+03	9.09E+03	8.33E+03	6.86E+03	2.83E+03	1.36E+00	1.35E-02	1.35E-02	1.35E-02	1.34E-02
pu239	4.5	a	7.79E+05	7.79E+05	7.79E+05	7.79E+05	7.79E+05	7.79E+05	7.87E+05	7.87E+05	7.87E+05	7.87E+05	7.87E+05	7.87E+05
pu240	4.25	a	3.43E+05	3.43E+05	3.43E+05	3.43E+05	3.43E+05	3.43E+05	3.43E+05	3.43E+05	3.43E+05	3.43E+05	3.43E+05	3.44E+05
pu241	4.25	a	1.85E+05	1.85E+05	1.85E+05	1.85E+05	1.85E+05	1.85E+05	1.85E+05	1.84E+05	1.83E+05	1.81E+05	1.76E+05	1.60E+05
am241	4.5	a	7.40E+03	7.40E+03	7.40E+03	7.40E+03	7.41E+03	7.43E+03	7.50E+03	8.10E+03	9.55E+03	1.17E+04	1.60E+04	3.22E+04
pu242	4.25	a	7.27E+04	7.27E+04	7.27E+04	7.27E+04	7.27E+04	7.27E+04	7.27E+04	7.27E+04	7.27E+04	7.27E+04	7.27E+04	7.27E+04
am242m	4.5	a	1.58E+02	1.58E+02	1.58E+02	1.58E+02	1.58E+02	1.58E+02	1.58E+02	1.58E+02	1.57E+02	1.57E+02	1.57E+02	1.56E+02
am242	4.25	a	1.41E+01	1.41E+01	1.38E+01	1.35E+01	1.00E+01	5.00E+00	2.24E-01	2.04E-03	2.03E-03	2.03E-03	2.02E-03	2.01E-03
cm242	4.25	a	2.01E+03	2.01E+03	2.01E+03	2.01E+03	2.02E+03	2.01E+03	1.99E+03	1.78E+03	1.38E+03	9.40E+02	4.29E+02	1.96E+01
pu243	4.25	a	1.60E+01	1.60E+01	1.50E+01	1.40E+01	5.25E+00	5.60E-01	2.37E-05	1.51E-11	1.51E-11	1.51E-11	1.51E-11	1.51E-11
am243	4.25	a	1.57E+04	1.57E+04	1.57E+04	1.57E+04	1.57E+04	1.57E+04	1.57E+04	1.57E+04	1.57E+04	1.57E+04	1.57E+04	1.57E+04
cm243	4.25	a	5.57E+01	5.57E+01	5.57E+01	5.57E+01	5.57E+01	5.57E+01	5.57E+01	5.55E+01	5.54E+01	5.50E+01	5.43E+01	5.18E+01
cm244	4.25	a	4.81E+03	4.81E+03	4.81E+03	4.81E+03	4.81E+03	4.82E+03	4.82E+03	4.81E+03	4.78E+03	4.73E+03	4.64E+03	4.30E+03
cm245	4.25	a	1.80E+02	1.80E+02	1.80E+02	1.80E+02	1.80E+02	1.80E+02	1.80E+02	1.80E+02	1.80E+02	1.80E+02	1.80E+02	1.80E+02
cm246	4.25	a	2.44E+01	2.44E+01	2.44E+01	2.44E+01	2.44E+01	2.44E+01	2.44E+01	2.44E+01	2.44E+01	2.44E+01	2.44E+01	2.44E+01

Table 3.3

Activity (Curies) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
h 3	4.5	f	1.43E+02	1.43E+02	1.43E+02	1.43E+02	1.43E+02	1.43E+02	1.43E+02	1.43E+02	1.41E+02	1.39E+02	1.35E+02	1.21E+02
c 15	4.25	l	2.97E-02	2.24E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
n 16	4.25	l	2.73E-01	2.48E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ne 23	4.25	l	5.29E-02	5.19E-02	1.50E-16	4.11E-31	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
na 24	4.25	l	4.16E-01	4.16E-01	4.07E-01	3.97E-01	2.85E-01	1.34E-01	4.45E-03	6.81E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00
na 25	4.25	l	3.19E-02	3.16E-02	2.59E-11	2.09E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mg 27	4.25	l	1.61E+00	1.61E+00	1.79E-01	1.99E-02	8.63E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
al 28	4.25	l	3.16E+01	3.15E+01	2.95E-03	2.74E-07	8.78E-10	5.17E-10	4.75E-11	4.94E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i128	4.25	f	2.90E+03	2.90E+03	1.26E+03	5.49E+02	4.78E-03	1.29E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
al 29	4.25	l	8.38E-02	8.37E-02	3.53E-03	1.49E-04	8.30E-24	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
si 31	4.25	l	6.65E-01	6.65E-01	5.83E-01	5.11E-01	8.03E-02	1.17E-03	6.33E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ti 51	4.25	l	1.07E-01	1.07E-01	2.90E-03	7.84E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cr 51	4.25	l	9.20E+03	9.20E+03	9.19E+03	9.19E+03	9.12E+03	8.97E+03	8.32E+03	4.34E+03	9.68E+02	1.02E+02	9.90E-01	1.15E-08
v 52	4.25	l	2.05E+02	2.05E+02	8.02E-01	3.13E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
v 53	4.25	l	5.34E-01	5.30E-01	1.32E-06	3.22E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mn 54	4.25	l	6.65E+02	6.65E+02	6.65E+02	6.65E+02	6.65E+02	6.64E+02	6.59E+02	6.22E+02	5.45E+02	4.46E+02	2.96E+02	5.84E+01
cr 55	4.25	l	1.56E+02	1.55E+02	4.07E-01	1.06E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
fe 55	4.25	l	3.65E+03	3.65E+03	3.65E+03	3.65E+03	3.65E+03	3.65E+03	3.64E+03	3.58E+03	3.43E+03	3.22E+03	2.84E+03	1.71E+03
mn 56	4.25	l	1.95E+04	1.95E+04	1.70E+04	1.49E+04	2.27E+03	3.07E+01	1.21E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mn 57	4.25	l	2.35E+00	2.33E+00	1.44E-06	8.77E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
co 58	4.25	l	8.15E+02	8.15E+02	8.14E+02	8.14E+02	8.12E+02	8.07E+02	7.83E+02	6.07E+02	3.38E+02	1.40E+02	2.29E+01	1.82E-02
fe 59	4.25	l	2.18E+02	2.18E+02	2.18E+02	2.18E+02	2.17E+02	2.15E+02	2.05E+02	1.37E+02	5.37E+01	1.32E+01	7.38E-01	8.44E-06
ni 59	4.25	l	5.16E-01	5.16E-01	5.16E-01	5.16E-01	5.16E-01	5.16E-01	5.16E-01	5.16E-01	5.16E-01	5.16E-01	5.16E-01	5.16E-01
co 60	4.25	l	6.27E+02	6.27E+02	6.27E+02	6.27E+02	6.27E+02	6.27E+02	6.26E+02	6.21E+02	6.07E+02	5.88E+02	5.50E+02	4.23E+02
co 60m	4.25	l	8.40E+02	8.39E+02	1.15E+02	1.58E+01	1.33E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
co 61	4.25	l	1.68E+01	1.68E+01	1.36E+01	1.10E+01	5.82E-01	7.01E-04	5.13E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
co 62	4.25	l	8.50E-02	8.43E-02	8.12E-08	7.71E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ni 63	4.25	l	7.67E+01	7.67E+01	7.67E+01	7.67E+01	7.66E+01	7.66E+01	7.66E+01	7.66E+01	7.65E+01	7.64E+01	7.61E+01	7.51E+01
cu 64	4.25	l	1.01E+00	1.01E+00	9.85E-01	9.58E-01	6.54E-01	2.73E-01	5.37E-03	8.69E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ni 65	4.25	l	9.17E+01	9.17E+01	7.99E+01	6.97E+01	1.02E+01	1.25E-01	3.12E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cu 66	4.25	l	9.48E-01	9.46E-01	1.69E-02	1.09E-03	7.50E-04	6.12E-04	2.45E-04	8.92E-08	1.03E-15	1.28E-27	0.00E+00	0.00E+00
zn 72	4.25	f	2.56E+00	2.56E+00	2.55E+00	2.53E+00	2.28E+00	1.79E+00	6.13E-01	5.60E-05	2.66E-14	2.77E-28	0.00E+00	0.00E+00
ga 72	4.25	f	2.57E+00	2.57E+00	2.57E+00	2.57E+00	2.52E+00	2.23E+00	8.70E-01	8.03E-05	3.82E-14	3.97E-28	0.00E+00	0.00E+00
cu 73	4.5	f	3.65E+00	3.23E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zn 73	4.25	f	6.52E+00	6.43E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

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Table 3.3

Activity (Curies) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
ga 73	4.25	f	6.64E+00	6.64E+00	6.20E+00	5.77E+00	2.13E+00	2.17E-01	7.55E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 73m	4.25	f	6.58E+00	6.56E+00	6.12E+00	5.69E+00	2.10E+00	2.14E-01	7.46E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cu 74	4.5	f	4.23E+00	1.60E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zn 74	4.5	f	1.49E+01	1.48E+01	3.40E-05	7.68E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 74	4.5	f	4.80E+00	4.80E+00	4.42E-01	3.41E-02	9.17E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cu 75	4.5	f	4.98E+00	2.38E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zn 75	4.5	f	3.24E+01	3.05E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 75	4.5	f	4.04E+01	4.03E+01	2.17E-03	1.08E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 75	4.5	f	4.07E+01	4.07E+01	3.26E+01	2.53E+01	7.53E-01	2.43E-04	4.75E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cu 76	4.5	f	3.82E+00	2.78E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zn 76	4.5	f	6.54E+01	5.79E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 76	4.5	f	1.01E+02	1.00E+02	2.77E-15	6.85E-32	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 76	4.25	f	5.54E+00	5.54E+00	5.47E+00	5.40E+00	4.49E+00	2.95E+00	4.42E-01	3.23E-08	1.09E-24	0.00E+00	0.00E+00	0.00E+00
zn 77	4.5	f	8.17E+01	5.88E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 77	4.5	f	2.18E+02	2.10E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 77	4.5	f	8.69E+01	8.69E+01	8.44E+01	8.18E+01	5.32E+01	2.00E+01	2.41E-01	5.74E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 77m	4.5	f	2.24E+02	2.24E+02	1.72E-08	9.78E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 77	4.5	f	2.64E+02	2.64E+02	2.63E+02	2.61E+02	2.38E+02	1.87E+02	5.40E+01	7.87E-04	5.41E-15	9.13E-32	0.00E+00	0.00E+00
zn 78	4.5	f	1.13E+02	7.04E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 78	4.5	f	5.20E+02	4.65E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 78	4.5	f	9.12E+02	9.12E+02	7.20E+02	5.69E+02	2.08E+01	1.08E-02	1.80E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 78	4.5	f	9.26E+02	9.26E+02	9.02E+02	8.48E+02	1.04E+02	1.57E-01	1.75E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zn 79	4.5	f	5.11E+01	2.55E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 79	4.5	f	4.54E+02	3.68E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 79	4.5	f	1.50E+03	1.46E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 79	4.5	f	1.65E+03	1.64E+03	1.69E+02	1.68E+01	1.56E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 79m	4.5	f	1.64E+03	1.64E+03	2.90E+02	2.94E+01	2.73E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zn 80	4.25	f	2.20E+01	6.07E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 80	4.5	f	4.18E+02	2.79E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 80	4.5	f	3.45E+03	3.38E+03	1.50E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 80	4.5	f	4.13E+03	4.10E+03	3.10E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zn 81	4.25	f	5.16E+00	1.80E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 81	4.5	f	2.60E+02	1.48E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 81	4.5	f	3.66E+03	3.35E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 81	4.5	f	6.07E+03	6.01E+03	3.85E-13	2.03E-29	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.3

Activity (Curies) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
se 81	4.5	f	6.42E+03	6.42E+03	2.41E+03	9.63E+02	2.11E+00	1.89E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 81m	4.5	f	4.76E+02	4.76E+02	3.32E+02	2.31E+02	1.43E+00	1.28E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 82	4.5	f	1.33E+02	4.18E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 82	4.5	f	3.41E+03	2.94E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 82	4.5	f	5.62E+03	5.53E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 82m	4.5	f	2.21E+03	2.10E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 82	4.25	f	7.95E+02	7.95E+02	7.89E+02	7.81E+02	6.81E+02	4.97E+02	1.21E+02	5.78E-04	3.05E-16	0.00E+00	0.00E+00	0.00E+00
br 82m	4.25	f	7.01E+02	7.00E+02	2.36E+01	7.92E-01	1.88E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 83	4.25	f	2.14E+01	2.27E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 83	4.5	f	1.63E+03	1.13E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 83	4.5	f	8.83E+03	8.46E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 83	4.5	f	7.02E+03	7.01E+03	2.77E+03	1.09E+03	2.34E-03	2.56E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 83m	4.5	f	7.15E+03	7.14E+03	1.62E-04	2.99E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 83	4.5	f	1.45E+04	1.45E+04	1.32E+04	1.17E+04	1.57E+03	1.55E+01	1.44E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 83m	4.5	f	1.47E+04	1.47E+04	1.46E+04	1.42E+04	4.10E+03	5.93E+01	6.08E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ga 84	4.5	f	9.44E+01	8.10E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 84	4.5	f	1.07E+03	6.05E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 84	4.5	f	6.69E+03	5.99E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 84	4.5	f	2.57E+04	2.56E+04	3.91E+01	5.88E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 84	4.5	f	2.64E+04	2.64E+04	1.53E+04	7.94E+03	8.39E-01	6.86E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 84m	4.5	f	7.41E+02	7.39E+02	2.32E+01	7.23E-01	6.13E-22	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 85	4.5	f	1.90E+02	1.18E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 85	4.5	f	3.67E+03	2.62E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 85	4.5	f	1.17E+04	1.14E+04	9.55E-14	7.53E-31	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 85m	4.5	f	1.02E+04	9.82E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 85	4.5	f	2.85E+04	2.85E+04	2.32E+01	1.65E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 85	4.5	f	2.49E+03	2.49E+03	2.49E+03	2.49E+03	2.49E+03	2.49E+03	2.49E+03	2.48E+03	2.45E+03	2.41E+03	2.33E+03	2.05E+03
kr 85m	4.5	f	2.88E+04	2.88E+04	2.69E+04	2.49E+04	8.44E+03	7.10E+02	1.03E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ge 86	4.5	f	4.04E+01	2.42E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 86	4.5	f	1.95E+03	9.10E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 86	4.5	f	2.73E+04	2.61E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 86	4.5	f	3.42E+04	3.41E+04	6.58E-06	9.57E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 86m	4.5	f	6.93E+03	5.94E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 86	4.25	f	4.64E+02	4.64E+02	4.63E+02	4.63E+02	4.58E+02	4.47E+02	4.00E+02	1.52E+02	1.63E+01	5.73E-01	5.83E-04	9.22E-16
rb 86m	4.25	f	3.80E+01	3.76E+01	5.03E-08	6.58E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.3

Activity (Curies) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements

Decay time Following Burnup to 58 GWd/MTU

Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
ge 87	4.5	f	1.96E+02	1.10E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 87	4.5	f	9.52E+02	1.06E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 87	4.5	f	1.57E+04	1.39E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 87	4.5	f	4.46E+04	4.42E+04	8.66E-06	1.60E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 87	4.5	f	5.70E+04	5.70E+04	4.38E+04	3.34E+04	7.35E+02	1.20E-01	1.10E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 87m	4.25	f	4.45E+00	4.45E+00	3.94E+00	3.48E+00	6.20E-01	1.20E-02	2.34E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
as 88	4.5	f	3.78E+02	2.19E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 88	4.5	f	8.85E+03	5.59E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 88	4.5	f	4.18E+04	4.04E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 88	4.5	f	7.80E+04	7.80E+04	6.91E+04	6.11E+04	1.11E+04	2.23E+02	5.16E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 88	4.5	f	8.02E+04	8.02E+04	7.50E+04	6.76E+04	1.24E+04	2.48E+02	5.76E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr 89	4.25	l	8.71E+01	8.71E+01	8.67E+01	8.64E+01	8.12E+01	7.05E+01	3.73E+01	1.50E-01	4.48E-07	2.30E-15	2.28E-32	0.00E+00
as 89	4.25	f	1.15E+01	3.73E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 89	4.5	f	3.10E+03	5.71E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 89	4.5	f	2.98E+04	2.57E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 89	4.5	f	9.44E+04	9.42E+04	1.35E+02	1.91E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 89	4.5	f	1.03E+05	1.03E+05	3.26E+04	8.30E+03	3.99E-05	3.88E-24	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 89	4.5	l f	1.07E+05	1.07E+05	1.07E+05	1.07E+05	1.06E+05	1.05E+05	1.01E+05	7.07E+04	3.11E+04	9.04E+03	7.13E+02	3.19E-02
y 89m	4.25	l f	3.10E+02	3.01E+02	9.62E+01	9.58E+01	9.06E+01	7.98E+01	4.64E+01	6.73E+00	2.89E+00	8.41E-01	6.63E-02	2.96E-06
se 90	4.5	f	7.19E+02	1.41E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 90	4.5	f	1.68E+04	1.18E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 90	4.5	f	9.99E+04	9.80E+04	1.74E-12	2.95E-29	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 90	4.5	f	9.33E+04	9.32E+04	4.86E+01	1.32E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 90m	4.5	f	3.15E+04	3.15E+04	2.64E+02	2.09E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 90	4.5	f	2.24E+04	2.24E+04	2.24E+04	2.24E+04	2.24E+04	2.24E+04	2.24E+04	2.23E+04	2.23E+04	2.21E+04	2.18E+04	2.08E+04
y 90	4.5	l f	2.36E+04	2.36E+04	2.36E+04	2.36E+04	2.35E+04	2.33E+04	2.28E+04	2.24E+04	2.23E+04	2.21E+04	2.19E+04	2.08E+04
zr 90m	4.25	f	2.45E+01	1.04E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 91	4.5	f	7.99E+01	6.09E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 91	4.5	f	5.17E+03	1.64E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 91	4.5	f	6.90E+04	6.39E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 91	4.5	f	1.29E+05	1.28E+05	7.43E-05	3.89E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 91	4.5	l f	1.41E+05	1.41E+05	1.36E+05	1.31E+05	7.86E+04	2.45E+04	1.30E+02	2.40E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 91	4.5	l f	1.44E+05	1.44E+05	1.44E+05	1.44E+05	1.44E+05	1.43E+05	1.39E+05	1.02E+05	5.00E+04	1.72E+04	1.92E+03	3.35E-01
y 91m	4.5	f	8.15E+04	8.15E+04	8.10E+04	7.97E+04	4.99E+04	1.56E+04	8.23E+01	1.52E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00
se 92	4.25	f	7.00E+00	1.13E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.3
Activity (Curies) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
br 92	4.5	f	9.37E+02	1.40E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 92	4.5	f	3.77E+04	2.60E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 92	4.5	f	1.16E+05	1.04E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 92	4.5	f	1.55E+05	1.55E+05	1.36E+05	1.20E+05	2.00E+04	3.34E+02	3.36E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 92	4.5	l f	1.56E+05	1.56E+05	1.55E+05	1.52E+05	7.28E+04	4.93E+03	4.51E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr 93	4.25	l	6.27E-02	6.27E-02	6.27E-02	6.27E-02	6.27E-02	6.27E-02	6.27E-02	6.27E-02	6.27E-02	6.27E-02	6.27E-02	6.27E-02
br 93	4.25	f	3.98E+02	7.73E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 93	4.5	f	1.33E+04	7.76E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 93	4.5	f	1.02E+05	9.18E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 93	4.5	l f	1.81E+05	1.81E+05	1.11E+04	6.72E+02	6.25E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 93	4.5	l f	1.23E+05	1.23E+05	1.20E+05	1.16E+05	7.19E+04	2.40E+04	1.71E+02	4.32E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br 94	4.25	f	1.75E+01	3.32E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 94	4.5	f	6.43E+03	2.36E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 94	4.5	f	5.45E+04	4.25E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 94	4.5	f	1.84E+05	1.83E+05	1.16E-02	7.20E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 94	4.5	l f	2.00E+05	2.00E+05	7.02E+04	2.31E+04	4.00E-03	1.41E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr 95	4.25	f	7.80E+02	3.20E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 95	4.5	f	2.69E+04	4.75E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 95	4.5	f	1.66E+05	1.62E+05	4.33E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 95	4.5	f	2.12E+05	2.12E+05	3.02E+04	4.17E+03	3.79E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr 95	4.5	l f	2.33E+05	2.33E+05	2.33E+05	2.33E+05	2.32E+05	2.30E+05	2.23E+05	1.68E+05	8.79E+04	3.32E+04	4.46E+03	1.64E+00
nb 95	4.5	l f	2.34E+05	2.34E+05	2.34E+05	2.34E+05	2.34E+05	2.34E+05	2.33E+05	2.17E+05	1.47E+05	6.51E+04	9.62E+03	3.61E+00
nb 95m	4.5	l f	2.59E+03	2.59E+03	2.59E+03	2.59E+03	2.59E+03	2.59E+03	2.55E+03	1.98E+03	1.03E+03	3.90E+02	5.25E+01	1.93E-02
kr 96	4.25	f	1.27E+02	1.18E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 96	4.5	f	6.76E+03	2.29E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 96	4.5	f	1.23E+05	6.48E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 96	4.5	f	2.08E+05	1.95E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb 96	4.25	l f	4.78E+02	4.78E+02	4.71E+02	4.64E+02	3.78E+02	2.35E+02	2.77E+01	2.50E-07	6.80E-26	0.00E+00	0.00E+00	0.00E+00
rb 97	4.5	f	1.61E+03	2.82E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 97	4.5	f	6.20E+04	1.20E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 97	4.5	f	1.72E+05	1.45E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr 97	4.5	l f	2.43E+05	2.43E+05	2.38E+05	2.34E+05	1.75E+05	9.09E+04	4.74E+03	3.63E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb 97	4.5	l f	2.45E+05	2.45E+05	2.44E+05	2.41E+05	1.88E+05	9.13E+04	4.76E+03	3.92E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb 97m	4.5	l f	2.30E+05	2.30E+05	2.26E+05	2.22E+05	1.66E+05	8.62E+04	4.50E+03	3.46E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 98	4.25	f	2.21E+02	5.22E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.3

Activity (Curies) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr			0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
sr 98	4.5		f	2.69E+04	9.23E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 98	4.5		f	1.28E+05	5.32E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr 98	4.5		f	2.34E+05	2.31E+05	5.36E-13	1.19E-30	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb 98	4.5		f	2.37E+05	2.36E+05	5.91E-13	1.30E-30	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb 98m	4.25		f	2.12E+03	2.12E+03	1.41E+03	9.41E+02	3.23E+00	7.51E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 99	4.25		f	6.59E+00	5.07E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 99	4.5		f	9.12E+03	7.03E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 99	4.5		f	8.07E+04	5.14E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr 99	4.5		f	2.34E+05	1.86E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb 99	4.5		f	1.51E+05	1.50E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb 99m	4.5		f	1.04E+05	1.04E+05	3.56E+01	1.19E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo 99	4.5	l	f	2.61E+05	2.61E+05	2.59E+05	2.58E+05	2.40E+05	2.02E+05	9.50E+04	1.35E+02	3.60E-05	4.96E-15	0.00E+00	0.00E+00
tc 99	4.5		f	3.97E+00	3.97E+00	3.97E+00	3.97E+00	3.97E+00	3.98E+00	3.98E+00	3.98E+00	3.98E+00	3.98E+00	3.98E+00	3.98E+00
tc 99m	4.5		f	2.32E+05	2.32E+05	2.32E+05	2.31E+05	2.24E+05	1.95E+05	9.20E+04	1.30E+02	3.48E-05	4.80E-15	0.00E+00	0.00E+00
sr100	4.25		f	1.57E+03	5.03E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y100	4.25		f	2.92E+04	1.15E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr100	4.5		f	2.25E+05	2.06E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb100	4.5		f	2.48E+05	2.36E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb100m	4.25		f	2.37E+04	1.87E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc100	4.25	l	f	1.04E+05	9.93E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr101	4.25		f	2.17E+02	6.05E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y101	4.5		f	1.25E+04	3.14E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr101	4.5		f	1.36E+05	9.83E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb101	4.5		f	2.28E+05	2.18E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo101	4.25	l	f	2.43E+05	2.43E+05	5.89E+04	1.42E+04	3.10E-05	4.99E-25	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc101	4.25	l	f	2.43E+05	2.43E+05	1.41E+05	5.29E+04	5.53E-04	1.56E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr102	4.5		f	3.20E+01	2.85E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y102	4.5		f	4.65E+03	2.15E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr102	4.5		f	9.18E+04	7.29E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb102	4.25		f	1.98E+05	1.50E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo102	4.25		f	2.38E+05	2.38E+05	3.79E+04	6.01E+03	3.89E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc102	4.25		f	2.38E+05	2.38E+05	3.82E+04	6.06E+03	3.92E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc102m	4.25		f	2.61E+02	2.60E+02	2.19E+00	1.84E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y103	4.5		f	1.30E+03	9.04E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr103	4.25		f	3.71E+04	2.19E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.3

Activity (Curies) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
nb103	4.25	f	1.55E+05	1.08E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo103	4.25	f	2.43E+05	2.42E+05	2.33E-03	2.17E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc103	4.25	f	2.47E+05	2.47E+05	1.17E-02	1.10E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru103	4.25	f	2.48E+05	2.48E+05	2.48E+05	2.48E+05	2.47E+05	2.44E+05	2.31E+05	1.46E+05	5.07E+04	1.03E+04	3.93E+02	9.83E-04
rh103m	4.25	f	2.48E+05	2.48E+05	2.48E+05	2.48E+05	2.46E+05	2.43E+05	2.31E+05	1.46E+05	5.06E+04	1.03E+04	3.92E+02	9.82E-04
y104	4.25	f	6.42E+01	2.90E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr104	4.25	f	1.21E+04	9.22E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb104	4.25	f	7.82E+04	6.91E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo104	4.25	f	2.04E+05	2.03E+05	1.98E-04	1.83E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc104	4.25	f	2.15E+05	2.15E+05	7.28E+04	2.34E+04	2.88E-03	4.65E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh104	4.25	f	1.98E+05	1.95E+05	1.43E+02	1.19E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh104m	4.25	f	1.45E+04	1.44E+04	1.20E+02	9.95E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr105	4.5	f	2.53E+03	6.18E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb105	4.25	f	3.24E+04	2.58E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo105	4.25	f	1.55E+05	1.53E+05	9.58E-11	5.70E-26	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc105	4.25	f	1.85E+05	1.85E+05	1.29E+04	8.34E+02	1.93E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru105	4.25	f	1.89E+05	1.89E+05	1.80E+05	1.66E+05	5.58E+04	4.59E+03	6.01E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh105	4.25	f	1.74E+05	1.74E+05	1.74E+05	1.74E+05	1.64E+05	1.25E+05	3.07E+04	1.50E-01	8.25E-14	2.28E-32	0.00E+00	0.00E+00
rh105m	4.25	f	5.36E+04	5.36E+04	5.12E+04	4.74E+04	1.59E+04	1.31E+03	1.71E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr106	4.25	f	1.87E+02	8.69E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb106	4.25	f	7.32E+03	3.71E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo106	4.25	f	9.33E+04	8.63E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc106	4.25	f	1.40E+05	1.39E+05	1.51E-10	1.33E-25	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru106	4.25	f	1.26E+05	1.26E+05	1.26E+05	1.26E+05	1.26E+05	1.26E+05	1.25E+05	1.19E+05	1.06E+05	8.99E+04	6.36E+04	1.63E+04
rh106	4.25	f	1.36E+05	1.36E+05	1.26E+05	1.26E+05	1.26E+05	1.26E+05	1.25E+05	1.19E+05	1.06E+05	8.99E+04	6.36E+04	1.63E+04
rh106m	4.25	f	4.69E+03	4.69E+03	4.00E+03	3.41E+03	3.63E+02	2.17E+00	2.16E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb107	4.25	f	1.38E+03	5.56E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo107	4.25	f	4.02E+04	3.31E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc107	4.25	f	9.95E+04	9.75E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru107	4.25	f	1.18E+05	1.18E+05	5.05E+02	1.97E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh107	4.25	f	1.18E+05	1.18E+05	5.56E+04	2.14E+04	3.19E-02	1.53E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd107m	4.25	f	1.78E+03	1.72E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb108	4.25	f	4.41E+01	2.53E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo108	4.25	f	6.47E+03	4.08E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc108	4.25	f	3.67E+04	3.27E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.3

Activity (Curies) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
ru108	4.25	f	7.74E+04	7.73E+04	8.09E+02	8.37E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh108	4.25	f	7.85E+04	7.84E+04	8.63E+02	8.92E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh108m	4.25	f	1.11E+03	1.11E+03	3.47E+01	1.08E+00	9.18E-22	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo109	4.25	f	6.76E+02	4.13E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc109	4.25	f	1.22E+04	7.63E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru109	4.25	f	5.11E+04	5.03E+04	1.71E-11	5.59E-27	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh109	4.25	f	5.88E+04	5.87E+04	3.06E-02	5.14E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh109m	4.25	f	2.94E+04	2.93E+04	1.31E-06	1.89E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd109	4.25	f	7.74E+04	7.74E+04	7.56E+04	7.37E+04	5.18E+04	2.30E+04	6.03E+02	1.17E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd109m	4.25	f	5.25E+02	5.24E+02	6.24E+00	7.39E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag109m	4.25	f	7.74E+04	7.74E+04	7.57E+04	7.38E+04	5.18E+04	2.30E+04	6.03E+02	3.55E-04	3.24E-04	2.83E-04	2.15E-04	7.18E-05
mo110	4.25	f	6.94E+01	5.40E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc110	4.25	f	2.02E+03	9.10E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru110	4.25	f	1.70E+04	1.63E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh110	4.25	f	2.58E+03	2.07E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh110m	4.25	f	1.96E+04	1.96E+04	3.79E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag110	4.25	f	4.66E+04	4.53E+04	1.74E+01	1.74E+01	1.74E+01	1.74E+01	1.73E+01	1.60E+01	1.36E+01	1.06E+01	6.33E+00	8.34E-01
ag110m	4.25	f	1.28E+03	1.28E+03	1.28E+03	1.28E+03	1.28E+03	1.28E+03	1.27E+03	1.18E+03	9.99E+02	7.78E+02	4.65E+02	6.13E+01
mo111	4.25	f	5.45E+00	1.23E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc111	4.25	f	3.82E+02	2.70E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru111	4.25	f	5.73E+03	3.82E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh111	4.25	f	1.07E+04	1.03E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd111	4.25	f	1.14E+04	1.14E+04	4.94E+03	2.21E+03	1.35E+02	1.80E+01	2.06E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd111m	4.25	f	4.71E+02	4.71E+02	4.43E+02	4.16E+02	1.72E+02	2.29E+01	2.62E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag111	4.25	f	1.16E+04	1.16E+04	1.16E+04	1.16E+04	1.13E+04	1.06E+04	8.01E+03	7.13E+02	2.68E+00	6.20E-04	2.03E-11	0.00E+00
ag111m	4.25	f	1.16E+04	1.16E+04	5.22E+03	2.38E+03	1.68E+02	2.24E+01	2.57E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd111m	4.25	f	9.55E+01	9.55E+01	6.23E+01	4.06E+01	1.02E-01	1.15E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc112	4.25	f	5.59E+01	1.15E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru112	4.25	f	1.80E+03	1.48E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh112	4.25	f	4.12E+03	3.20E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd112	4.25	f	5.01E+03	5.01E+03	4.93E+03	4.85E+03	3.85E+03	2.27E+03	2.12E+02	2.51E-07	6.34E-28	0.00E+00	0.00E+00	0.00E+00
ag112	4.25	f	5.03E+03	5.03E+03	5.02E+03	5.01E+03	4.38E+03	2.67E+03	2.49E+02	2.95E-07	7.45E-28	0.00E+00	0.00E+00	0.00E+00
in113m	4.25	l	5.27E+02	5.27E+02	5.27E+02	5.27E+02	5.26E+02	5.24E+02	5.14E+02	4.40E+02	3.06E+02	1.78E+02	5.84E+01	7.17E-01
sn113	4.25	l	5.27E+02	5.27E+02	5.27E+02	5.27E+02	5.26E+02	5.23E+02	5.14E+02	4.40E+02	3.06E+02	1.78E+02	5.84E+01	7.17E-01
sn113m	4.25	l	1.70E+02	1.70E+02	6.45E+01	2.44E+01	3.02E-05	9.44E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.3

Activity (Curies) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements

Decay time Following Burnup to 58 GWd/MTU

Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
tc113	4.25	f	1.13E+01	3.88E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru113	4.25	f	5.43E+02	4.32E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh113	4.25	f	1.94E+03	1.16E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd113	4.25	f	2.82E+03	2.81E+03	4.27E-03	6.34E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag113	4.25	f	2.74E+03	2.74E+03	2.58E+03	2.42E+03	9.81E+02	1.24E+02	1.14E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag113m	4.25	f	5.48E+02	5.48E+02	3.01E-03	4.49E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd113m	4.25	f	1.12E+01	1.12E+01	1.12E+01	1.12E+01	1.12E+01	1.12E+01	1.12E+01	1.12E+01	1.11E+01	1.10E+01	1.07E+01	9.69E+00
in114	4.25	l	1.36E+02	1.35E+02	8.61E+01	8.61E+01	8.57E+01	8.49E+01	8.14E+01	5.66E+01	2.44E+01	6.93E+00	5.18E-01	1.88E-05
in114m	4.25	l	9.00E+01	9.00E+01	8.99E+01	8.99E+01	8.95E+01	8.87E+01	8.51E+01	5.91E+01	2.55E+01	7.24E+00	5.41E-01	1.96E-05
ru114	4.25	f	1.72E+02	1.58E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh114	4.25	f	9.05E+02	6.57E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd114	4.25	f	2.06E+03	2.05E+03	4.29E-01	8.80E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag114	4.25	f	2.13E+03	2.12E+03	4.42E-01	9.09E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru115	4.25	f	3.60E+01	1.63E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh115	4.25	f	3.54E+02	3.27E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd115	4.25	f	1.20E+03	1.19E+03	7.21E-12	3.93E-26	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag115	4.25	f	9.53E+02	9.53E+02	3.48E+02	1.23E+02	5.86E-05	2.08E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag115m	4.25	f	3.99E+02	3.96E+02	3.70E-12	2.02E-26	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd115	4.25	l f	1.43E+03	1.43E+03	1.42E+03	1.41E+03	1.29E+03	1.05E+03	4.13E+02	1.27E-01	9.90E-10	6.85E-22	0.00E+00	0.00E+00
cd115m	4.25	f	7.03E+01	7.03E+01	7.03E+01	7.03E+01	7.00E+01	6.93E+01	6.61E+01	4.41E+01	1.74E+01	4.29E+00	2.41E-01	2.82E-06
in115m	4.25	f	1.43E+03	1.43E+03	1.43E+03	1.43E+03	1.37E+03	1.14E+03	4.50E+02	1.43E-01	1.92E-03	4.74E-04	2.66E-05	3.12E-10
ru116	4.25	f	1.06E+01	7.04E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh116	4.25	f	1.56E+02	7.95E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd116	4.25	f	1.20E+03	1.15E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag116	4.25	f	1.39E+03	1.39E+03	6.39E-01	2.72E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag116m	4.25	f	1.82E+02	1.70E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in116	4.25	l f	2.83E+02	2.69E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in116m	4.25	f	1.06E+03	1.06E+03	7.25E+02	4.94E+02	2.28E+00	1.05E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh117	4.25	f	6.24E+01	3.55E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd117	4.25	f	8.40E+02	7.37E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag117	4.25	f	7.65E+02	7.61E+02	2.88E-05	1.03E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag117m	4.25	f	7.65E+02	7.19E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd117	4.25	f	1.34E+03	1.34E+03	1.17E+03	1.02E+03	1.46E+02	1.69E+00	3.34E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd117m	4.25	f	3.09E+02	3.09E+02	2.79E+02	2.52E+02	5.95E+01	2.19E+00	7.81E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in117	4.25	l f	1.00E+03	1.00E+03	9.94E+02	9.67E+02	3.06E+02	6.25E+00	1.01E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.3

Activity (Curies) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr			0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
in117m	4.25		f	1.23E+03	1.23E+03	1.21E+03	1.18E+03	3.53E+02	6.24E+00	4.16E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn117m	4.25	l	f	3.80E+03	3.80E+03	3.80E+03	3.79E+03	3.74E+03	3.61E+03	3.11E+03	8.24E+02	3.87E+01	3.94E-01	3.13E-05	2.12E-21
rh118	4.25		f	1.51E+01	1.79E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd118	4.25		f	4.11E+02	3.30E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag118	4.25		f	6.89E+02	6.34E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag118m	4.25		f	4.89E+02	3.99E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd118	4.25		f	1.21E+03	1.21E+03	8.02E+02	5.30E+02	1.63E+00	2.92E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in118	4.25	l	f	1.21E+03	1.21E+03	8.04E+02	5.31E+02	1.63E+00	2.93E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh119	4.25		f	4.45E+00	1.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd119	4.25		f	1.70E+02	1.15E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag119	4.25		f	7.07E+02	5.47E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd119	4.25		f	8.14E+02	8.13E+02	3.62E-01	1.59E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd119m	4.25		f	4.18E+02	4.17E+02	3.31E-02	2.59E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in119	4.25	l	f	5.47E+02	5.47E+02	8.63E+00	2.49E+00	2.35E-07	2.07E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in119m	4.25		f	7.39E+02	7.39E+02	2.74E+02	8.62E+01	8.16E-06	7.19E-22	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn119m	4.25	l	f	3.85E+03	3.85E+03	3.85E+03	3.85E+03	3.85E+03	3.84E+03	3.81E+03	3.58E+03	3.11E+03	2.52E+03	1.62E+03	2.88E+02
pd120	4.25		f	1.10E+02	9.22E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag120	4.25		f	5.18E+02	3.31E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd120	4.25		f	1.17E+03	1.16E+03	2.58E-08	5.52E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in120	4.25		f	1.19E+03	1.19E+03	2.75E-08	5.87E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in120m	4.25		f	2.20E+01	2.17E+01	4.13E-11	7.65E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd121	4.25		f	4.52E+01	1.54E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag121	4.25		f	3.64E+02	1.68E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd121	4.25		f	1.15E+03	1.11E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in121	4.25		f	1.16E+02	1.13E+02	7.85E-02	3.69E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in121m	4.25		f	1.18E+03	1.18E+03	5.89E+00	2.77E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn121	4.25	l	f	2.31E+03	2.31E+03	2.28E+03	2.25E+03	1.89E+03	1.25E+03	1.99E+02	8.96E-01	8.93E-01	8.91E-01	8.86E-01	8.63E-01
sn121m	4.25	l		3.97E-01	3.97E-01	3.97E-01	3.97E-01	3.97E-01	3.97E-01	3.97E-01	3.97E-01	3.96E-01	3.95E-01	3.92E-01	3.83E-01
pd122	4.25		f	1.47E+01	8.99E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag122	4.25		f	1.86E+02	5.22E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd122	4.25		f	1.23E+03	1.09E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in122	4.25		f	1.38E+03	1.29E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in122m	4.25		f	1.49E+02	1.40E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb122	4.25	l	f	5.06E+02	5.06E+02	5.02E+02	5.00E+02	4.64E+02	3.91E+02	1.81E+02	2.28E-01	4.68E-08	4.33E-18	0.00E+00	0.00E+00
sb122m	4.25	l	f	3.02E+01	3.02E+01	2.16E-01	1.55E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.3

Activity (Curies) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr			0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
te123m	4.25	l	f	8.01E+00	8.01E+00	8.00E+00	8.00E+00	8.00E+00	7.96E+00	7.82E+00	6.74E+00	4.76E+00	2.83E+00	9.65E-01	1.40E-02
pd123	4.25		f	2.59E+00	2.56E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag123	4.25		f	7.58E+01	1.34E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd123	4.25		f	8.03E+02	7.45E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in123	4.25		f	1.04E+03	9.94E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in123m	4.25		f	2.84E+02	2.83E+02	1.51E-09	6.92E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn123	4.25	l	f	1.97E+02	1.97E+02	1.97E+02	1.97E+02	1.96E+02	1.95E+02	1.92E+02	1.67E+02	1.21E+02	7.48E+01	2.76E+01	5.49E-01
sn123m	4.25	l	f	1.34E+03	1.34E+03	8.03E+02	4.78E+02	3.35E-01	2.07E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag124	4.25		f	5.49E+01	3.90E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd124	4.25		f	1.17E+03	5.48E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in124	4.25		f	2.26E+03	1.97E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb124	4.25	l	f	2.39E+02	2.39E+02	2.39E+02	2.39E+02	2.38E+02	2.36E+02	2.29E+02	1.69E+02	8.48E+01	3.01E+01	3.56E+00	7.92E-04
sb124m	4.25	l	f	4.11E+00	4.08E+00	6.14E-06	9.11E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag125	4.25		f	2.27E+01	2.86E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd125	4.25		f	6.82E+02	4.38E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in125	4.25		f	1.27E+03	1.04E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in125m	4.25		f	9.99E+02	9.53E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn125	4.25	l	f	8.09E+02	8.09E+02	8.08E+02	8.07E+02	7.89E+02	7.52E+02	6.07E+02	9.36E+01	1.25E+00	1.94E-03	3.18E-09	0.00E+00
sn125m	4.25	l	f	3.08E+03	3.08E+03	3.49E+02	3.93E+01	2.06E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb125	4.25	l	f	2.47E+03	2.47E+03	2.47E+03	2.47E+03	2.47E+03	2.47E+03	2.47E+03	2.43E+03	2.33E+03	2.19E+03	1.92E+03	1.16E+03
te125m	4.25	l	f	5.65E+02	5.65E+02	5.65E+02	5.65E+02	5.65E+02	5.65E+02	5.65E+02	5.65E+02	5.55E+02	5.30E+02	4.69E+02	2.83E+02
ag126	4.25		f	1.13E+01	8.74E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd126	4.25		f	8.65E+02	2.20E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in126	4.25		f	3.13E+03	1.94E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb126	4.25	l	f	8.31E+01	8.31E+01	8.31E+01	8.29E+01	8.16E+01	7.86E+01	6.65E+01	1.56E+01	5.67E-01	2.88E-02	2.53E-02	2.53E-02
sb126m	4.25	l	f	9.43E+01	9.42E+01	3.17E+01	1.07E+01	1.81E-01	1.81E-01	1.81E-01	1.81E-01	1.81E-01	1.81E-01	1.81E-01	1.81E-01
ag127	4.25		f	6.47E+00	1.23E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd127	4.25		f	6.90E+02	2.05E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in127	4.25		f	2.82E+03	1.62E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in127m	4.25		f	2.82E+03	2.37E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn127	4.25		f	5.62E+03	5.62E+03	4.77E+03	4.04E+03	4.01E+02	2.04E+00	9.74E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn127m	4.25		f	7.52E+03	7.51E+03	4.93E+01	3.21E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb127	4.25		f	1.39E+04	1.39E+04	1.38E+04	1.38E+04	1.32E+04	1.17E+04	6.81E+03	6.31E+01	1.28E-03	1.18E-10	3.84E-25	0.00E+00
te127	4.25	l	f	1.38E+04	1.38E+04	1.38E+04	1.38E+04	1.36E+04	1.28E+04	8.54E+03	2.04E+03	1.35E+03	7.63E+02	2.35E+02	2.26E+00
te127m	4.25	l	f	2.36E+03	2.36E+03	2.36E+03	2.36E+03	2.36E+03	2.36E+03	2.34E+03	2.02E+03	1.38E+03	7.79E+02	2.40E+02	2.31E+00

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Table 3.3

Activity (Curies) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
ag128	4.25	f	2.90E+00	1.83E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd128	4.25	f	5.82E+02	3.01E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in128	4.25	f	4.42E+03	2.26E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn128	4.25	f	2.08E+04	2.08E+04	1.46E+04	1.03E+04	7.45E+01	9.60E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb128	4.25	f	2.32E+03	2.32E+03	2.26E+03	2.20E+03	1.31E+03	3.84E+02	1.51E+00	2.15E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb128m	4.25	f	2.22E+04	2.22E+04	1.73E+04	1.24E+04	9.04E+01	1.17E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd129	4.25	f	2.94E+02	2.87E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in129	4.25	f	5.14E+03	1.65E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn129	4.25	f	1.89E+04	1.88E+04	1.25E+00	1.80E-04	1.32E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn129m	4.5	f	1.65E+04	1.65E+04	7.42E+02	3.33E+01	4.48E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb129	4.25	f	4.77E+04	4.77E+04	4.46E+04	4.12E+04	1.37E+04	1.10E+03	1.31E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te129	4.25	f	4.54E+04	4.54E+04	4.50E+04	4.39E+04	2.10E+04	7.03E+03	5.45E+03	3.19E+03	9.25E+02	1.45E+02	3.16E+00	9.02E-07
te129m	4.25	f	9.19E+03	9.19E+03	9.19E+03	9.19E+03	9.17E+03	9.05E+03	8.51E+03	4.98E+03	1.44E+03	2.25E+02	4.94E+00	1.41E-06
xe129m	4.25	f	2.23E+01	2.23E+01	2.22E+01	2.22E+01	2.17E+01	2.06E+01	1.63E+01	2.15E+00	2.00E-02	1.79E-05	9.55E-12	0.00E+00
cd130	4.25	f	1.05E+02	2.45E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in130	4.5	f	3.36E+03	4.20E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn130	4.5	f	4.60E+04	4.59E+04	1.72E+02	6.42E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb130	4.25	f	1.58E+04	1.58E+04	9.36E+03	5.53E+03	3.48E+00	1.68E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb130m	4.5	f	6.18E+04	6.18E+04	4.48E+03	1.73E+02	1.49E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i130	4.25	f	6.42E+03	6.42E+03	6.27E+03	6.10E+03	4.12E+03	1.68E+03	2.97E+01	1.89E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i130m	4.25	f	3.45E+03	3.44E+03	3.42E+02	3.39E+01	3.04E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cd131	4.25	f	1.69E+01	2.43E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
in131	4.5	f	1.56E+03	1.20E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn131	4.5	f	4.04E+04	3.97E+04	5.19E-10	6.56E-24	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb131	4.5	f	1.08E+05	1.08E+05	4.42E+04	1.79E+04	5.70E-02	1.55E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te131	4.5	f	1.18E+05	1.18E+05	9.07E+04	5.76E+04	5.70E+03	3.94E+03	7.46E+02	4.09E-04	1.45E-18	0.00E+00	0.00E+00	0.00E+00
te131m	4.25	f	3.04E+04	3.04E+04	3.01E+04	2.98E+04	2.53E+04	1.75E+04	3.32E+03	1.82E-03	6.45E-18	0.00E+00	0.00E+00	0.00E+00
i131	4.5	f	1.41E+05	1.41E+05	1.41E+05	1.41E+05	1.38E+05	1.32E+05	1.04E+05	1.11E+04	6.28E+01	2.68E-02	3.11E-09	0.00E+00
xe131m	4.25	f	2.15E+03	2.15E+03	2.15E+03	2.15E+03	2.13E+03	2.11E+03	1.97E+03	7.02E+02	2.75E+01	1.52E-01	3.14E-06	1.03E-24
in132	4.25	f	4.10E+02	9.86E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn132	4.25	f	3.32E+04	3.26E+04	9.48E-10	2.67E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb132	4.25	f	6.78E+04	6.76E+04	4.80E+02	3.39E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb132m	4.5	f	6.13E+04	6.12E+04	4.27E+01	2.54E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te132	4.5	f	2.01E+05	2.01E+05	2.01E+05	2.00E+05	1.88E+05	1.63E+05	8.61E+04	3.41E+02	9.75E-04	4.71E-12	3.61E-29	0.00E+00
i132	4.5	f	2.06E+05	2.06E+05	2.05E+05	2.05E+05	1.93E+05	1.68E+05	8.87E+04	3.51E+02	1.00E-03	4.86E-12	3.72E-29	0.00E+00

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Table 3.3

Activity (Curies) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
cs132	4.25	f	9.34E+00	9.34E+00	9.32E+00	9.30E+00	9.01E+00	8.39E+00	6.09E+00	3.77E-01	6.15E-04	4.05E-08	1.00E-16	0.00E+00
in133	4.25	f	1.45E+01	2.86E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn133	4.25	f	9.02E+03	5.57E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb133	4.5	f	8.70E+04	8.66E+04	2.13E+01	5.18E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te133	4.5	f	1.52E+05	1.52E+05	4.62E+04	1.84E+04	6.96E+01	4.23E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te133m	4.5	f	1.24E+05	1.24E+05	8.59E+04	5.90E+04	3.08E+02	1.87E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i133	4.5	f	2.85E+05	2.85E+05	2.83E+05	2.80E+05	2.24E+05	1.32E+05	1.19E+04	1.11E-05	1.60E-26	0.00E+00	0.00E+00	0.00E+00
i133m	4.25	f	2.30E+04	2.22E+04	8.78E+03	6.03E+03	3.15E+01	1.91E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe133	4.5	f	2.74E+05	2.74E+05	2.74E+05	2.74E+05	2.73E+05	2.65E+05	1.95E+05	6.43E+03	2.31E+00	1.57E-05	3.63E-16	0.00E+00
xe133m	4.25	f	9.22E+03	9.22E+03	9.21E+03	9.20E+03	9.03E+03	8.25E+03	3.93E+03	1.11E+00	6.25E-09	2.65E-21	0.00E+00	0.00E+00
sn134	4.25	f	1.62E+03	8.29E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb134	4.5	f	1.55E+04	7.35E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb134m	4.25	f	1.25E+04	1.17E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te134	4.5	f	2.39E+05	2.39E+05	1.46E+05	8.85E+04	8.36E+01	1.02E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i134	4.5	f	3.12E+05	3.12E+05	2.72E+05	2.20E+05	1.90E+03	7.08E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i134m	4.25	f	3.11E+04	3.10E+04	1.11E+02	3.96E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe134m	4.25	f	9.61E+03	1.53E+03	2.56E+00	9.13E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs134	4.25	f	5.58E+04	5.58E+04	5.58E+04	5.58E+04	5.58E+04	5.58E+04	5.56E+04	5.43E+04	5.14E+04	4.73E+04	3.99E+04	2.04E+04
cs134m	4.25	f	9.91E+03	9.91E+03	8.80E+03	7.81E+03	1.47E+03	3.27E+01	1.17E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sn135	4.25	f	1.39E+02	2.64E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb135	4.25	f	7.90E+03	5.28E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te135	4.5	f	1.37E+05	1.32E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i135	4.5	f	2.74E+05	2.74E+05	2.60E+05	2.46E+05	1.18E+05	2.18E+04	1.09E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe135	4.5	f	7.50E+04	7.50E+04	8.26E+04	8.90E+04	1.22E+05	7.02E+04	5.08E+02	1.50E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe135m	4.25	f	6.25E+04	6.25E+04	4.70E+04	4.14E+04	1.92E+04	3.55E+03	1.78E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs135m	4.25	f	1.38E+04	1.38E+04	9.33E+03	6.30E+03	2.60E+01	9.15E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba135m	4.25	f	1.82E+02	1.82E+02	1.80E+02	1.78E+02	1.50E+02	1.02E+02	1.79E+01	5.10E-06	3.98E-21	0.00E+00	0.00E+00	0.00E+00
sn136	4.25	f	1.21E+01	4.60E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sb136	4.25	f	1.28E+03	5.51E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te136	4.5	f	5.96E+04	5.73E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i136	4.5	f	1.21E+05	1.20E+05	4.36E-02	1.38E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i136m	4.5	f	5.99E+04	5.90E+04	1.68E-07	4.67E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs136	4.5	f	1.54E+04	1.54E+04	1.54E+04	1.53E+04	1.51E+04	1.46E+04	1.25E+04	3.17E+03	1.34E+02	1.17E+00	6.79E-05	1.32E-21
ba136m	4.5	f	1.77E+03	1.73E+03	1.72E+03	1.72E+03	1.69E+03	1.63E+03	1.40E+03	3.55E+02	1.50E+01	1.31E-01	7.61E-06	1.48E-22
sb137	4.5	f	7.20E+02	1.68E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.3

Activity (Curies) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
te137	4.5	f	2.07E+04	1.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i137	4.5	f	1.32E+05	1.29E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe137	4.5	f	2.61E+05	2.60E+05	1.19E+03	5.13E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs137	4.5	f	3.30E+04	3.30E+04	3.30E+04	3.30E+04	3.30E+04	3.30E+04	3.30E+04	3.30E+04	3.28E+04	3.27E+04	3.23E+04	3.08E+04
ba137m	4.5	f	3.14E+04	3.14E+04	3.12E+04	3.12E+04	3.12E+04	3.12E+04	3.12E+04	3.11E+04	3.10E+04	3.08E+04	3.05E+04	2.91E+04
sb138	4.25	f	1.83E+01	3.33E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te138	4.25	f	5.41E+03	3.30E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i138	4.5	f	6.81E+04	6.16E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe138	4.5	f	2.36E+05	2.35E+05	5.39E+04	1.23E+04	1.29E-05	3.85E-26	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs138	4.5	f	2.58E+05	2.58E+05	1.90E+05	1.12E+05	1.44E+01	1.53E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs138m	4.25	f	1.24E+04	1.23E+04	9.77E+00	7.69E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te139	4.25	f	8.49E+02	2.56E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i139	4.5	f	2.92E+04	2.17E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe139	4.5	f	1.69E+05	1.66E+05	3.79E-09	8.28E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs139	4.5	f	2.38E+05	2.38E+05	2.67E+04	2.83E+03	6.51E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba139	4.5	f	2.46E+05	2.46E+05	2.13E+05	1.69E+05	5.43E+03	2.09E+00	9.00E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te140	4.25	f	1.18E+02	5.41E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i140	4.5	f	8.51E+03	3.83E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe140	4.5	f	1.15E+05	1.10E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs140	4.5	f	2.13E+05	2.12E+05	7.64E-04	2.37E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba140	4.5	f	2.47E+05	2.47E+05	2.46E+05	2.46E+05	2.42E+05	2.34E+05	1.99E+05	4.83E+04	1.85E+03	1.39E+01	5.91E-04	3.39E-21
la140	4.5	f	2.70E+05	2.70E+05	2.70E+05	2.70E+05	2.67E+05	2.60E+05	2.26E+05	5.57E+04	2.13E+03	1.60E+01	6.81E-04	3.91E-21
pr140	4.25	f	7.32E+00	7.29E+00	1.59E-02	3.44E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i141	4.25	f	1.19E+03	2.64E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe141	4.5	f	4.56E+04	3.06E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs141	4.5	f	1.61E+05	1.58E+05	3.12E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba141	4.5	f	2.21E+05	2.21E+05	7.21E+04	2.31E+04	2.77E-03	4.18E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la141	4.5	f	2.24E+05	2.24E+05	2.16E+05	2.01E+05	5.90E+04	3.48E+03	1.03E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce141	4.5	f	2.25E+05	2.25E+05	2.25E+05	2.25E+05	2.24E+05	2.21E+05	2.08E+05	1.19E+05	3.31E+04	4.86E+03	9.35E+01	1.60E-05
i142	4.25	f	3.85E+02	1.22E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe142	4.5	f	1.86E+04	1.05E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs142	4.5	f	9.28E+04	6.64E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba142	4.5	f	2.08E+05	2.08E+05	2.93E+04	4.11E+03	4.86E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la142	4.5	f	2.17E+05	2.17E+05	1.90E+05	1.54E+05	6.33E+03	4.25E+00	2.26E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr142	4.25	f	1.72E+04	1.72E+04	1.69E+04	1.66E+04	1.29E+04	7.20E+03	5.29E+02	7.93E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.3
Activity (Curies) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
i143	4.25	f	3.87E+00	6.85E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe143	4.25	f	2.95E+03	1.43E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs143	4.5	f	4.52E+04	3.13E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba143	4.5	f	1.74E+05	1.67E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la143	4.5	f	2.01E+05	2.01E+05	4.70E+04	1.08E+04	1.24E-05	4.51E-26	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce143	4.5	f	2.04E+05	2.04E+05	2.03E+05	2.01E+05	1.73E+05	1.24E+05	2.73E+04	5.55E-02	4.06E-15	0.00E+00	0.00E+00	0.00E+00
pr143	4.5	f	1.97E+05	1.97E+05	1.97E+05	1.97E+05	1.96E+05	1.95E+05	1.76E+05	4.74E+04	2.21E+03	2.22E+01	1.72E-03	1.06E-19
xe144	4.25	f	6.53E+02	3.47E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs144	4.5	f	1.42E+04	7.40E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba144	4.5	f	1.31E+05	1.24E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la144	4.5	f	1.76E+05	1.75E+05	1.29E-08	7.19E-22	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce144	4.5	f	1.89E+05	1.89E+05	1.89E+05	1.89E+05	1.89E+05	1.88E+05	1.87E+05	1.76E+05	1.52E+05	1.22E+05	7.77E+04	1.31E+04
pr144	4.5	f	1.90E+05	1.90E+05	1.89E+05	1.89E+05	1.89E+05	1.88E+05	1.87E+05	1.76E+05	1.52E+05	1.22E+05	7.77E+04	1.31E+04
pr144m	4.5	f	2.65E+03	2.65E+03	2.64E+03	2.64E+03	2.64E+03	2.64E+03	2.62E+03	2.46E+03	2.12E+03	1.71E+03	1.09E+03	1.84E+02
xe145	4.25	f	6.98E+01	3.23E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs145	4.25	f	3.74E+03	1.19E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba145	4.5	f	6.02E+04	5.15E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la145	4.5	f	1.24E+05	1.22E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce145	4.5	f	1.40E+05	1.40E+05	1.61E+02	1.61E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr145	4.5	f	1.40E+05	1.40E+05	1.33E+05	1.26E+05	5.59E+04	8.75E+03	2.09E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe146	4.25	f	6.05E+00	1.76E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs146	4.5	f	7.26E+02	9.81E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba146	4.5	f	2.98E+04	2.18E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la146	4.5	f	7.97E+04	7.40E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce146	4.5	f	1.13E+05	1.13E+05	2.45E+04	5.26E+03	2.34E-06	9.87E-28	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr146	4.5	f	1.14E+05	1.14E+05	7.85E+04	3.96E+04	2.70E-01	2.91E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs147	4.25	f	1.91E+01	5.35E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba147	4.25	f	5.88E+03	2.19E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la147	4.5	f	3.67E+04	3.19E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce147	4.5	f	8.66E+04	8.59E+04	2.22E-05	5.45E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr147	4.5	f	9.15E+04	9.15E+04	2.13E+04	4.61E+03	2.33E-06	1.31E-27	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd147	4.5	f	9.29E+04	9.29E+04	9.28E+04	9.27E+04	9.10E+04	8.73E+04	7.22E+04	1.40E+04	3.17E+02	1.08E+00	9.01E-06	8.47E-26
pm147	4.5	f	3.14E+04	3.14E+04	3.14E+04	3.14E+04	3.14E+04	3.15E+04	3.16E+04	3.16E+04	3.05E+04	2.85E+04	2.50E+04	1.47E+04
cs148	4.25	f	3.51E+00	1.19E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba148	4.25	f	1.21E+03	3.85E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

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Table 3.3

Activity (Curies) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements

Decay time Following Burnup to 58 GWd/MTU

Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
la148	4.5	f	1.20E+04	6.54E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce148	4.5	f	6.18E+04	6.11E+04	1.31E-05	2.75E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr148	4.5	f	7.23E+04	7.23E+04	1.22E+01	1.28E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm148	4.5	f	2.88E+04	2.88E+04	2.87E+04	2.87E+04	2.76E+04	2.54E+04	1.73E+04	7.46E+02	5.61E+01	1.23E+01	5.50E-01	2.60E-06
pm148m	4.5	f	4.78E+03	4.78E+03	4.78E+03	4.78E+03	4.75E+03	4.70E+03	4.47E+03	2.89E+03	1.05E+03	2.33E+02	1.04E+01	4.92E-05
ba149	4.25	f	1.37E+02	5.04E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la149	4.25	f	3.61E+03	2.73E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce149	4.5	f	3.27E+04	2.90E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr149	4.25	f	5.15E+04	5.14E+04	5.34E+00	5.38E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd149	4.25	f	5.70E+04	5.70E+04	4.76E+04	3.90E+04	2.34E+03	3.77E+00	1.03E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm149	4.25	f	9.22E+04	9.22E+04	9.19E+04	9.16E+04	8.47E+04	6.88E+04	2.69E+04	7.77E+00	5.30E-08	2.99E-20	0.00E+00	0.00E+00
ba150	4.25	f	1.25E+01	6.05E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la150	4.25	f	6.50E+02	2.13E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce150	4.25	f	1.59E+04	1.34E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr150	4.25	f	3.38E+04	3.18E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm150	4.25	f	1.00E+03	1.00E+03	8.81E+02	7.74E+02	1.27E+02	2.02E+00	1.65E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la151	4.25	f	1.11E+02	4.23E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce151	4.25	f	4.82E+03	2.47E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr151	4.25	f	1.84E+04	1.78E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd151	4.25	f	3.12E+04	3.12E+04	5.96E+03	1.12E+03	7.69E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm151	4.25	f	3.15E+04	3.15E+04	3.13E+04	3.10E+04	2.61E+04	1.77E+04	3.05E+03	7.37E-04	3.96E-19	0.00E+00	0.00E+00	0.00E+00
sm151	4.5	f	9.85E+01	9.85E+01	9.85E+01	9.86E+01	9.87E+01	9.90E+01	9.95E+01	9.96E+01	9.95E+01	9.93E+01	9.89E+01	9.74E+01
ce152	4.25	f	6.58E+02	6.01E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr152	4.25	f	6.44E+03	5.88E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd152	4.25	f	2.11E+04	2.11E+04	3.42E+03	5.51E+02	4.47E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm152	4.25	f	2.19E+04	2.19E+04	5.26E+03	8.60E+02	6.99E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm152m	4.25	f	8.20E+02	8.19E+02	5.17E+01	3.25E+00	5.00E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
eu152m	4.5	f	3.63E+01	3.63E+01	3.50E+01	3.37E+01	2.00E+01	6.10E+00	2.88E-02	2.01E-22	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce153	4.25	f	2.29E+02	1.43E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr153	4.25	f	2.78E+03	2.41E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd153	4.25	f	1.23E+04	1.22E+04	1.18E-04	1.10E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm153	4.25	f	1.49E+04	1.49E+04	3.86E+02	8.20E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sm153	4.25	l f	1.04E+05	1.04E+05	1.03E+05	1.02E+05	9.22E+04	7.26E+04	2.47E+04	2.15E+00	9.25E-10	8.24E-24	0.00E+00	0.00E+00
gd153	4.25	l f	1.05E+03	1.05E+03	1.05E+03	1.05E+03	1.05E+03	1.05E+03	1.04E+03	9.67E+02	8.13E+02	6.28E+02	3.69E+02	4.54E+01
ce154	4.25	f	2.37E+01	1.68E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

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Table 3.3

Activity (Curies) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr			0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
pr154	4.25		f	5.78E+02	3.10E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd154	4.25		f	6.56E+03	6.45E+03	1.88E-10	5.28E-24	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm154	4.25		f	8.20E+03	8.19E+03	6.95E-02	3.89E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm154m	4.25		f	1.64E+03	1.64E+03	7.02E-01	2.99E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
eu154	4.25	l	f	2.41E+03	2.41E+03	2.41E+03	2.41E+03	2.41E+03	2.41E+03	2.41E+03	2.39E+03	2.36E+03	2.32E+03	2.22E+03	1.89E+03
gd155m	4.25	l		2.28E+01	4.22E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr155	4.25		f	1.27E+02	6.89E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd155	4.25		f	2.39E+03	2.30E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm155	4.25		f	5.31E+03	5.27E+03	3.50E-08	1.79E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sm155	4.25	l	f	6.80E+03	6.80E+03	2.77E+03	1.09E+03	2.33E-03	2.56E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
eu155	4.25	l	f	1.01E+03	1.01E+03	1.01E+03	1.01E+03	1.01E+03	1.01E+03	1.01E+03	1.00E+03	9.78E+02	9.42E+02	8.74E+02	6.50E+02
pr156	4.25		f	1.91E+01	3.11E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd156	4.25		f	9.18E+02	8.86E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm156	4.25		f	2.85E+03	2.75E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sm156	4.25		f	4.23E+03	4.23E+03	4.07E+03	3.93E+03	2.34E+03	7.20E+02	3.56E+00	3.70E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00
eu156	4.25	l	f	7.32E+04	7.32E+04	7.31E+04	7.31E+04	7.22E+04	7.00E+04	6.11E+04	1.87E+04	1.21E+03	1.98E+01	4.21E-03	1.39E-17
nd157	4.25		f	2.39E+02	1.81E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm157	4.25		f	1.36E+03	1.35E+03	1.91E-06	2.64E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sm157	4.25		f	2.73E+03	2.73E+03	2.23E+02	1.69E+01	3.65E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
eu157	4.25		f	5.95E+03	5.95E+03	5.84E+03	5.71E+03	4.15E+03	2.00E+03	7.46E+01	3.15E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr157	4.25		f	2.78E+00	4.48E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd158	4.25		f	3.95E+01	3.05E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm158	4.25		f	3.84E+02	3.26E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sm158	4.25		f	1.46E+03	1.45E+03	3.36E+01	7.70E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
eu158	4.25		f	1.62E+03	1.62E+03	1.15E+03	7.37E+02	1.30E+00	6.56E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd159	4.25		f	3.44E+00	1.17E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm159	4.25		f	9.18E+01	7.32E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sm159	4.25		f	6.22E+02	6.20E+02	2.82E-01	1.27E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
eu159	4.25		f	8.36E+02	8.36E+02	3.00E+02	9.50E+01	9.83E-06	1.06E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
gd159	4.25	l	f	4.36E+04	4.36E+04	4.28E+04	4.21E+04	3.24E+04	1.78E+04	1.21E+03	9.18E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm160	4.25		f	1.02E+01	4.01E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sm160	4.25		f	1.88E+02	1.86E+02	6.45E-06	2.19E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
eu160	4.25		f	3.36E+02	3.33E+02	1.64E-05	5.57E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tb160	4.25	l	f	1.64E+04	1.64E+04	1.64E+04	1.64E+04	1.63E+04	1.62E+04	1.58E+04	1.23E+04	6.91E+03	2.92E+03	4.95E+02	4.50E-01
sm161	4.25		f	4.18E+01	3.63E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

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Table 3.3

Activity (Curies) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
eu161	4.25	f	1.31E+02	1.29E+02	1.78E-11	2.29E-24	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
gd161	4.25	l f	5.06E+03	5.04E+03	1.73E+01	5.90E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tb161	4.25	l f	8.08E+03	8.08E+03	8.06E+03	8.05E+03	7.81E+03	7.30E+03	5.41E+03	3.97E+02	9.57E-01	1.13E-04	9.41E-13	0.00E+00
sm162	4.25	f	5.74E+00	5.04E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
eu162	4.25	f	3.19E+01	3.18E+01	1.48E-02	6.81E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
gd162	4.25	l f	7.14E+01	7.13E+01	7.29E+00	6.13E-01	5.45E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tb162	4.25	l f	7.25E+01	7.25E+01	2.27E+01	3.04E+00	6.89E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
eu163	4.25	f	6.44E+00	5.91E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
gd163	4.25	f	2.48E+01	2.47E+01	3.67E-05	5.28E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tb163	4.25	f	2.82E+01	2.82E+01	1.05E+01	3.61E+00	1.18E-06	1.79E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
gd164	4.25	f	7.37E+00	7.36E+00	2.82E+00	1.08E+00	1.60E-06	7.51E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tb164	4.25	f	1.01E+01	1.01E+01	3.28E+00	1.26E+00	1.85E-06	8.72E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tb165	4.25	f	3.40E+00	3.39E+00	2.21E-04	1.16E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
dy165	4.25	l f	3.06E+03	3.06E+03	2.65E+03	2.29E+03	2.86E+02	2.47E+00	1.28E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
dy165m	4.25	l f	1.96E+03	1.94E+03	6.00E-04	2.46E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
dy166	4.25	l	1.45E+01	1.45E+01	1.45E+01	1.44E+01	1.36E+01	1.19E+01	6.43E+00	3.21E-02	1.57E-07	1.69E-15	6.85E-32	0.00E+00
ho166	4.25	l f	7.57E+02	7.57E+02	7.48E+02	7.38E+02	6.18E+02	4.13E+02	7.14E+01	5.01E-02	2.44E-07	2.63E-15	9.13E-32	0.00E+00
er167m	4.25	l	1.81E+01	1.34E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
er169	4.25	l	3.18E-01	3.18E-01	3.18E-01	3.17E-01	3.10E-01	2.96E-01	2.37E-01	3.48E-02	4.17E-04	5.48E-07	6.40E-13	0.00E+00
tm170	4.25	l	5.36E-02	5.36E-02	5.36E-02	5.36E-02	5.35E-02	5.34E-02	5.25E-02	4.56E-02	3.30E-02	2.03E-02	7.49E-03	1.46E-04
hf175	4.5	l	2.86E+00	2.86E+00	2.86E+00	2.86E+00	2.85E+00	2.83E+00	2.75E+00	2.13E+00	1.17E+00	4.82E-01	7.70E-02	5.57E-05
lu176m	4.25	l	2.60E+00	2.60E+00	2.36E+00	2.14E+00	5.65E-01	2.67E-02	2.93E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
lu177	4.25	l	1.12E+00	1.12E+00	1.12E+00	1.11E+00	1.08E+00	1.01E+00	7.40E-01	5.10E-02	6.23E-04	3.53E-04	1.59E-04	6.83E-06
hf178m	4.5	l	3.02E-01	2.54E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
hf179m	4.5	l	1.87E+02	1.81E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ta180	4.25	l	2.30E-02	2.30E-02	2.22E-02	2.13E-02	1.25E-02	3.45E-03	8.16E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
hf180m	4.5	l	1.19E+01	1.19E+01	1.12E+01	1.05E+01	4.35E+00	5.80E-01	6.64E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
hf181	4.25	l	4.48E+02	4.48E+02	4.48E+02	4.47E+02	4.45E+02	4.40E+02	4.19E+02	2.74E+02	1.03E+02	2.36E+01	1.14E+00	7.39E-06
w181	4.25	l	1.87E+00	1.87E+00	1.87E+00	1.87E+00	1.87E+00	1.86E+00	1.83E+00	1.58E+00	1.12E+00	6.68E-01	2.32E-01	3.55E-03
ta182	4.25	l	5.61E+01	5.61E+01	5.61E+01	5.61E+01	5.60E+01	5.58E+01	5.48E+01	4.68E+01	3.26E+01	1.90E+01	6.21E+00	7.60E-02
ta182m	4.25	l	1.06E-01	1.06E-01	2.86E-02	7.70E-03	7.97E-11	4.46E-29	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ta183	4.25	l	1.54E+02	1.54E+02	1.53E+02	1.53E+02	1.47E+02	1.34E+02	8.94E+01	2.62E+00	7.57E-04	3.72E-09	4.42E-20	0.00E+00
w183m	4.25	l	1.69E+02	1.67E+02	1.53E+02	1.53E+02	1.47E+02	1.34E+02	8.94E+01	2.62E+00	7.57E-04	3.72E-09	4.42E-20	0.00E+00
w185	4.25	l	5.96E+01	5.96E+01	5.96E+01	5.96E+01	5.94E+01	5.91E+01	5.74E+01	4.52E+01	2.60E+01	1.13E+01	2.05E+00	2.42E-03
w185m	4.25	l	1.09E-01	1.08E-01	4.26E-07	1.66E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

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Table 3.3

Activity (Curies) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Activity (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
re186	4.25	l	5.05E+01	5.05E+01	5.03E+01	5.01E+01	4.75E+01	4.21E+01	2.42E+01	2.05E-01	3.39E-06	2.27E-13	3.89E-28	0.00E+00
w187	4.25	l	5.17E+02	5.17E+02	5.10E+02	5.02E+02	4.10E+02	2.58E+02	3.19E+01	4.41E-07	3.22E-25	0.00E+00	0.00E+00	0.00E+00
w188	4.25	l	2.72E+00	2.72E+00	2.71E+00	2.71E+00	2.71E+00	2.69E+00	2.61E+00	2.01E+00	1.11E+00	4.50E-01	7.09E-02	4.83E-05
re188	4.25	l	3.39E+02	3.39E+02	3.36E+02	3.31E+02	2.50E+02	1.31E+02	9.44E+00	2.03E+00	1.12E+00	4.55E-01	7.16E-02	4.88E-05
re188m	4.25	l	3.29E+02	3.29E+02	1.08E+02	3.52E+01	5.61E-06	1.63E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
os191	4.25	l	6.33E-01	6.33E-01	6.33E-01	6.33E-01	6.29E-01	6.17E-01	5.43E-01	1.69E-01	1.13E-02	1.98E-04	4.74E-08	2.52E-22
os191m	4.25	l	4.70E-01	4.70E-01	4.58E-01	4.46E-01	3.08E-01	1.32E-01	2.93E-03	1.34E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ir192	4.25	l	3.14E-01	3.14E-01	3.14E-01	3.14E-01	3.13E-01	3.12E-01	3.03E-01	2.37E-01	1.35E-01	5.80E-02	1.02E-02	1.12E-05
ir194	4.25	l	2.47E-02	2.47E-02	2.43E-02	2.39E-02	1.85E-02	1.04E-02	7.66E-04	1.85E-08	1.82E-08	1.77E-08	1.67E-08	1.32E-08
u237	4.5	a	1.67E+05	1.67E+05	1.66E+05	1.66E+05	1.61E+05	1.50E+05	1.11E+05	7.65E+03	1.69E+01	7.36E-01	7.16E-01	6.50E-01
u239	4.25	a	3.27E+06	3.27E+06	1.35E+06	5.56E+05	2.28E+00	1.10E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
np236m	4.5	a	1.15E+00	1.15E+00	1.14E+00	1.12E+00	9.01E-01	5.51E-01	5.99E-02	2.69E-10	1.46E-29	0.00E+00	0.00E+00	0.00E+00
pu237	4.25	a	2.52E+00	2.52E+00	2.52E+00	2.52E+00	2.51E+00	2.48E+00	2.37E+00	1.59E+00	6.33E-01	1.59E-01	9.28E-03	1.26E-07
np238	4.25	a	1.11E+05	1.11E+05	1.11E+05	1.10E+05	9.99E+04	8.03E+04	3.01E+04	6.05E+00	1.35E-02	1.35E-02	1.35E-02	1.33E-02
pu238	4.5	a	1.38E+03	1.38E+03	1.38E+03	1.38E+03	1.38E+03	1.38E+03	1.39E+03	1.40E+03	1.41E+03	1.43E+03	1.45E+03	1.44E+03
np239	4.25	a	3.26E+06	3.26E+06	3.26E+06	3.24E+06	2.98E+06	2.45E+06	1.01E+06	4.94E+02	1.23E+01	1.23E+01	1.23E+01	1.23E+01
pu239	4.5	a	6.13E+01	6.13E+01	6.13E+01	6.13E+01	6.14E+01	6.16E+01	6.19E+01	6.22E+01	6.22E+01	6.22E+01	6.22E+01	6.22E+01
np240	4.25	a	7.18E+03	7.18E+03	5.13E+03	3.67E+03	3.33E+01	7.13E-04	4.03E-17	4.55E-17	5.72E-17	7.48E-17	1.11E-16	2.54E-16
pu240	4.25	a	1.32E+02	1.32E+02	1.32E+02	1.32E+02	1.32E+02	1.32E+02	1.32E+02	1.32E+02	1.32E+02	1.32E+02	1.32E+02	1.33E+02
pu241	4.5	a	3.14E+04	3.14E+04	3.14E+04	3.14E+04	3.14E+04	3.14E+04	3.14E+04	3.13E+04	3.10E+04	3.07E+04	2.99E+04	2.72E+04
pu242	4.25	a	8.48E-01	8.48E-01	8.48E-01	8.48E-01	8.48E-01	8.48E-01	8.48E-01	8.48E-01	8.48E-01	8.48E-01	8.48E-01	8.48E-01
am241	4.5	a	4.63E+01	4.63E+01	4.63E+01	4.63E+01	4.64E+01	4.65E+01	4.69E+01	5.05E+01	5.86E+01	7.08E+01	9.54E+01	1.86E+02
am242m	4.5	a	3.01E+00	3.01E+00	3.01E+00	3.01E+00	3.01E+00	3.01E+00	3.01E+00	3.00E+00	3.00E+00	3.00E+00	2.99E+00	2.96E+00
am242	4.25	a	2.46E+04	2.46E+04	2.40E+04	2.35E+04	1.74E+04	8.70E+03	3.88E+02	2.99E+00	2.99E+00	2.98E+00	2.98E+00	2.95E+00
cm242	4.25	a	1.71E+04	1.71E+04	1.71E+04	1.71E+04	1.71E+04	1.71E+04	1.69E+04	1.52E+04	1.17E+04	8.01E+03	3.64E+03	1.65E+02
pu243	4.25	a	1.22E+05	1.22E+05	1.14E+05	1.06E+05	4.00E+04	4.26E+03	1.80E-01	9.63E-07	9.63E-07	9.63E-07	9.63E-07	9.63E-07
am243	4.25	a	1.23E+01	1.23E+01	1.23E+01	1.23E+01	1.23E+01	1.23E+01	1.23E+01	1.23E+01	1.23E+01	1.23E+01	1.23E+01	1.23E+01
cm243	4.25	a	1.05E+01	1.05E+01	1.05E+01	1.05E+01	1.05E+01	1.05E+01	1.05E+01	1.05E+01	1.05E+01	1.04E+01	1.03E+01	9.80E+00
am244	4.25	a	6.13E+04	6.13E+04	5.92E+04	5.72E+04	3.54E+04	1.18E+04	8.44E+01	2.13E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cm244	4.25	a	2.62E+03	2.62E+03	2.62E+03	2.62E+03	2.62E+03	2.62E+03	2.62E+03	2.62E+03	2.60E+03	2.58E+03	2.53E+03	2.34E+03

Table 3.4

Mass (grams) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Isotopic Mass (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
h 1	4.5	l	4.88E+02	4.88E+02	4.88E+02	4.88E+02	4.88E+02	4.88E+02	4.88E+02	4.88E+02	4.88E+02	4.88E+02	4.88E+02	4.88E+02
h 2	4.25	l	1.12E+00	1.12E+00	1.12E+00	1.12E+00	1.12E+00	1.12E+00	1.12E+00	1.12E+00	1.12E+00	1.12E+00	1.12E+00	1.12E+00
h 3	4.5	f	1.48E-02	1.48E-02	1.48E-02	1.48E-02	1.48E-02	1.48E-02	1.48E-02	1.47E-02	1.46E-02	1.44E-02	1.40E-02	1.25E-02
he 4	4.25	a l	3.16E-01	3.16E-01	3.16E-01	3.16E-01	3.16E-01	3.17E-01	3.18E-01	3.29E-01	3.52E-01	3.78E-01	4.17E-01	5.00E-01
c 13	4.25	l	1.11E-02	1.11E-02	1.11E-02	1.11E-02	1.11E-02	1.11E-02	1.11E-02	1.11E-02	1.11E-02	1.11E-02	1.11E-02	1.11E-02
o 16	4.5	l	7.98E+01	7.98E+01	7.98E+01	7.98E+01	7.98E+01	7.98E+01	7.98E+01	7.98E+01	7.98E+01	7.98E+01	7.98E+01	7.98E+01
o 17	4.5	l	3.23E-02	3.23E-02	3.23E-02	3.23E-02	3.23E-02	3.23E-02	3.23E-02	3.23E-02	3.23E-02	3.23E-02	3.23E-02	3.23E-02
o 18	4.5	l	1.84E-01	1.84E-01	1.84E-01	1.84E-01	1.84E-01	1.84E-01	1.84E-01	1.84E-01	1.84E-01	1.84E-01	1.84E-01	1.84E-01
mg 24	4.5	l	1.56E+00	1.56E+00	1.56E+00	1.56E+00	1.56E+00	1.56E+00	1.56E+00	1.56E+00	1.56E+00	1.56E+00	1.56E+00	1.56E+00
mg 25	4.5	l	2.06E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01
mg 26	4.5	l	2.36E-01	2.36E-01	2.36E-01	2.36E-01	2.36E-01	2.36E-01	2.36E-01	2.36E-01	2.36E-01	2.36E-01	2.36E-01	2.36E-01
al 27	4.5	l	5.99E+00	5.99E+00	5.99E+00	5.99E+00	5.99E+00	5.99E+00	5.99E+00	5.99E+00	5.99E+00	5.99E+00	5.99E+00	5.99E+00
si 28	4.5	l	8.73E+00	8.73E+00	8.73E+00	8.73E+00	8.73E+00	8.73E+00	8.73E+00	8.73E+00	8.73E+00	8.73E+00	8.73E+00	8.73E+00
si 29	4.25	l	4.63E-01	4.63E-01	4.63E-01	4.63E-01	4.63E-01	4.63E-01	4.63E-01	4.63E-01	4.63E-01	4.63E-01	4.63E-01	4.63E-01
si 30	4.5	l	3.14E-01	3.14E-01	3.14E-01	3.14E-01	3.14E-01	3.14E-01	3.14E-01	3.14E-01	3.14E-01	3.14E-01	3.14E-01	3.14E-01
v 50	4.25	l	2.33E-02	2.33E-02	2.33E-02	2.33E-02	2.33E-02	2.33E-02	2.33E-02	2.33E-02	2.33E-02	2.33E-02	2.33E-02	2.33E-02
cr 50	4.5	l	6.22E+01	6.22E+01	6.22E+01	6.22E+01	6.22E+01	6.22E+01	6.22E+01	6.22E+01	6.22E+01	6.22E+01	6.22E+01	6.22E+01
v 51	4.25	l	3.55E+00	3.55E+00	3.55E+00	3.55E+00	3.55E+00	3.56E+00	3.56E+00	3.61E+00	3.64E+00	3.65E+00	3.65E+00	3.65E+00
cr 51	4.25	l	9.95E-02	9.95E-02	9.94E-02	9.94E-02	9.87E-02	9.70E-02	9.00E-02	4.70E-02	1.05E-02	1.10E-03	1.07E-05	1.24E-13
cr 52	4.5	l	1.31E+03	1.31E+03	1.31E+03	1.31E+03	1.31E+03	1.31E+03	1.31E+03	1.31E+03	1.31E+03	1.31E+03	1.31E+03	1.31E+03
cr 53	4.5	l	1.47E+02	1.47E+02	1.47E+02	1.47E+02	1.47E+02	1.47E+02	1.47E+02	1.47E+02	1.47E+02	1.47E+02	1.47E+02	1.47E+02
cr 54	4.25	l	4.86E+01	4.86E+01	4.86E+01	4.86E+01	4.86E+01	4.86E+01	4.86E+01	4.86E+01	4.86E+01	4.86E+01	4.86E+01	4.86E+01
mn 54	4.25	l	8.58E-02	8.58E-02	8.58E-02	8.58E-02	8.57E-02	8.56E-02	8.51E-02	8.03E-02	7.03E-02	5.75E-02	3.81E-02	7.53E-03
fe 54	4.5	l	3.12E+02	3.12E+02	3.12E+02	3.12E+02	3.12E+02	3.12E+02	3.12E+02	3.12E+02	3.12E+02	3.12E+02	3.12E+02	3.12E+02
mn 55	4.5	l	1.23E+02	1.23E+02	1.23E+02	1.23E+02	1.23E+02	1.23E+02	1.23E+02	1.23E+02	1.23E+02	1.23E+02	1.23E+02	1.24E+02
fe 55	4.25	l	1.53E+00	1.53E+00	1.53E+00	1.53E+00	1.53E+00	1.53E+00	1.53E+00	1.50E+00	1.44E+00	1.35E+00	1.19E+00	7.16E-01
fe 56	4.5	l	5.11E+03	5.11E+03	5.11E+03	5.11E+03	5.11E+03	5.11E+03	5.11E+03	5.11E+03	5.11E+03	5.11E+03	5.11E+03	5.11E+03
fe 57	4.25	l	1.72E+02	1.72E+02	1.72E+02	1.72E+02	1.72E+02	1.72E+02	1.72E+02	1.72E+02	1.72E+02	1.72E+02	1.72E+02	1.72E+02
fe 58	4.25	l	1.86E+01	1.86E+01	1.86E+01	1.86E+01	1.86E+01	1.86E+01	1.86E+01	1.86E+01	1.86E+01	1.86E+01	1.86E+01	1.86E+01
co 58	4.5	l	2.56E-02	2.56E-02	2.56E-02	2.56E-02	2.56E-02	2.54E-02	2.47E-02	1.91E-02	1.06E-02	4.41E-03	7.22E-04	5.72E-07
ni 58	4.5	l	4.95E+02	4.95E+02	4.95E+02	4.95E+02	4.95E+02	4.95E+02	4.95E+02	4.95E+02	4.95E+02	4.95E+02	4.95E+02	4.95E+02
co 59	4.5	l	3.03E+00	3.03E+00	3.03E+00	3.03E+00	3.03E+00	3.03E+00	3.03E+00	3.04E+00	3.04E+00	3.04E+00	3.04E+00	3.04E+00
ni 59	4.25	l	6.39E+00	6.39E+00	6.39E+00	6.39E+00	6.39E+00	6.39E+00	6.39E+00	6.39E+00	6.39E+00	6.39E+00	6.39E+00	6.39E+00
co 60	4.25	l	5.55E-01	5.55E-01	5.55E-01	5.55E-01	5.55E-01	5.54E-01	5.54E-01	5.49E-01	5.37E-01	5.20E-01	4.86E-01	3.74E-01
ni 60	4.5	l	1.99E+02	1.99E+02	1.99E+02	1.99E+02	1.99E+02	1.99E+02	1.99E+02	1.99E+02	1.99E+02	1.99E+02	1.99E+02	2.00E+02

Mass (grams) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Isotopic Mass (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
ni 61	4.25	l	1.08E+01	1.08E+01	1.08E+01	1.08E+01	1.08E+01	1.08E+01	1.08E+01	1.08E+01	1.08E+01	1.08E+01	1.08E+01	1.08E+01
ni 62	4.5	l	2.71E+01	2.71E+01	2.71E+01	2.71E+01	2.71E+01	2.71E+01	2.71E+01	2.71E+01	2.71E+01	2.71E+01	2.71E+01	2.71E+01
ni 63	4.25	l	1.35E+00	1.35E+00	1.35E+00	1.35E+00	1.35E+00	1.35E+00	1.35E+00	1.35E+00	1.35E+00	1.35E+00	1.34E+00	1.32E+00
cu 63	4.25	l	2.29E-02	2.29E-02	2.29E-02	2.29E-02	2.29E-02	2.30E-02	2.30E-02	2.37E-02	2.52E-02	2.75E-02	3.23E-02	5.07E-02
ni 64	4.5	l	7.41E+00	7.41E+00	7.41E+00	7.41E+00	7.41E+00	7.41E+00	7.41E+00	7.41E+00	7.41E+00	7.41E+00	7.41E+00	7.41E+00
cu 65	4.25	l	4.57E-02	4.57E-02	4.57E-02	4.57E-02	4.57E-02	4.57E-02	4.57E-02	4.57E-02	4.57E-02	4.57E-02	4.57E-02	4.57E-02
ge 72	4.25	f	1.61E-03	1.61E-03	1.61E-03	1.61E-03	1.61E-03	1.61E-03	1.62E-03	1.62E-03	1.62E-03	1.62E-03	1.62E-03	1.62E-03
ge 73	4.25	f	4.51E-03	4.51E-03	4.51E-03	4.51E-03	4.51E-03	4.51E-03	4.51E-03	4.51E-03	4.51E-03	4.51E-03	4.51E-03	4.51E-03
ge 74	4.25	f	4.15E-03	4.15E-03	4.15E-03	4.15E-03	4.15E-03	4.15E-03	4.15E-03	4.15E-03	4.15E-03	4.15E-03	4.15E-03	4.15E-03
as 75	4.5	f	3.46E-02	3.46E-02	3.46E-02	3.46E-02	3.46E-02	3.46E-02	3.46E-02	3.46E-02	3.46E-02	3.46E-02	3.46E-02	3.46E-02
ge 76	4.5	f	1.07E-01	1.07E-01	1.07E-01	1.07E-01	1.07E-01	1.07E-01	1.07E-01	1.07E-01	1.07E-01	1.07E-01	1.07E-01	1.07E-01
se 76	4.25	f	1.81E-03	1.81E-03	1.81E-03	1.81E-03	1.81E-03	1.81E-03	1.81E-03	1.81E-03	1.81E-03	1.81E-03	1.81E-03	1.81E-03
se 77	4.5	f	2.33E-01	2.33E-01	2.33E-01	2.33E-01	2.33E-01	2.33E-01	2.34E-01	2.34E-01	2.34E-01	2.34E-01	2.34E-01	2.34E-01
se 78	4.5	f	8.13E-01	8.13E-01	8.13E-01	8.13E-01	8.13E-01	8.13E-01	8.13E-01	8.13E-01	8.13E-01	8.13E-01	8.13E-01	8.13E-01
se 79	4.5	f	1.51E+00	1.51E+00	1.51E+00	1.51E+00	1.51E+00	1.51E+00	1.51E+00	1.51E+00	1.51E+00	1.51E+00	1.51E+00	1.51E+00
se 80	4.5	f	4.16E+00	4.16E+00	4.16E+00	4.16E+00	4.16E+00	4.16E+00	4.16E+00	4.16E+00	4.16E+00	4.16E+00	4.16E+00	4.16E+00
br 81	4.5	f	6.06E+00	6.06E+00	6.06E+00	6.06E+00	6.06E+00	6.06E+00	6.06E+00	6.06E+00	6.06E+00	6.06E+00	6.06E+00	6.06E+00
se 82	4.5	f	9.96E+00	9.96E+00	9.96E+00	9.96E+00	9.96E+00	9.96E+00	9.96E+00	9.96E+00	9.96E+00	9.96E+00	9.96E+00	9.96E+00
kr 82	4.25	f	2.96E-01	2.96E-01	2.96E-01	2.96E-01	2.96E-01	2.96E-01	2.96E-01	2.96E-01	2.96E-01	2.96E-01	2.96E-01	2.96E-01
w183	4.25	l	1.72E+00	1.72E+00	1.72E+00	1.72E+00	1.72E+00	1.72E+00	1.72E+00	1.72E+00	1.72E+00	1.72E+00	1.72E+00	1.72E+00
kr 83	4.5	f	1.09E+01	1.09E+01	1.09E+01	1.09E+01	1.09E+01	1.09E+01	1.09E+01	1.09E+01	1.09E+01	1.09E+01	1.09E+01	1.09E+01
w184	4.25	l	3.07E+00	3.07E+00	3.07E+00	3.07E+00	3.07E+00	3.07E+00	3.07E+00	3.07E+00	3.07E+00	3.07E+00	3.07E+00	3.07E+00
kr 84	4.5	f	3.69E+01	3.69E+01	3.69E+01	3.69E+01	3.69E+01	3.69E+01	3.69E+01	3.69E+01	3.69E+01	3.69E+01	3.69E+01	3.69E+01
kr 85	4.5	f	6.35E+00	6.35E+00	6.35E+00	6.35E+00	6.35E+00	6.34E+00	6.34E+00	6.31E+00	6.25E+00	6.15E+00	5.95E+00	5.23E+00
kr 85m	4.5	f	3.49E-03	3.49E-03	3.27E-03	3.03E-03	1.03E-03	8.63E-05	1.25E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 85	4.5	f	2.88E+01	2.88E+01	2.88E+01	2.88E+01	2.88E+01	2.88E+01	2.88E+01	2.89E+01	2.89E+01	2.90E+01	2.92E+01	2.99E+01
w186	4.5	l	1.11E+00	1.11E+00	1.11E+00	1.11E+00	1.11E+00	1.11E+00	1.11E+00	1.11E+00	1.11E+00	1.11E+00	1.11E+00	1.11E+00
kr 86	4.5	f	5.63E+01	5.63E+01	5.63E+01	5.63E+01	5.63E+01	5.63E+01	5.63E+01	5.63E+01	5.63E+01	5.63E+01	5.63E+01	5.63E+01
rb 86	4.25	f	5.69E-03	5.69E-03	5.68E-03	5.68E-03	5.62E-03	5.48E-03	4.90E-03	1.86E-03	2.00E-04	7.03E-06	7.15E-09	1.13E-20
sr 86	4.25	f	1.90E-01	1.90E-01	1.90E-01	1.90E-01	1.90E-01	1.90E-01	1.91E-01	1.94E-01	1.96E-01	1.96E-01	1.96E-01	1.96E-01
kr 87	4.5	f	2.01E-03	2.01E-03	1.55E-03	1.18E-03	2.60E-05	4.24E-09	3.87E-26	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb 87	4.5	f	7.32E+01	7.32E+01	7.32E+01	7.32E+01	7.32E+01	7.32E+01	7.32E+01	7.32E+01	7.32E+01	7.32E+01	7.32E+01	7.32E+01
sr 87	4.25	f	1.27E-03	1.27E-03	1.27E-03	1.27E-03	1.27E-03	1.27E-03	1.27E-03	1.27E-03	1.27E-03	1.27E-03	1.27E-03	1.27E-03
kr 88	4.5	f	6.21E-03	6.21E-03	5.50E-03	4.87E-03	8.82E-04	1.77E-05	4.11E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr 88	4.5	l f	1.05E+02	1.05E+02	1.05E+02	1.05E+02	1.05E+02	1.05E+02	1.05E+02	1.05E+02	1.05E+02	1.05E+02	1.05E+02	1.05E+02

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Table 3.4

Mass (grams) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Isotopic Mass (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
sr 89	4.5	f	3.67E+00	3.67E+00	3.67E+00	3.67E+00	3.66E+00	3.62E+00	3.48E+00	2.44E+00	1.07E+00	3.11E-01	2.46E-02	1.10E-06
y 89	4.5	l f	1.36E+02	1.36E+02	1.36E+02	1.36E+02	1.36E+02	1.36E+02	1.36E+02	1.37E+02	1.39E+02	1.39E+02	1.40E+02	1.40E+02
sr 90	4.5	f	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.58E+02	1.58E+02	1.57E+02	1.55E+02	1.47E+02
y 90	4.5	f	4.29E-02	4.29E-02	4.29E-02	4.29E-02	4.27E-02	4.25E-02	4.18E-02	4.11E-02	4.10E-02	4.07E-02	4.02E-02	3.83E-02
zr 90	4.5	l f	3.88E+04	3.88E+04	3.88E+04	3.88E+04	3.88E+04	3.88E+04	3.88E+04	3.88E+04	3.88E+04	3.88E+04	3.88E+04	3.88E+04
sr 91	4.5	f	3.88E-02	3.88E-02	3.75E-02	3.62E-02	2.17E-02	6.77E-03	3.58E-05	6.63E-25	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 91	4.5	f	5.88E+00	5.88E+00	5.88E+00	5.88E+00	5.88E+00	5.85E+00	5.65E+00	4.15E+00	2.04E+00	7.02E-01	7.82E-02	1.36E-05
y 91m	4.5	f	1.96E-03	1.96E-03	1.95E-03	1.92E-03	1.20E-03	3.75E-04	1.98E-06	3.67E-26	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr 91	4.5	l f	8.64E+03	8.64E+03	8.64E+03	8.64E+03	8.64E+03	8.64E+03	8.64E+03	8.64E+03	8.64E+03	8.64E+03	8.64E+03	8.64E+03
mo 92	4.5	l	5.66E-01	5.66E-01	5.66E-01	5.66E-01	5.66E-01	5.66E-01	5.66E-01	5.66E-01	5.66E-01	5.66E-01	5.66E-01	5.66E-01
sr 92	4.5	f	1.23E-02	1.23E-02	1.08E-02	9.54E-03	1.59E-03	2.66E-05	2.67E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y 92	4.5	f	1.62E-02	1.62E-02	1.61E-02	1.58E-02	7.56E-03	5.11E-04	4.68E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr 92	4.5	l f	1.34E+04	1.34E+04	1.34E+04	1.34E+04	1.34E+04	1.34E+04	1.34E+04	1.34E+04	1.34E+04	1.34E+04	1.34E+04	1.34E+04
y 93	4.5	f	3.69E-02	3.69E-02	3.60E-02	3.48E-02	2.15E-02	7.19E-03	5.13E-05	1.29E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr 93	4.5	l f	1.68E+02	1.68E+02	1.68E+02	1.68E+02	1.68E+02	1.68E+02	1.68E+02	1.68E+02	1.68E+02	1.68E+02	1.68E+02	1.68E+02
mo 94	4.5	l	3.63E-01	3.63E-01	3.63E-01	3.63E-01	3.63E-01	3.63E-01	3.63E-01	3.63E-01	3.63E-01	3.63E-01	3.63E-01	3.63E-01
y 94	4.5	f	1.87E-03	1.87E-03	6.56E-04	2.16E-04	3.74E-11	1.31E-26	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr 94	4.5	l f	1.39E+04	1.39E+04	1.39E+04	1.39E+04	1.39E+04	1.39E+04	1.39E+04	1.39E+04	1.39E+04	1.39E+04	1.39E+04	1.39E+04
y 95	4.5	f	1.12E-03	1.12E-03	1.60E-04	2.21E-05	2.01E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr 95	4.5	l f	1.09E+01	1.09E+01	1.09E+01	1.09E+01	1.08E+01	1.08E+01	1.04E+01	7.84E+00	4.09E+00	1.54E+00	2.08E-01	7.63E-05
nb 95	4.5	l f	5.94E+00	5.94E+00	5.94E+00	5.94E+00	5.94E+00	5.94E+00	5.93E+00	5.51E+00	3.73E+00	1.66E+00	2.45E-01	9.18E-05
nb 95m	4.5	f	6.39E-03	6.39E-03	6.39E-03	6.39E-03	6.39E-03	6.38E-03	6.30E-03	4.88E-03	2.55E-03	9.62E-04	1.29E-04	4.75E-08
mo 95	4.5	l f	2.22E+02	2.22E+02	2.22E+02	2.22E+02	2.22E+02	2.22E+02	2.22E+02	2.26E+02	2.31E+02	2.36E+02	2.38E+02	2.39E+02
zr 96	4.5	l f	2.48E+03	2.48E+03	2.48E+03	2.48E+03	2.48E+03	2.48E+03	2.48E+03	2.48E+03	2.48E+03	2.48E+03	2.48E+03	2.48E+03
mo 96	4.25	l f	2.16E+01	2.16E+01	2.16E+01	2.16E+01	2.16E+01	2.16E+01	2.16E+01	2.16E+01	2.16E+01	2.16E+01	2.16E+01	2.16E+01
zr 97	4.5	l f	1.27E-01	1.27E-01	1.24E-01	1.22E-01	9.15E-02	4.75E-02	2.48E-03	1.91E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb 97	4.5	f	8.23E-03	8.23E-03	8.18E-03	8.10E-03	6.30E-03	3.06E-03	1.60E-04	1.32E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo 97	4.5	l f	2.57E+02	2.57E+02	2.57E+02	2.57E+02	2.57E+02	2.57E+02	2.57E+02	2.57E+02	2.57E+02	2.57E+02	2.57E+02	2.57E+02
mo 98	4.25	l f	2.61E+02	2.61E+02	2.61E+02	2.61E+02	2.61E+02	2.61E+02	2.61E+02	2.61E+02	2.61E+02	2.61E+02	2.61E+02	2.61E+02
tc 98	4.25	f	3.26E-03	3.26E-03	3.26E-03	3.26E-03	3.26E-03	3.26E-03	3.26E-03	3.26E-03	3.26E-03	3.26E-03	3.26E-03	3.26E-03
mo 99	4.5	f	5.42E-01	5.42E-01	5.40E-01	5.37E-01	4.99E-01	4.22E-01	1.98E-01	2.80E-04	7.49E-11	1.03E-20	5.23E-41	0.00E+00
tc 99	4.5	f	2.32E+02	2.32E+02	2.32E+02	2.32E+02	2.32E+02	2.32E+02	2.33E+02	2.33E+02	2.33E+02	2.33E+02	2.33E+02	2.33E+02
tc 99m	4.5	f	4.40E-02	4.40E-02	4.40E-02	4.39E-02	4.25E-02	3.70E-02	1.75E-02	2.47E-05	6.61E-12	9.12E-22	0.00E+00	0.00E+00
ru 99	4.5	f	1.04E-02	1.04E-02	1.04E-02	1.04E-02	1.04E-02	1.04E-02	1.04E-02	1.05E-02	1.06E-02	1.08E-02	1.12E-02	1.27E-02
mo100	4.5	l f	2.94E+02	2.94E+02	2.94E+02	2.94E+02	2.94E+02	2.94E+02	2.94E+02	2.94E+02	2.94E+02	2.94E+02	2.94E+02	2.94E+02

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Table 3.4

Mass (grams) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Isotopic Mass (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
ru100	4.25	f	5.14E+01	5.14E+01	5.14E+01	5.14E+01	5.14E+01	5.14E+01	5.14E+01	5.14E+01	5.14E+01	5.14E+01	5.14E+01	5.14E+01
mo101	4.5	f	1.90E-03	1.90E-03	4.62E-04	1.11E-04	2.43E-13	3.91E-33	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc101	4.5	f	1.85E-03	1.85E-03	1.07E-03	4.03E-04	4.22E-12	1.19E-31	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru101	4.5	f	2.36E+02	2.36E+02	2.36E+02	2.36E+02	2.36E+02	2.36E+02	2.36E+02	2.36E+02	2.36E+02	2.36E+02	2.36E+02	2.36E+02
mo102	4.25	f	1.46E-03	1.46E-03	2.32E-04	3.68E-05	2.39E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru102	4.25	f	2.56E+02	2.56E+02	2.56E+02	2.56E+02	2.56E+02	2.56E+02	2.56E+02	2.56E+02	2.56E+02	2.56E+02	2.56E+02	2.56E+02
ru103	4.25	f	7.68E+00	7.68E+00	7.68E+00	7.68E+00	7.64E+00	7.55E+00	7.16E+00	4.52E+00	1.57E+00	3.20E-01	1.22E-02	3.04E-08
rh103	4.5	f	1.16E+02	1.16E+02	1.16E+02	1.16E+02	1.16E+02	1.16E+02	1.16E+02	1.19E+02	1.22E+02	1.23E+02	1.23E+02	1.23E+02
rh103m	4.25	f	7.62E-03	7.62E-03	7.61E-03	7.61E-03	7.57E-03	7.48E-03	7.09E-03	4.48E-03	1.55E-03	3.17E-04	1.20E-05	3.02E-11
tc104	4.25	f	2.17E-03	2.17E-03	7.36E-04	2.36E-04	2.92E-11	4.71E-27	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru104	4.25	f	1.81E+02	1.81E+02	1.81E+02	1.81E+02	1.81E+02	1.81E+02	1.81E+02	1.81E+02	1.81E+02	1.81E+02	1.81E+02	1.81E+02
pd104	4.25	f	1.04E+02	1.04E+02	1.04E+02	1.04E+02	1.04E+02	1.04E+02	1.04E+02	1.04E+02	1.04E+02	1.04E+02	1.04E+02	1.04E+02
ru105	4.25	f	2.81E-02	2.81E-02	2.67E-02	2.47E-02	8.30E-03	6.82E-04	8.93E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh105	4.25	f	2.06E-01	2.06E-01	2.06E-01	2.06E-01	1.95E-01	1.48E-01	3.64E-02	1.77E-07	9.77E-20	4.00E-38	0.00E+00	0.00E+00
pd105	4.25	f	1.30E+02	1.30E+02	1.30E+02	1.30E+02	1.30E+02	1.30E+02	1.30E+02	1.30E+02	1.30E+02	1.30E+02	1.30E+02	1.30E+02
ru106	4.25	f	3.79E+01	3.79E+01	3.79E+01	3.79E+01	3.79E+01	3.79E+01	3.76E+01	3.59E+01	3.21E+01	2.71E+01	1.92E+01	4.91E+00
pd106	4.25	f	9.28E+01	9.28E+01	9.28E+01	9.28E+01	9.28E+01	9.29E+01	9.31E+01	9.48E+01	9.86E+01	1.04E+02	1.12E+02	1.26E+02
rh107	4.25	f	1.46E-03	1.46E-03	6.87E-04	2.64E-04	3.94E-10	1.89E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pd107	4.25	f	7.88E+01	7.88E+01	7.88E+01	7.88E+01	7.88E+01	7.88E+01	7.88E+01	7.88E+01	7.88E+01	7.88E+01	7.88E+01	7.88E+01
pd108	4.25	f	5.22E+01	5.22E+01	5.22E+01	5.22E+01	5.22E+01	5.22E+01	5.22E+01	5.22E+01	5.22E+01	5.22E+01	5.22E+01	5.22E+01
pd109	4.25	f	3.69E-02	3.69E-02	3.60E-02	3.51E-02	2.47E-02	1.10E-02	2.87E-04	5.58E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ag109	4.25	f	2.78E+01	2.78E+01	2.78E+01	2.78E+01	2.79E+01	2.79E+01	2.79E+01	2.79E+01	2.79E+01	2.79E+01	2.79E+01	2.79E+01
pd110	4.25	f	1.59E+01	1.59E+01	1.59E+01	1.59E+01	1.59E+01	1.59E+01	1.59E+01	1.59E+01	1.59E+01	1.59E+01	1.59E+01	1.59E+01
ag110m	4.25	f	2.70E-01	2.70E-01	2.70E-01	2.70E-01	2.70E-01	2.69E-01	2.67E-01	2.48E-01	2.10E-01	1.64E-01	9.79E-02	1.29E-02
cd110	4.25	f	2.05E+01	2.05E+01	2.05E+01	2.05E+01	2.05E+01	2.05E+01	2.05E+01	2.06E+01	2.06E+01	2.06E+01	2.07E+01	2.08E+01
ag111	4.25	f	7.33E-02	7.33E-02	7.33E-02	7.32E-02	7.13E-02	6.70E-02	5.07E-02	4.51E-03	1.70E-05	3.92E-09	1.28E-16	0.00E+00
cd111	4.25	f	8.37E+00	8.37E+00	8.37E+00	8.37E+00	8.37E+00	8.37E+00	8.39E+00	8.44E+00	8.44E+00	8.44E+00	8.44E+00	8.44E+00
sn112	4.5	l	1.03E+01	1.03E+01	1.03E+01	1.03E+01	1.03E+01	1.03E+01	1.03E+01	1.03E+01	1.03E+01	1.03E+01	1.03E+01	1.03E+01
pd112	4.25	f	3.76E-03	3.76E-03	3.70E-03	3.64E-03	2.89E-03	1.71E-03	1.59E-04	1.89E-13	4.76E-34	0.00E+00	0.00E+00	0.00E+00
cd112	4.25	f	4.32E+00	4.32E+00	4.32E+00	4.32E+00	4.32E+00	4.32E+00	4.32E+00	4.32E+00	4.32E+00	4.32E+00	4.32E+00	4.32E+00
sn113	4.25	l	5.24E-02	5.24E-02	5.24E-02	5.24E-02	5.23E-02	5.21E-02	5.12E-02	4.38E-02	3.05E-02	1.77E-02	5.81E-03	7.14E-05
cd113	4.5	f	1.84E-02	1.84E-02	1.85E-02	1.85E-02	1.88E-02	1.89E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02
cd113m	4.25	f	5.00E-02	5.00E-02	5.00E-02	5.00E-02	5.00E-02	5.00E-02	5.00E-02	4.98E-02	4.94E-02	4.88E-02	4.76E-02	4.32E-02
in113	4.25	l f	4.41E-01	4.41E-01	4.41E-01	4.41E-01	4.41E-01	4.41E-01	4.42E-01	4.50E-01	4.63E-01	4.77E-01	4.90E-01	4.99E-01
sn114	4.5	l	7.36E+00	7.36E+00	7.36E+00	7.36E+00	7.36E+00	7.36E+00	7.36E+00	7.36E+00	7.37E+00	7.37E+00	7.37E+00	7.37E+00

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Table 3.4

Mass (grams) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Isotopic Mass (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
cd114	4.25	f	4.15E+00	4.15E+00	4.15E+00	4.15E+00	4.15E+00	4.15E+00	4.15E+00	4.15E+00	4.15E+00	4.15E+00	4.15E+00	4.15E+00
cd115	4.25	f	2.80E-03	2.80E-03	2.79E-03	2.77E-03	2.53E-03	2.06E-03	8.09E-04	2.48E-07	1.94E-15	1.34E-27	0.00E+00	0.00E+00
cd115m	4.25	f	2.76E-03	2.76E-03	2.76E-03	2.76E-03	2.75E-03	2.72E-03	2.60E-03	1.73E-03	6.82E-04	1.68E-04	9.46E-06	1.11E-10
in115	4.5	f	3.41E-01	3.41E-01	3.41E-01	3.41E-01	3.41E-01	3.41E-01	3.43E-01	3.44E-01	3.45E-01	3.46E-01	3.46E-01	3.46E-01
sn115	4.5	l f	3.68E+00	3.68E+00	3.68E+00	3.68E+00	3.68E+00	3.68E+00	3.68E+00	3.68E+00	3.68E+00	3.68E+00	3.68E+00	3.68E+00
cd116	4.25	f	1.62E+00	1.62E+00	1.62E+00	1.62E+00	1.62E+00	1.62E+00	1.62E+00	1.62E+00	1.62E+00	1.62E+00	1.62E+00	1.62E+00
sn116	4.25	l f	1.64E+02	1.64E+02	1.64E+02	1.64E+02	1.64E+02	1.64E+02	1.64E+02	1.64E+02	1.64E+02	1.64E+02	1.64E+02	1.64E+02
sn117m	4.25	l	4.59E-02	4.59E-02	4.59E-02	4.58E-02	4.52E-02	4.37E-02	3.75E-02	9.96E-03	4.68E-04	4.76E-06	3.78E-10	2.55E-26
sn117	4.25	l f	8.98E+01	8.98E+01	8.98E+01	8.98E+01	8.98E+01	8.98E+01	8.98E+01	8.98E+01	8.98E+01	8.98E+01	8.98E+01	8.98E+01
sn118	4.25	l f	2.79E+02	2.79E+02	2.79E+02	2.79E+02	2.79E+02	2.79E+02	2.79E+02	2.79E+02	2.79E+02	2.79E+02	2.79E+02	2.79E+02
sn119	4.25	l f	1.02E+02	1.02E+02	1.02E+02	1.02E+02	1.02E+02	1.02E+02	1.02E+02	1.02E+02	1.02E+02	1.02E+02	1.03E+02	1.03E+02
sn119m	4.25	l f	1.03E+00	1.03E+00	1.03E+00	1.03E+00	1.03E+00	1.03E+00	1.02E+00	9.57E-01	8.30E-01	6.71E-01	4.33E-01	7.70E-02
sn120	4.25	l f	3.78E+02	3.78E+02	3.78E+02	3.78E+02	3.78E+02	3.78E+02	3.78E+02	3.78E+02	3.78E+02	3.78E+02	3.78E+02	3.78E+02
sn121	4.25	f	1.33E-03	1.33E-03	1.32E-03	1.30E-03	1.09E-03	7.21E-04	1.14E-04	6.14E-07	6.12E-07	6.11E-07	6.07E-07	5.92E-07
sn121m	4.25	f	1.41E-02	1.41E-02	1.41E-02	1.41E-02	1.41E-02	1.41E-02	1.41E-02	1.41E-02	1.41E-02	1.40E-02	1.39E-02	1.36E-02
sb121	4.25	l f	2.14E+00	2.14E+00	2.14E+00	2.14E+00	2.14E+00	2.14E+00	2.14E+00	2.14E+00	2.14E+00	2.14E+00	2.14E+00	2.14E+00
sn122	4.25	l f	5.58E+01	5.58E+01	5.58E+01	5.58E+01	5.58E+01	5.58E+01	5.58E+01	5.58E+01	5.58E+01	5.58E+01	5.58E+01	5.58E+01
te122	4.25	l f	2.54E-01	2.54E-01	2.54E-01	2.54E-01	2.54E-01	2.54E-01	2.54E-01	2.54E-01	2.54E-01	2.54E-01	2.54E-01	2.54E-01
sn123	4.25	l f	2.39E-02	2.39E-02	2.39E-02	2.39E-02	2.39E-02	2.38E-02	2.34E-02	2.03E-02	1.47E-02	9.10E-03	3.37E-03	6.68E-05
sb123	4.25	l f	1.51E+00	1.51E+00	1.51E+00	1.51E+00	1.51E+00	1.51E+00	1.51E+00	1.51E+00	1.52E+00	1.53E+00	1.53E+00	1.53E+00
te123	4.25	f	1.49E-03	1.49E-03	1.49E-03	1.49E-03	1.49E-03	1.50E-03	1.50E-03	1.57E-03	1.68E-03	1.79E-03	1.90E-03	1.95E-03
sn124	4.25	l f	6.92E+01	6.92E+01	6.92E+01	6.92E+01	6.92E+01	6.92E+01	6.92E+01	6.92E+01	6.92E+01	6.92E+01	6.92E+01	6.92E+01
sb124	4.25	f	1.28E-02	1.28E-02	1.28E-02	1.28E-02	1.28E-02	1.27E-02	1.23E-02	9.09E-03	4.56E-03	1.62E-03	1.91E-04	4.26E-08
te124	4.25	l f	1.29E-01	1.29E-01	1.29E-01	1.29E-01	1.29E-01	1.29E-01	1.29E-01	1.32E-01	1.37E-01	1.41E-01	1.42E-01	1.43E-01
sn125	4.25	f	7.27E-03	7.27E-03	7.26E-03	7.25E-03	7.10E-03	6.77E-03	5.45E-03	8.41E-04	1.12E-05	1.74E-08	2.86E-14	0.00E+00
sb125	4.25	l f	2.36E+00	2.36E+00	2.36E+00	2.36E+00	2.36E+00	2.36E+00	2.36E+00	2.32E+00	2.22E+00	2.09E+00	1.84E+00	1.11E+00
te125	4.25	l f	1.70E+00	1.70E+00	1.70E+00	1.70E+00	1.70E+00	1.70E+00	1.71E+00	1.75E+00	1.85E+00	1.98E+00	2.23E+00	2.98E+00
te125m	4.25	f	2.58E-02	2.58E-02	2.58E-02	2.58E-02	2.58E-02	2.58E-02	2.58E-02	2.58E-02	2.53E-02	2.42E-02	2.14E-02	1.29E-02
sn126	4.25	f	6.37E+00	6.37E+00	6.37E+00	6.37E+00	6.37E+00	6.37E+00	6.37E+00	6.37E+00	6.37E+00	6.37E+00	6.37E+00	6.37E+00
te126	4.25	l f	1.56E-01	1.56E-01	1.56E-01	1.56E-01	1.56E-01	1.56E-01	1.57E-01	1.58E-01	1.58E-01	1.58E-01	1.58E-01	1.58E-01
sb127	4.25	f	5.19E-02	5.19E-02	5.18E-02	5.16E-02	4.93E-02	4.38E-02	2.55E-02	2.36E-04	4.80E-09	4.40E-16	1.44E-30	0.00E+00
te127	4.25	f	5.21E-03	5.21E-03	5.21E-03	5.21E-03	5.16E-03	4.85E-03	3.24E-03	7.72E-04	5.13E-04	2.89E-04	8.90E-05	8.55E-07
te127m	4.25	f	2.50E-01	2.50E-01	2.50E-01	2.50E-01	2.50E-01	2.50E-01	2.48E-01	2.14E-01	1.46E-01	8.26E-02	2.54E-02	2.44E-04
i127	4.25	f	1.32E+01	1.32E+01	1.32E+01	1.32E+01	1.32E+01	1.32E+01	1.32E+01	1.33E+01	1.33E+01	1.34E+01	1.35E+01	1.35E+01
te128	4.25	f	2.89E+01	2.89E+01	2.89E+01	2.89E+01	2.89E+01	2.89E+01	2.89E+01	2.89E+01	2.89E+01	2.89E+01	2.89E+01	2.89E+01

Table 3.4

Mass (grams) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Isotopic Mass (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
xe128	4.25	f	1.43E+00	1.43E+00	1.43E+00	1.43E+00	1.43E+00	1.43E+00	1.43E+00	1.43E+00	1.43E+00	1.43E+00	1.43E+00	1.43E+00
sb129	4.25	f	8.63E-03	8.63E-03	8.08E-03	7.47E-03	2.48E-03	1.99E-04	2.36E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te129	4.25	f	2.17E-03	2.17E-03	2.15E-03	2.10E-03	1.00E-03	3.36E-04	2.60E-04	1.52E-04	4.42E-05	6.90E-06	1.51E-07	4.31E-14
te129m	4.25	f	3.05E-01	3.05E-01	3.05E-01	3.05E-01	3.04E-01	3.00E-01	2.82E-01	1.65E-01	4.79E-02	7.48E-03	1.64E-04	4.67E-11
i129	4.25	f	5.61E+01	5.61E+01	5.61E+01	5.61E+01	5.61E+01	5.61E+01	5.61E+01	5.62E+01	5.64E+01	5.64E+01	5.64E+01	5.64E+01
xe129	4.25	f	1.41E-02	1.41E-02	1.41E-02	1.41E-02	1.41E-02	1.42E-02	1.42E-02	1.43E-02	1.43E-02	1.43E-02	1.43E-02	1.43E-02
te130	4.25	f	1.15E+02	1.15E+02	1.15E+02	1.15E+02	1.15E+02	1.15E+02	1.15E+02	1.15E+02	1.15E+02	1.15E+02	1.15E+02	1.15E+02
i130	4.25	f	3.29E-03	3.29E-03	3.22E-03	3.13E-03	2.11E-03	8.61E-04	1.52E-05	9.67E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe130	4.25	f	3.84E+00	3.84E+00	3.84E+00	3.84E+00	3.84E+00	3.84E+00	3.84E+00	3.84E+00	3.84E+00	3.84E+00	3.84E+00	3.84E+00
sb131	4.5	f	1.73E-03	1.73E-03	7.09E-04	2.87E-04	9.13E-10	2.49E-22	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te131	4.5	f	2.06E-03	2.06E-03	1.58E-03	1.00E-03	9.93E-05	6.86E-05	1.30E-05	7.12E-12	2.53E-26	0.00E+00	0.00E+00	0.00E+00
te131m	4.25	f	3.81E-02	3.81E-02	3.77E-02	3.73E-02	3.17E-02	2.19E-02	4.16E-03	2.28E-09	8.09E-24	0.00E+00	0.00E+00	0.00E+00
i131	4.5	f	1.14E+00	1.14E+00	1.14E+00	1.14E+00	1.12E+00	1.06E+00	8.36E-01	8.94E-02	5.07E-04	2.16E-07	2.51E-14	0.00E+00
xe131	4.5	f	1.08E+02	1.08E+02	1.08E+02	1.08E+02	1.08E+02	1.08E+02	1.08E+02	1.09E+02	1.09E+02	1.09E+02	1.09E+02	1.09E+02
xe131m	4.25	f	2.56E-02	2.56E-02	2.56E-02	2.56E-02	2.55E-02	2.52E-02	2.36E-02	8.38E-03	3.28E-04	1.82E-06	3.75E-11	1.23E-29
te132	4.5	f	6.63E-01	6.63E-01	6.61E-01	6.58E-01	6.18E-01	5.37E-01	2.83E-01	1.12E-03	3.21E-09	1.55E-17	1.19E-34	0.00E+00
i132	4.5	f	1.98E-02	1.98E-02	1.97E-02	1.97E-02	1.86E-02	1.61E-02	8.53E-03	3.38E-05	9.66E-11	4.67E-19	3.57E-36	0.00E+00
xe132	4.25	f	3.71E+02	3.71E+02	3.71E+02	3.71E+02	3.71E+02	3.71E+02	3.71E+02	3.72E+02	3.72E+02	3.72E+02	3.72E+02	3.72E+02
te133	4.5	f	1.34E-03	1.34E-03	4.09E-04	1.63E-04	6.15E-07	3.74E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te133m	4.5	f	4.87E-03	4.87E-03	3.36E-03	2.31E-03	1.21E-05	7.33E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i133	4.5	f	2.52E-01	2.52E-01	2.50E-01	2.47E-01	1.98E-01	1.16E-01	1.05E-02	9.81E-12	1.42E-32	0.00E+00	0.00E+00	0.00E+00
xe133	4.5	f	1.46E+00	1.46E+00	1.46E+00	1.46E+00	1.46E+00	1.41E+00	1.04E+00	3.43E-02	1.23E-05	8.38E-11	1.94E-21	0.00E+00
xe133m	4.25	f	2.06E-02	2.06E-02	2.05E-02	2.05E-02	2.01E-02	1.84E-02	8.75E-03	2.47E-06	1.39E-14	5.91E-27	0.00E+00	0.00E+00
cs133	4.5	f	3.33E+02	3.33E+02	3.33E+02	3.33E+02	3.33E+02	3.33E+02	3.34E+02	3.35E+02	3.35E+02	3.35E+02	3.35E+02	3.35E+02
te134	4.5	f	7.12E-03	7.12E-03	4.33E-03	2.63E-03	2.49E-06	3.04E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i134	4.5	f	1.17E-02	1.17E-02	1.02E-02	8.24E-03	7.11E-05	2.65E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe134	4.5	f	4.75E+02	4.75E+02	4.75E+02	4.75E+02	4.75E+02	4.75E+02	4.75E+02	4.75E+02	4.75E+02	4.75E+02	4.75E+02	4.75E+02
cs134	4.25	f	4.31E+01	4.31E+01	4.31E+01	4.31E+01	4.31E+01	4.31E+01	4.30E+01	4.20E+01	3.97E+01	3.66E+01	3.08E+01	1.57E+01
cs134m	4.25	f	1.23E-03	1.23E-03	1.09E-03	9.72E-04	1.84E-04	4.07E-06	1.46E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba134	4.25	f	3.01E+01	3.01E+01	3.01E+01	3.01E+01	3.01E+01	3.01E+01	3.02E+01	3.13E+01	3.35E+01	3.67E+01	4.24E+01	5.75E+01
i135	4.5	f	7.74E-02	7.74E-02	7.35E-02	6.97E-02	3.33E-02	6.16E-03	3.09E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe135	4.5	f	2.95E-02	2.95E-02	3.25E-02	3.50E-02	4.81E-02	2.76E-02	2.00E-04	5.89E-25	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs135	4.5	f	1.50E+02	1.50E+02	1.50E+02	1.50E+02	1.50E+02	1.50E+02	1.50E+02	1.50E+02	1.50E+02	1.50E+02	1.50E+02	1.50E+02
ba135	4.25	f	3.89E-01	3.89E-01	3.89E-01	3.89E-01	3.89E-01	3.89E-01	3.89E-01	3.89E-01	3.89E-01	3.89E-01	3.89E-01	3.89E-01
xe136	4.25	f	6.92E+02	6.92E+02	6.92E+02	6.92E+02	6.92E+02	6.92E+02	6.92E+02	6.92E+02	6.92E+02	6.92E+02	6.92E+02	6.92E+02

Table 3.4

Mass (grams) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Isotopic Mass (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
cs136	4.5	f	2.11E-01	2.11E-01	2.11E-01	2.10E-01	2.07E-01	2.00E-01	1.71E-01	4.34E-02	1.84E-03	1.61E-05	9.30E-10	1.81E-26
ba136	4.5	f	1.05E+01	1.05E+01	1.05E+01	1.05E+01	1.05E+01	1.05E+01	1.05E+01	1.07E+01	1.07E+01	1.07E+01	1.07E+01	1.07E+01
cs137	4.5	f	3.80E+02	3.80E+02	3.80E+02	3.80E+02	3.80E+02	3.80E+02	3.80E+02	3.79E+02	3.77E+02	3.75E+02	3.71E+02	3.54E+02
ba137	4.5	f	2.44E+01	2.44E+01	2.44E+01	2.44E+01	2.44E+01	2.45E+01	2.45E+01	2.51E+01	2.66E+01	2.87E+01	3.31E+01	4.99E+01
xe138	4.5	f	2.43E-03	2.43E-03	5.57E-04	1.27E-04	1.33E-13	3.98E-34	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs138	4.5	f	6.10E-03	6.10E-03	4.50E-03	2.65E-03	3.41E-07	3.61E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba138	4.5	f	4.03E+02	4.03E+02	4.03E+02	4.03E+02	4.03E+02	4.03E+02	4.03E+02	4.03E+02	4.03E+02	4.03E+02	4.03E+02	4.03E+02
la138	4.25	f	2.85E-03	2.85E-03	2.85E-03	2.85E-03	2.85E-03	2.85E-03	2.85E-03	2.85E-03	2.85E-03	2.85E-03	2.85E-03	2.85E-03
cs139	4.5	f	1.63E-03	1.63E-03	1.83E-04	1.94E-05	4.46E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba139	4.5	f	1.54E-02	1.54E-02	1.33E-02	1.06E-02	3.39E-04	1.31E-07	5.63E-23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la139	4.5	f	3.76E+02	3.76E+02	3.76E+02	3.76E+02	3.76E+02	3.76E+02	3.76E+02	3.76E+02	3.76E+02	3.76E+02	3.76E+02	3.76E+02
ba140	4.5	f	3.37E+00	3.37E+00	3.37E+00	3.37E+00	3.31E+00	3.19E+00	2.71E+00	6.61E-01	2.53E-02	1.90E-04	8.08E-09	4.64E-26
la140	4.5	f	4.86E-01	4.86E-01	4.86E-01	4.86E-01	4.80E-01	4.68E-01	4.06E-01	1.00E-01	3.84E-03	2.89E-05	1.22E-09	7.03E-27
ce140	4.5	f	4.08E+02	4.08E+02	4.08E+02	4.08E+02	4.08E+02	4.08E+02	4.08E+02	4.11E+02	4.12E+02	4.12E+02	4.12E+02	4.12E+02
ba141	4.5	f	3.03E-03	3.03E-03	9.88E-04	3.16E-04	3.79E-11	5.73E-27	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la141	4.5	f	3.94E-02	3.94E-02	3.81E-02	3.55E-02	1.04E-02	6.14E-04	1.81E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce141	4.5	f	7.88E+00	7.88E+00	7.88E+00	7.88E+00	7.86E+00	7.76E+00	7.28E+00	4.18E+00	1.16E+00	1.71E-01	3.28E-03	5.62E-10
pr141	4.5	f	3.38E+02	3.38E+02	3.38E+02	3.38E+02	3.38E+02	3.38E+02	3.38E+02	3.42E+02	3.45E+02	3.46E+02	3.46E+02	3.46E+02
ba142	4.5	f	1.66E-03	1.66E-03	2.34E-04	3.29E-05	3.89E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la142	4.5	f	1.49E-02	1.49E-02	1.31E-02	1.06E-02	4.35E-04	2.93E-07	1.55E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce142	4.5	f	3.53E+02	3.53E+02	3.53E+02	3.53E+02	3.53E+02	3.53E+02	3.53E+02	3.53E+02	3.53E+02	3.53E+02	3.53E+02	3.53E+02
pr142	4.25	f	1.49E-02	1.49E-02	1.46E-02	1.44E-02	1.11E-02	6.24E-03	4.58E-04	6.86E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd142	4.25	f	1.05E+01	1.05E+01	1.05E+01	1.05E+01	1.05E+01	1.05E+01	1.05E+01	1.05E+01	1.05E+01	1.05E+01	1.05E+01	1.05E+01
la143	4.5	f	2.16E-03	2.16E-03	5.05E-04	1.16E-04	1.33E-13	4.85E-34	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce143	4.5	f	3.07E-01	3.07E-01	3.05E-01	3.02E-01	2.61E-01	1.87E-01	4.11E-02	8.35E-08	6.11E-21	1.21E-40	0.00E+00	0.00E+00
pr143	4.5	f	2.92E+00	2.92E+00	2.92E+00	2.92E+00	2.92E+00	2.89E+00	2.61E+00	7.04E-01	3.28E-02	3.30E-04	2.56E-08	1.57E-24
nd143	4.5	f	2.07E+02	2.07E+02	2.07E+02	2.07E+02	2.07E+02	2.07E+02	2.08E+02	2.10E+02	2.11E+02	2.11E+02	2.11E+02	2.11E+02
ce144	4.5	f	5.93E+01	5.93E+01	5.93E+01	5.93E+01	5.93E+01	5.92E+01	5.87E+01	5.51E+01	4.77E+01	3.83E+01	2.44E+01	4.13E+00
pr144	4.5	f	2.51E-03	2.51E-03	2.50E-03	2.50E-03	2.50E-03	2.49E-03	2.47E-03	2.32E-03	2.01E-03	1.61E-03	1.03E-03	1.74E-04
nd144	4.25	f	3.91E+02	3.91E+02	3.91E+02	3.91E+02	3.91E+02	3.91E+02	3.92E+02	3.95E+02	4.03E+02	4.12E+02	4.26E+02	4.46E+02
pr145	4.5	f	3.87E-02	3.87E-02	3.69E-02	3.48E-02	1.55E-02	2.42E-03	5.78E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd145	4.5	f	1.95E+02	1.95E+02	1.95E+02	1.95E+02	1.95E+02	1.95E+02	1.95E+02	1.95E+02	1.95E+02	1.95E+02	1.95E+02	1.95E+02
ce146	4.5	f	1.19E-03	1.19E-03	2.57E-04	5.52E-05	2.46E-14	1.04E-35	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr146	4.5	f	2.14E-03	2.14E-03	1.47E-03	7.43E-04	5.05E-09	5.46E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd146	4.5	f	2.32E+02	2.32E+02	2.32E+02	2.32E+02	2.32E+02	2.32E+02	2.32E+02	2.32E+02	2.32E+02	2.32E+02	2.32E+02	2.32E+02

Table 3.4

Mass (grams) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Isotopic Mass (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
pm146	4.25	f	2.61E-03	2.61E-03	2.61E-03	2.61E-03	2.61E-03	2.61E-03	2.61E-03	2.58E-03	2.53E-03	2.45E-03	2.30E-03	1.79E-03
sm146	4.5	f	3.84E-03	3.84E-03	3.84E-03	3.84E-03	3.84E-03	3.84E-03	3.85E-03	3.85E-03	3.87E-03	3.90E-03	3.95E-03	4.12E-03
nd147	4.5	f	1.15E+00	1.15E+00	1.15E+00	1.15E+00	1.12E+00	1.08E+00	8.92E-01	1.73E-01	3.91E-03	1.33E-05	1.11E-10	1.05E-30
pm147	4.5	f	3.39E+01	3.39E+01	3.39E+01	3.39E+01	3.39E+01	3.39E+01	3.40E+01	3.41E+01	3.28E+01	3.08E+01	2.69E+01	1.59E+01
sm147	4.5	f	2.54E+01	2.54E+01	2.54E+01	2.54E+01	2.54E+01	2.54E+01	2.55E+01	2.61E+01	2.76E+01	2.97E+01	3.35E+01	4.46E+01
nd148	4.5	f	1.14E+02	1.14E+02	1.14E+02	1.14E+02	1.14E+02	1.14E+02	1.14E+02	1.14E+02	1.14E+02	1.14E+02	1.14E+02	1.14E+02
pm148	4.5	f	1.75E-01	1.75E-01	1.75E-01	1.74E-01	1.68E-01	1.54E-01	1.05E-01	4.54E-03	3.41E-04	7.49E-05	3.34E-06	1.58E-11
pm148m	4.5	f	2.24E-01	2.24E-01	2.23E-01	2.23E-01	2.22E-01	2.20E-01	2.09E-01	1.35E-01	4.94E-02	1.09E-02	4.86E-04	2.30E-09
sm148	4.25	f	5.25E+01	5.25E+01	5.25E+01	5.25E+01	5.25E+01	5.25E+01	5.26E+01	5.28E+01	5.29E+01	5.29E+01	5.29E+01	5.29E+01
nd149	4.25	f	4.68E-03	4.68E-03	3.90E-03	3.19E-03	1.92E-04	3.09E-07	8.43E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm149	4.25	f	2.33E-01	2.33E-01	2.32E-01	2.31E-01	2.14E-01	1.74E-01	6.78E-02	1.96E-05	1.34E-13	7.54E-26	0.00E+00	0.00E+00
sm149	4.5	f	3.99E-01	3.99E-01	4.01E-01	4.02E-01	4.23E-01	4.63E-01	5.68E-01	6.35E-01	6.35E-01	6.35E-01	6.35E-01	6.35E-01
nd150	4.25	f	5.70E+01	5.70E+01	5.70E+01	5.70E+01	5.70E+01	5.70E+01	5.70E+01	5.70E+01	5.70E+01	5.70E+01	5.70E+01	5.70E+01
sm150	4.25	f	1.02E+02	1.02E+02	1.02E+02	1.02E+02	1.02E+02	1.02E+02	1.02E+02	1.02E+02	1.02E+02	1.02E+02	1.02E+02	1.02E+02
pm151	4.25	f	4.31E-02	4.31E-02	4.29E-02	4.24E-02	3.57E-02	2.42E-02	4.17E-03	1.01E-09	5.42E-25	0.00E+00	0.00E+00	0.00E+00
sm151	4.5	f	3.74E+00	3.74E+00	3.74E+00	3.74E+00	3.75E+00	3.76E+00	3.78E+00	3.78E+00	3.78E+00	3.77E+00	3.76E+00	3.70E+00
eu151	4.5	f	4.47E-03	4.47E-03	4.47E-03	4.47E-03	4.50E-03	4.55E-03	4.79E-03	6.86E-03	1.16E-02	1.88E-02	3.35E-02	9.10E-02
sm152	4.25	f	3.69E+01	3.69E+01	3.69E+01	3.69E+01	3.69E+01	3.69E+01	3.69E+01	3.69E+01	3.69E+01	3.69E+01	3.69E+01	3.69E+01
eu152	4.5	f	1.09E-02	1.09E-02	1.09E-02	1.09E-02	1.09E-02	1.09E-02	1.09E-02	1.09E-02	1.08E-02	1.06E-02	1.04E-02	9.34E-03
gd152	4.5	l f	2.28E+00	2.28E+00	2.28E+00	2.28E+00	2.28E+00	2.28E+00	2.28E+00	2.28E+00	2.28E+00	2.28E+00	2.28E+00	2.29E+00
sm153	4.25	f	2.35E-01	2.35E-01	2.33E-01	2.31E-01	2.08E-01	1.64E-01	5.57E-02	4.87E-06	2.09E-15	1.86E-29	0.00E+00	0.00E+00
eu153	4.25	l f	4.01E+01	4.01E+01	4.01E+01	4.01E+01	4.01E+01	4.01E+01	4.02E+01	4.03E+01	4.03E+01	4.04E+01	4.04E+01	4.05E+01
gd153	4.25	l f	2.98E-01	2.98E-01	2.98E-01	2.98E-01	2.97E-01	2.97E-01	2.94E-01	2.73E-01	2.30E-01	1.77E-01	1.05E-01	1.28E-02
sm154	4.25	f	1.24E+01	1.24E+01	1.24E+01	1.24E+01	1.24E+01	1.24E+01	1.24E+01	1.24E+01	1.24E+01	1.24E+01	1.24E+01	1.24E+01
eu154	4.25	l f	8.90E+00	8.90E+00	8.90E+00	8.90E+00	8.90E+00	8.90E+00	8.90E+00	8.84E+00	8.72E+00	8.55E+00	8.21E+00	6.99E+00
gd154	4.5	l f	2.78E+01	2.78E+01	2.78E+01	2.78E+01	2.78E+01	2.78E+01	2.78E+01	2.78E+01	2.80E+01	2.81E+01	2.85E+01	2.97E+01
eu155	4.25	l f	2.06E+00	2.06E+00	2.06E+00	2.06E+00	2.06E+00	2.06E+00	2.05E+00	2.03E+00	1.98E+00	1.91E+00	1.77E+00	1.32E+00
gd155	4.5	l f	1.63E-01	1.63E-01	1.63E-01	1.63E-01	1.63E-01	1.64E-01	1.66E-01	1.87E-01	2.35E-01	3.05E-01	4.41E-01	8.93E-01
sm156	4.25	f	1.98E-03	1.98E-03	1.91E-03	1.84E-03	1.10E-03	3.37E-04	1.67E-06	1.73E-26	0.00E+00	0.00E+00	0.00E+00	0.00E+00
eu156	4.25	l f	1.33E+00	1.33E+00	1.33E+00	1.33E+00	1.31E+00	1.27E+00	1.10E+00	3.38E-01	2.19E-02	3.59E-04	7.63E-08	2.52E-22
gd156	4.25	l f	7.13E+02	7.13E+02	7.13E+02	7.13E+02	7.13E+02	7.13E+02	7.13E+02	7.14E+02	7.14E+02	7.14E+02	7.14E+02	7.14E+02
eu157	4.25	f	4.52E-03	4.52E-03	4.44E-03	4.34E-03	3.15E-03	1.52E-03	5.67E-05	2.40E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00
gd157	4.5	l f	1.82E-01	1.82E-01	1.82E-01	1.82E-01	1.84E-01	1.85E-01	1.87E-01	1.87E-01	1.87E-01	1.87E-01	1.87E-01	1.87E-01
gd158	4.25	l f	8.70E+02	8.70E+02	8.70E+02	8.70E+02	8.70E+02	8.70E+02	8.70E+02	8.70E+02	8.70E+02	8.70E+02	8.70E+02	8.70E+02
gd159	4.25	l f	4.10E-02	4.10E-02	4.03E-02	3.96E-02	3.05E-02	1.68E-02	1.14E-03	8.64E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 3.4

Mass (grams) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Isotopic Mass (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
tb159	4.25	l f	4.04E+01	4.04E+01	4.04E+01	4.04E+01	4.04E+01	4.04E+01	4.04E+01	4.04E+01	4.04E+01	4.04E+01	4.04E+01	4.04E+01
gd160	4.25	l f	4.53E+02	4.53E+02	4.53E+02	4.53E+02	4.53E+02	4.53E+02	4.53E+02	4.53E+02	4.53E+02	4.53E+02	4.53E+02	4.53E+02
tb160	4.25	l f	1.45E+00	1.45E+00	1.45E+00	1.45E+00	1.45E+00	1.44E+00	1.40E+00	1.09E+00	6.13E-01	2.59E-01	4.37E-02	3.98E-05
dy160	4.25	l f	6.17E+00	6.17E+00	6.17E+00	6.17E+00	6.17E+00	6.18E+00	6.22E+00	6.54E+00	7.01E+00	7.36E+00	7.58E+00	7.62E+00
tb161	4.25	l f	6.87E-02	6.87E-02	6.85E-02	6.84E-02	6.65E-02	6.21E-02	4.60E-02	3.38E-03	8.14E-06	9.65E-10	8.00E-18	0.00E+00
dy161	4.25	l f	5.73E+00	5.73E+00	5.73E+00	5.73E+00	5.73E+00	5.74E+00	5.75E+00	5.79E+00	5.80E+00	5.80E+00	5.80E+00	5.80E+00
dy162	4.25	l f	3.82E+00	3.82E+00	3.82E+00	3.82E+00	3.82E+00	3.82E+00	3.82E+00	3.82E+00	3.82E+00	3.82E+00	3.82E+00	3.82E+00
dy163	4.25	l f	3.00E+00	3.00E+00	3.00E+00	3.00E+00	3.00E+00	3.00E+00	3.00E+00	3.00E+00	3.00E+00	3.00E+00	3.00E+00	3.00E+00
dy164	4.25	l f	5.69E-01	5.69E-01	5.69E-01	5.69E-01	5.69E-01	5.69E-01	5.69E-01	5.69E-01	5.69E-01	5.69E-01	5.69E-01	5.69E-01
ho165	4.25	l f	9.31E-01	9.31E-01	9.31E-01	9.31E-01	9.31E-01	9.32E-01	9.32E-01	9.32E-01	9.32E-01	9.32E-01	9.32E-01	9.32E-01
er166	4.25	l f	2.17E-01	2.17E-01	2.17E-01	2.17E-01	2.17E-01	2.18E-01	2.18E-01	2.18E-01	2.18E-01	2.18E-01	2.18E-01	2.18E-01
hf176	4.5	l	8.84E-02	8.84E-02	8.84E-02	8.84E-02	8.84E-02	8.84E-02	8.84E-02	8.84E-02	8.84E-02	8.84E-02	8.84E-02	8.84E-02
hf177	4.5	l	9.62E-03	9.62E-03	9.62E-03	9.62E-03	9.62E-03	9.62E-03	9.62E-03	9.63E-03	9.63E-03	9.63E-03	9.63E-03	9.63E-03
hf178	4.5	l	1.10E-01	1.10E-01	1.10E-01	1.10E-01	1.10E-01	1.10E-01	1.10E-01	1.10E-01	1.10E-01	1.10E-01	1.10E-01	1.10E-01
hf179	4.5	l	1.78E+00	1.78E+00	1.78E+00	1.78E+00	1.78E+00	1.78E+00	1.78E+00	1.78E+00	1.78E+00	1.78E+00	1.78E+00	1.78E+00
hf180	4.25	l	5.58E+00	5.58E+00	5.58E+00	5.58E+00	5.58E+00	5.58E+00	5.58E+00	5.58E+00	5.58E+00	5.58E+00	5.58E+00	5.58E+00
hf181	4.25	l	2.63E-02	2.63E-02	2.63E-02	2.63E-02	2.62E-02	2.59E-02	2.46E-02	1.61E-02	6.03E-03	1.38E-03	6.69E-05	4.34E-10
ta181	4.25	l	3.46E-01	3.46E-01	3.46E-01	3.46E-01	3.46E-01	3.46E-01	3.47E-01	3.56E-01	3.66E-01	3.71E-01	3.72E-01	3.72E-01
ta182	4.25	l	8.99E-03	8.99E-03	8.99E-03	8.99E-03	8.97E-03	8.94E-03	8.78E-03	7.50E-03	5.23E-03	3.04E-03	9.95E-04	1.22E-05
w182	4.5	l	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.01E+00	1.01E+00	1.01E+00
re185	4.25	l	3.18E-02	3.18E-02	3.18E-02	3.18E-02	3.18E-02	3.19E-02	3.21E-02	3.34E-02	3.54E-02	3.70E-02	3.79E-02	3.82E-02
os186	4.25	l	4.61E-02	4.61E-02	4.61E-02	4.61E-02	4.61E-02	4.62E-02	4.63E-02	4.64E-02	4.64E-02	4.64E-02	4.64E-02	4.64E-02
re187	4.25	l	8.95E-01	8.95E-01	8.95E-01	8.95E-01	8.95E-01	8.95E-01	8.95E-01	8.96E-01	8.96E-01	8.96E-01	8.96E-01	8.96E-01
os188	4.25	l	3.05E-01	3.05E-01	3.05E-01	3.05E-01	3.05E-01	3.05E-01	3.05E-01	3.05E-01	3.05E-01	3.05E-01	3.05E-01	3.05E-01
os189	4.25	l	2.32E-02	2.32E-02	2.32E-02	2.32E-02	2.32E-02	2.32E-02	2.32E-02	2.32E-02	2.32E-02	2.32E-02	2.32E-02	2.32E-02
os190	4.25	l	1.10E-02	1.10E-02	1.10E-02	1.10E-02	1.10E-02	1.10E-02	1.10E-02	1.10E-02	1.10E-02	1.10E-02	1.10E-02	1.10E-02
u234	4.5	a	4.61E+00	4.61E+00	4.61E+00	4.61E+00	4.61E+00	4.62E+00	4.62E+00	4.67E+00	4.77E+00	4.93E+00	5.26E+00	6.57E+00
u235	4.5	a	1.03E+03	1.03E+03	1.03E+03	1.03E+03	1.03E+03	1.03E+03	1.03E+03	1.03E+03	1.03E+03	1.03E+03	1.03E+03	1.03E+03
u236	4.5	a	1.16E+03	1.16E+03	1.16E+03	1.16E+03	1.16E+03	1.16E+03	1.16E+03	1.16E+03	1.16E+03	1.16E+03	1.16E+03	1.16E+03
u237	4.5	a	2.04E+00	2.04E+00	2.04E+00	2.03E+00	1.97E+00	1.84E+00	1.35E+00	9.37E-02	2.07E-04	9.01E-06	8.77E-06	7.96E-06
np237	4.5	a	1.48E+02	1.48E+02	1.48E+02	1.48E+02	1.48E+02	1.48E+02	1.48E+02	1.50E+02	1.50E+02	1.50E+02	1.50E+02	1.50E+02
u238	4.5	a	1.65E+05	1.65E+05	1.65E+05	1.65E+05	1.65E+05	1.65E+05	1.65E+05	1.65E+05	1.65E+05	1.65E+05	1.65E+05	1.65E+05
np238	4.25	a	4.30E-01	4.30E-01	4.27E-01	4.24E-01	3.85E-01	3.10E-01	1.16E-01	2.33E-05	5.21E-08	5.20E-08	5.19E-08	5.14E-08
pu238	4.5	a	8.05E+01	8.05E+01	8.05E+01	8.05E+01	8.06E+01	8.07E+01	8.09E+01	8.15E+01	8.24E+01	8.34E+01	8.43E+01	8.40E+01
u239	4.25	a	9.76E-02	9.76E-02	4.02E-02	1.66E-02	6.79E-08	3.29E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

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Table 3.4

Mass (grams) per Single Assembly of ATRIUM-10 Fuel - Sum of Actinides, Fission Products and Light Elements
Decay time Following Burnup to 58 GWd/MTU
Maximum Isotopic Mass (4.25 wt% U-235 and 4.50 wt% U-235)

Nuclide	limit enr		0.0 d	1 sec	30 min	1 hr	8 hr	1.0 d	4.0 d	30.0 d	90.0 d	180.0 d	1 yr	3 yr
np239	4.25	a	1.41E+01	1.41E+01	1.40E+01	1.40E+01	1.28E+01	1.06E+01	4.36E+00	2.13E-03	5.32E-05	5.32E-05	5.32E-05	5.32E-05
pu239	4.5	a	9.88E+02	9.88E+02	9.88E+02	9.88E+02	9.89E+02	9.92E+02	9.98E+02	1.00E+03	1.00E+03	1.00E+03	1.00E+03	1.00E+03
pu240	4.25	a	5.81E+02	5.81E+02	5.81E+02	5.81E+02	5.81E+02	5.81E+02	5.81E+02	5.81E+02	5.81E+02	5.82E+02	5.82E+02	5.84E+02
pu241	4.5	a	3.04E+02	3.04E+02	3.04E+02	3.04E+02	3.04E+02	3.04E+02	3.04E+02	3.03E+02	3.00E+02	2.97E+02	2.89E+02	2.63E+02
am241	4.5	a	1.35E+01	1.35E+01	1.35E+01	1.35E+01	1.35E+01	1.35E+01	1.37E+01	1.47E+01	1.71E+01	2.06E+01	2.78E+01	5.43E+01
pu242	4.25	a	2.14E+02	2.14E+02	2.14E+02	2.14E+02	2.14E+02	2.14E+02	2.14E+02	2.14E+02	2.14E+02	2.14E+02	2.14E+02	2.14E+02
am242	4.25	a	3.04E-02	3.04E-02	2.97E-02	2.91E-02	2.15E-02	1.08E-02	4.80E-04	3.70E-06	3.70E-06	3.69E-06	3.68E-06	3.65E-06
am242m	4.5	a	2.87E-01	2.87E-01	2.87E-01	2.87E-01	2.87E-01	2.87E-01	2.87E-01	2.87E-01	2.86E-01	2.86E-01	2.85E-01	2.83E-01
am242	4.25	a	3.04E-02	3.04E-02	2.97E-02	2.91E-02	2.15E-02	1.08E-02	4.80E-04	3.70E-06	3.70E-06	3.69E-06	3.68E-06	3.65E-06
cm242	4.25	a	5.17E+00	5.17E+00	5.17E+00	5.17E+00	5.17E+00	5.17E+00	5.11E+00	4.58E+00	3.55E+00	2.42E+00	1.10E+00	4.99E-02
pu243	4.25	a	4.70E-02	4.70E-02	4.38E-02	4.08E-02	1.53E-02	1.64E-03	6.92E-08	3.70E-13	3.70E-13	3.70E-13	3.70E-13	3.70E-13
am243	4.25	a	6.17E+01	6.17E+01	6.17E+01	6.17E+01	6.18E+01	6.18E+01	6.18E+01	6.18E+01	6.18E+01	6.18E+01	6.18E+01	6.18E+01
cm243	4.25	a	2.04E-01	2.04E-01	2.04E-01	2.04E-01	2.04E-01	2.04E-01	2.04E-01	2.04E-01	2.03E-01	2.02E-01	1.99E-01	1.90E-01
cm244	4.25	a	3.24E+01	3.24E+01	3.24E+01	3.24E+01	3.24E+01	3.24E+01	3.24E+01	3.23E+01	3.21E+01	3.18E+01	3.12E+01	2.89E+01
cm245	4.25	a	1.45E+00	1.45E+00	1.45E+00	1.45E+00	1.45E+00	1.45E+00	1.45E+00	1.45E+00	1.45E+00	1.45E+00	1.45E+00	1.45E+00
cm246	4.25	a	3.96E-01	3.96E-01	3.96E-01	3.96E-01	3.96E-01	3.96E-01	3.96E-01	3.96E-01	3.96E-01	3.96E-01	3.96E-01	3.96E-01

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