Attachment A – List of Affected FSAR Sections

FSAR Section	Title	Change Description	Page
Chapter 1: Introduction and General Description of the Plant			
1.1.7	Design Bases	replaced 10 CFR 100 with 10 CFR 50.67	1.1-3
Table 1.1-1 (Sheet 3)	Acronyms	Added CEDE, EDE, and TEDE acronyms	Table
Table 1.1-1 (Sheet 5)	Acronyms	Added CEDE, EDE, and TEDE acronyms	Table
Table 1.3-4 (Sheet 14)	Compliance with NRC Regulations	Added 50.67	Table
Table 1.3-4 (Sheet 26)	Compliance with NRC Regulations	Deleted text under 10 CFR 100.11	Table
Chapter 2: Site Characteristics			
2.3.4	Short-Term (Accident) Diffusion Estimates	Reference to Regulatory Guide 1.194, removed old methodology	2.3-1
2.3.4	Short-Term (Accident) Diffusion Estimates	Added insert 2.3.4.2.2.2, removed reference 1	2.3-2
Table 2.3-1	Limiting Atmospheric Dispersion Factor	Revised with NAI-1990-006	Table
Chapter 2 Addendum: Site Characteristics			
Addendum 2	TOC	Added new Section 2.3.4.4	2.0-ii
Addendum 2	List of Tables	Added Tables 2.3-87 through 2.3-98	2.0-xi
Addendum 2, 2.3.4	Short-Term Diffusion Estimates	Added note that Sections 2.3.4.1-2.3.4.3 are historical	2.3-59
Addendum 2	Alternative Source Term Short-Term Diffusion Estimates	Added new Section 2.3.4.4	2.3-64
Addendum 2 References	References Section 2.3	Added 2 References	2.3-75
Addendum 2 References	References Section 2.3	Add Reference	2.3-76
Chapter 3: Design Of Structures, Components, Equipment and Systems			
3.0	Design of Structures, Components, Equipment and Systems	Replaced 10 CFR 100 with 10 CFR 50.67 for offsite exposures	3.1-1
3.1.4	Criterion 19 - Control Room	replaced "whole body or its equivalent…" with TEDE	3.1-16
3.1.4	Criterion 19 - Control Room	Replaced 10 CFR 100 with 10 CFR 50.67 for offsite exposures	3.1-16
3.1.4	Criterion 19 - Control Room	deleted TID-14844 in discussion	3.1-16
3.1.4	Criterion 19 - Control Room	deleted "thyroid"	3.1-16
3.1.6	Criterion 41, Containment Atmosphere Cleanup	Replaced RG 1.4 with RG 1.183	3.1-32
3.2c	Classification of Structures, Components and Systems	replaced 10 CFR 100 with 10 CFR 50.67	3.2-1
3.6.2.3.2.1		replaced 10 CFR 100 with 10 CFR 50.67	3.6-23

FSAR Section	Title	Change Description	Page
3.7(N)	Seismic Design	replaced 10 CFR 100 with 10 CFR 50.67	3.7(N)-1
3.8.1.2.1	Regulations	Added c. 10 CFR 50.67, Accident Source Term	3.8-4
3.11(B).1.1.2c	Safety-Related System Listing	replaced 10 CFR 100 with 10 CFR 50.67	3.11(B)-3
3.11(B).1.2.2	Accident Environments - Inside Containment	Added note pointing to Chapter 15.6.5	3.11(B)-4
3.11(B).1.2.2	Safety-Related System Listing	replaced spray removal coefficients, DFs with AST values	3.11(B)-5
3.11(B).5.2	Equipment Operability	Replaced RG 1.4 with RG 1.183	3.11(B)-21
Appendix 3A	Conformance to NRC Regulatory guides	Added 1.183 and removed 1.4 and 1.25	3A-1
Appendix 3A	Conformance to NRC Regulatory Guides	Noted that RG 1.4 has been replaced with RG 1.183, deleted reference to Table 15.6.7	3A-3
Appendix 3A	Conformance to NRC Regulatory Guides	Noted the RG 1.25 has been replaced with RG 1.183, deleted reference to Table 15.7.2	3A-11
Appendix 3A	Conformance to NRC Regulatory Guides	Added Regulatory Guide 1.183 to list	3A-55
Appendix 3A	Conformance to NRC Regulatory Guides	Added Regulatory Guide 1.194 to list	3A-58
Chapter 3 Addendum: Design Of Structures, Components, Equipment and Systems			
Addendum 3	Appendix 3.A -Confirmance to NRC Regulatory Guides	Added Reg Guide 1.183 and 1.194 to the list, deleted REG 1.4 and 1.25 from list	3.A-1
Addendum 3	Appendix 3.A -Confirmance to NRC Regulatory Guides	Added Rev 1 of Reg Guide 1.145 and referred to Site Addendum Section 2.3.4.4	3.A-9
Addendum 3	Appendix 3.A -Confirmance to NRC Regulatory Guides	Added Reg Guide 1.194 and referred to Site Addendum Section 2.3.4.4	3.A-9
Chapter 4: Reactor			
4.2.1	Fuel System Design	replaced 10 CFR 100 with 10 CFR 50.67	4.2-1
4.3.1	Nuclear Design, Design Bases	replaced 10 CFR 100 with 10 CFR 50.67	4.3-1

FSAR Section	Title	Change Description	Page
Chapter 6: Engineered Safety Features			
6	TOC	Revised title, added section	6.0-vi
6.0	Engineered Safety Features	replaced 10 CFR 100 with 10 CFR 50.67	6.1-1
6.2.1.1.1	Design Bases	replaced 10 CFR 100 with 10 CFR 50.67	6.2.1-1
6.2.1.1.1f	Design Bases	replaced 10 CFR 100 with 10 CFR 50.67	6.2.1-4
6.3.3	Safety Evaluation	replaced 10 CFR 100 with 10 CFR 50.67	6.3-26
6.4	SAFETY DESIGN BASIS NINE	added (event) to SAFETY DESIGN BASIS NINE description	6.4-2
6.4.4	SAFETY EVALUATION SEVEN	revised to address inhalation, immersion, and transit	6.4-6
6.5.1.3	Safety Evaluation	replaced 10 CFR 100 with 10 CFR 50.67	6.5-3
6.5.2.1.1	Safety Design Bases	replaced 10 CFR 100 with 10 CFR 50.67	6.5-4
6.5.2.3	Safety Evaluation	change to reference Section 6.5A-4, change spray removal coefficient from 10 to 20	6.5-6
6.5.2.3	Safety Evaluation	deleted "iodine"	6.5-7
6.5.2.3	Safety Evaluation	replaced DF's and spray removal coefficients, replaced 10 CFR 100 with 10 CFR 50.67, deleted "iodine"	6.5-8
6.5.3.1	Primary Containment	replaced 10 CFR 100 with 10 CFR 50.67	6.5-9
6.5.5	References	Added reference 4	6.5-10
Table 6.5-1	ESF Filtration Systems Input Parameters…	A.3 and 6.5A.44	Table
Table 6.5-2	Input Parameters and Results of Spray Removal Analysis	Updated input parameters based on NAI-1990-004. Additional work required for some values	Table
6.5A	Appendix 6.5A Removal Models for the Containment Spray System	Deleted "iodine" from title	6.5A-1
6.5A.1	Particulate Iodine Model (for EQ Dose Consequences)	Renamed title	6.5A-2
6.5A.2	Elemental iodine Model for EQ Dose Calculations	removed pointer to use in Chapter 15.6.5	6.5A-10
6.5A.3	Elemental Iodine Model for Offsite and Control Room Dose Calculations	Replaced entire section	6.5A-11
6.5A.4	PARTICULATE IODINE MODEL FOR OFFSITE AND CONTROL ROOM DOSE CALCULATIONS	New section, moved references to 6.5.A.5	6.5A-11
6.5.A5	References	Added references 14 and 15	6.5A-12
Chapter 7: Instrumentation and Controls			
7.1.1	Identification of Safety-Related Systems	added 10 CFR 50.67 and RG 1.183, as appropriate	7.1-1
Chapter 9			
9.1.4.1.1	Safety Design Bases (of the Fuel Handling System), SAFETY DESIGN BASIS SIX	Replaced 10 CFR 100 with 10 CFR 50.67 and RG 1.183	9.1-26
9.1.4.3	Safety Evaluation	Replaced 10 CFR 100 with 10 CFR 50.67 and RG 1.183	9.1-50
Table 9.1-3	DESIGN COMPARISION TO REGULATORY POSITIONS OF REGULATORY GUIDE 1.13 REVISION 1, DATED DECEMBER 1975, TITLED “SPENT FUEL STORAGE FACILITY DESIGN BASIS	Regulatory Guide 1.13, position 4 changed to point to RG 1.183 instead of RG 1.25 for assumptions for the inventory of radioactive materials available for leakage from the building	Table
9.4.1.2.1	General Description	Added reference to the model for control room dose analysis in Appendix 15A.	9.4-3
9.4.2	Fuel Building HVAC	added credit for emergency exhaust system	9.4-16
9.4.2.3	Safety Evaluation (of the Fuel Building HVAC)	replaced 10 CFR 100 with 10 CFR 50.67	9.4-24

FSAR Section	Title	Change Description	Page
Chapter 12			
12.3.2.2.6	Control Room Shielding Design	Replaced whole body with TEDE	12.3-12
12.3.3.2c	Design Criteria	Replaced 10 CFR 100 with 10 CFR 50.67	12.3-14
12.3.4.1.1.1	Safety Design Bases	replaced 10 CFR 100 with 10 CFR 50.67	12.3-18
Chapter 15			
TOC	Table of Contents	Updated Section 15A	15.0-viii
TOC	Table of Contents	Added Sections 15A.5, 15A.6 and Appendix 15B	15.0-ix
List of Tables	Table of Contents	deleted RG 1.4, 1.25 (Table 15.6-7, 15.7-2)	15.0-xii
List of Tables	Table of Contents	Renamed Table 15A-3, Added Tables 15A-6, -7 and -8 Added Conformance Tables 15B-1 through 15B-7 for RG 1.183	15.0-xiii
List of Figures	List of Figures	Replaced figures	15.0-xxv, -xxvi, -xxxiv, -xxxv
List of Figures	List of Figures	Figures 15A-1 and 15A-2 to be replaced	15.0-xxxvii
15.0.1.4	Condition IV - Limiting Faults	Replaced 10 CFR 100 with 10 CFR 50.67	15.0-5
15.0.9	Fission Product Inventories	Replaced with data from NAI-1990-002	15.0-12, -13
15.0.11.8	RETRAN	added discussion of RETRAN-3D	15.0-17
15.0.14	References	Added reference 22 to SCALE; added Reference 23 to RETRAN-3D	15.0-22
Table 15.0-2(Sheet 6)	Table 15.0-2	corrected typo	Table
Table 15.0-7	Single Failures Assumed in Accident Analyses	Added SGTR information	Table
MSLB			
15.1.5.1	Steam system piping failure	replaced 10 CFR 100 with 10 CFR 50.67	15.1-14
15.1.5.3.1.2	Assumptions and Conditions	Added pointer to Tables 15B-1 and 15B-4	15.1-21
15.1.5.3.1.2	Assumptions and Conditions	Added reference to Tables 15B-1 and 15B-4 for RG 1.183, revised timing	15.1-22
15.1.5.3.1.2 and 15.1.5.3.1.3	Assumptions and Conditions	added clarification text	15.1-23
15.1.5.3.2	Identification of Uncertainties and Conservatisms in the Analysis	revised paragraph a, added paragraph d.	15.1-24
15.1.5.3.3.2	Steam system piping failure	replaced 10 CFR 100 with 10 CFR 50.67, duration of accident	15.1-25
Table 15.1-3	Parameters used in evaluating radiological consequences of MSLB	revised input parameters	Table
Table 15.1-4	Radiological Consequences of MSLB	revised values	Table
Loss of Offiste Power			
15.2.6.3.1.1	Physical Model for LOOP Radiological Consequences	replace 8 with 7.275 hours	15.2-13
15.2.6.3.1.1	Physical Model for LOOP Radiological Consequences	revised assumptions and conditons	15.2-14
15.2.6.3.2	Identification of Uncertainties in, and Conservative of, the Analysis	revised values	15.2-15
15.2.6.3.3.2	Loss of NonEmergency AC Power Radiological Consequences	replaced 10 CFR 100 with 10 CFR 50.67	15.2-16
Table 15.2-2	Parameters Used In Evaluating Radiological Consequences of Loss of Nonemergency AC Power	revised parameters	Table
Table 15.2-3	Radiological Consequences	revised doses	Table

FSAR Section	Title	Change Description	Page
Locked Rotor			
15.3.3.3	Radiological Consequences	revised assumptions, models used, conservatisms and results	15.3-9, -10, -11, -12
Table 15.3-3	Parameters Used in Evaluating the Rsdiological Consequences of a Locked Rotor Event	revised parameters	Table
Table 15.3-4	Radiological Consequences of a Locked Rotor Event	revised doses	Table
Rod Ejection			
15.4.8.3	Radiological Consequences	revised assumptions, models used, conservatisms and results	15.4-43 through 15.4-47
Table 15.4-3	Parameters Used in Evaluating the RCCA Ejection Accident	revised parameters	sheet 1 and 2
Table 15.4-4	Radiological Consequences of a Rod-Ejection Accident	revised doses	Table
Letdown Line			
15.6.2.1.1.2	Assumptions and Conditions (Radiological Consequences of Letdown Line Break)	revised assumptions and conditons	15.6-4
15.6.2.1.1.3	Mathematical Models Uned in the Analysis	added in control room	15.6-5
15.6.2.1.3.2	Dose to Receptor for Letdown Line	renamed title, replaced 10 CFR 100 with 10 CFR 50.67	15.6-6
Steam Generator Tube Failure			
15.6.3.1.2	Analysis of Effects and Consequences	revised description	15.6-10, -11, -12, -13
15.6.3.1.3	Radiological Consequences	revised description	15.6-14 through -19
15.6.3.2.1	Identification of Causes and Accident Description	Revised discussion of ASD conservatism	15.6-20
15.6.3.2.1, 15.6.3.2.2	Identification of Causes and Accident Description	Revised length of time to RHR cut-in, revised feedwater temperature	15.6-21
15.6.3.2.2	Analysis of Effects and Consequences	Revised assumptions	15.6-22
15.6.3.2.2	Analysis of Effects and Consequences	Revised AFW flow to SGs, revised assumptions	15.6-23, -24, -25
15.6.3.2.2 results	Analysis of Effects and Consequences, Results	revised operator timing	15.6-27
15.6.3.2.3	Radiological Consequences	revised results	15.6-28, -29, -30
15.6.3.3.2	Conclusions	revised description	15.6-31
LOCA			
15.6.5.4	Radiological Consequences	revised description	15.6-44, -45, -46
15.6.5.4.1.2	Radioactive Releases Due to Leakage from ECCS and Containment Spray Recirculation Lines	added "air" to RWST description, revised RWST leakage discussion and release percentages	15.6-47
15.6.5.4.2	Identification of Uncertainties and Conservatisms in the Analysis	revised conservatisms	15.6-48, -49
15.6.5.4.3.2 and 15.6.5.4.3.3	Doses at EAB and LPZ	replaced total body and thyroid with TEDE, revised 10 CFR 100 (and GDC-19) with 10 CFR 50.67	15.6-50
Chapter 15.6 references			
15.6.7	References	Added FGR 11 and 12	15.6-53

FSAR Section	Title	Change Description	Page
Chapter 15.6 Tables			
Table 15.6-1 (Sheet 2)	Time Sequence of Events for Incidents which Result in a Decrease in Reactor Coolant Inventory	Revised timing	SGTR with overfill
Table 15.6-2	Parameters used In Evaluating the Radiological Consequence of the CVCS Letdown line Rupture Outside of Containment	revised parameters	Table
Table 15.6-3	Radiological Consequences of a CVCS Letodown Line Break Outside Containment	revised doses	Table
Table 15.6-4	Parameters used in Evaluating the Radiological Consequences of a SGTR	revised input papameters	Table
Table 15.6-5	Radiological Consequences of a Steam Generator Tube Rupture with Stuck-Open Atmospheric Steam Dump Valve	revised dose consequence values	Table
Table 15.6-5A	Radiological Consequences of a Steam Generator Tube Rupture with Overfill	revised dose consequence values	Table
Table 15.6-6	Parameters used in Evaluating the Radiological Consequences of a Loss-of-Coolant Accident	revised inputs	Table
Table 15.6.7	Design Comparison to RG 1.4	Deleted table	Table
Table 15.6-8	Radiological Consequences of a LOCA	Replaced values	Table
15.7 Radioactive Release from A Subsystem or Component			
15.7.1.5.1.3	Mathematical Models Used in the Analysis	Revised referenced section numbers	15.7-3
15.7.2.5.1.3	Mathematical Models Used in the Analysis	Revised referenced section numbers	15.7-7
Fuel Handling Accident			
15.7.4.4	Barrier Performance	added pointer to Section 15.7.4.5.1.2, removed factor of 100	15.7-9
15.7.4.5.1.2	Assumptions and Conditions	revised description	15.7-12, -13
15.7.4.5.2	Identification fo Uncertainties and Conservatisms in Analysis	revised to reference RG 1.183 guidance instead of description	15.7-14
15.7.4.5.2.2	Doses to Receptor at the EAB, LPZ and Control Room	revised to reference 50.67 and RG 1.183, added insert 15.7.5	15.7-15
Table 15.7-2	Design Comparison to RG 1.25	Deleted table	Table
Table 15.7-7	Parameters Used in Evalauting the Radiological Consequences of a Fuel-Handling Accident	Revised parameters	Table
Table 15.7-8	Radiological Consequences of a Fuel Handling Accident	Revised values	Table
Appendix 15A			
15A	General Acccident Parameters	entire section revision	15A-1 through 15A-14
Table 15A-1	Parameters used in Accident Analysis	revised	Table
Table 15A-2	Limiting short term Atmospheric Dispersion Factors	replaced	Table
Table 15A-3	Core Inventory (Ci)	Replaced and Renamed	Table
Table 15A-4	Dose Conversion Factors Used in Accident Analysis	Added DCFs for events reanalyzed with AST	Table
Table 15A-5	Initial Radioactivity for Accidents that use the Primary-to-Secondary Leakage Relase Pathway	added nuclides	Sheets 1-5

1.1.6 SCHEDULE FOR FUEL LOADING AND OPERATION

On June 11, 1984 Union Electric received a low-power (5%) license to operate the Callaway Plant with initial criticality being achieved on October 2, 1984. The full-power license was issued to Union Electric on October 18, 1984 and commercial operation began on April 9, 1985.

1.1.7 DESIGN BASES

As used within this FSAR, the design bases are a list of requirements that the system must meet in order to:

a. Perform directly a specified safety or power generation function including support of another function (e.g., provide cooling water flow for other components, maintain a given compartment temperature).

b. Comply with a regulatory or statutory requirement or guideline (e.g., a jurisdictional building code).

c. Meet a specific operator interface, startup, or specific testing requirement.

d. Meet a design classification or code requirement (e.g., be designed to withstand the safe shutdown earthquake). Items implicit in contemporary design practices (e.g., use of the English system of weights and measures or the exercise of good engineering practice) are not specified as design bases.

Safety design bases are engineering objectives which must be met by safety-related structures, systems, or components. Safety-related items are defined as those plant features necessary to ensure the following:

a. The integrity of the reactor coolant pressure boundary

b. The capability to shut down the reactor and maintain it in a safe shutdown condition

c. The capability to prevent or mitigate the consequences of accidents that could potentially result in offsite and control room operator exposures approaching the guideline exposures of 10 CFR ~~100~~ 50.67 and Regulatory Guide 1.183, or 10 CFR 100, as applicable.

Items which are associated with safety-related equipment, but which in themselves are not absolutely essential to the safety function of the equipment, are not considered safety-related. |

Power generation design bases support, either directly or indirectly, the major electrical power generation function of the station. Examples of power generation design bases

TABLE 1.1-1 ACRONYMS USED IN THE FSAR

AC	Alternating Current
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
A/E	Architect/Engineer
AFAS	Auxiliary Feedwater Actuation System
AFS	Auxiliary Feedwater System
AISC	American Institute of Steel Construction
ALARA	As Low as Reasonably Achievable
ANSI	American National Standards Institute
APRM	Average Power Range Monitor
ARM	Area Radiation Monitor
ARW	Chemical Waste
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transients Without Scram
AVT	All Volatile Treatment
AWS	American Welding Society
BOP	Balance of Plant
B&PVC	Boiler and Pressure Vessel Codes
BRS	Boron Recycle System
BTP	Branch Technical Position
CAS	Compressed Air System
CCS	Condensate Cleanup System
CCWS	Component Cooling Water System
CDS	Condensate Demineralizer System
CeCWS	Central Chilled Water System
CFR	Code of Federal Regulations
CFS	Condensate and Feedwater System
CGCS	Combustible Gas Control System
CHC	Cask Handling Crane
CHF	Critical Heat Flux
CIS	Containment Isolation Signal
CICWS	Closed Cooling Water System
CLP	Cask Loading Pit
CM	Center of Mass
CMAA	Crane Manufacturing Association of America

CEDE Committed Effective Dose Equivalent

TABLE 1.1-1 (Sheet 2)

CP	Construction Permit
CPR	Critical Power Ratio
CPIS	Containment Purge Isolation System/Signal
CR	Center of Rigidity
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CRDM	Control Rod Drive Mechanism
CREA	Control Rod Ejection Accident
CRVIS	Control Room Ventilation Isolation System/Signal
CRW	Tritiated Waste
CSD	Cold Shutdown
CST	Condensate Storage Tank
EDE	Effective Dose Equivalent for cloudshine
CSTS	Condensate Storage and Transfer System
CtCS	Containment Cooling System
CVCS	Chemical and Volume Control System
CWP	Cask Washdown Pit
CWS	Circulating Water System
DBA	Design Basis Accident
DBE	Design Basis Event
DC	Direct Current
DCSS	Dry Cask Storage System
DEHC	Digital Electrohydraulic Control
DEPSG	Double Ended Pump Suction Guillotine
DG	Diesel Generator
DGB	Diesel Generator Building
DoWS	Domestic Water System
DNB	Departure From Nucleate Boiling
DNBR	Departure From Nucleate Boiling Ratio
DRW	Potentially Radioactive Nontritiated Waste
DWMS	Demineralized Water Make-up System
DWST	Demineralized Water Storage Tank
DWSTS	Demineralized Water Storage and Transfer System
DWT	Dead Weight Test
ECCS	Emergency Core Cooling System
EHC	Electrohydraulic Control
EOL	End of Life
EDECAIES	Emergency Diesel Engine Combustion Air Intake and Exhaust System
EDECWS	Emergency Diesel Engine Cooling Water System
EDEFSTS	Emergency Diesel Engine Fuel Oil Storage and Transfer System

TABLE 1.1-1 (Sheet 5)

PSAR	Preliminary Safety Analysis Report
PSS	Process Sampling System
PWR	Pressurized Water Reactor
RCP	Reactor Coolant Pumps
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RMWCS	Reactor Makeup Water Control System
RMWS	Reactor Makeup Water System
RMWST	Reactor Makeup Water Storage Tank
RO	Reactor Operator
RPV	Reactor Pressure Vessel
RRS	Required Response Spectrum
RSG	Replacement Steam Generator
RWB	Radwaste Building
RWST	Refueling Water Storage Tank
SACF	Single Active Component Failure
SAR	Safety Analysis Report
SFSF	Spent Fuel Storage Facility
SGB	Steam Generator Blowdown
SGBIS	Steam Generator Blowdown Isolation System/Signal
SGBS	Steam Generator Blowdown System
SIS	Safety Injection Signal
SIT	Structural Integrity Test
SLWS	Secondary Liquid Waste System
SMA	Strong Motion Accelerometer
SNUPPS	Standard Nuclear Unit Power Plant System
SRO	Senior Reactor Operator
SRP	Standard Review Plan
SRS	Solid Radwaste System
SRSS	Square Root of the Sum of the Squares
SRW	Detergent Waste
SSE	Safe Shutdown Earthquake
SWS	Service Water System
TBS	Turbine Bypass System
TG	Turbine Generator
TGSS	Turbine Gland Sealing System
TRS	Test Response Spectrum
UHS	Ultimate Heat Sink

TEDE Total Effective Dose Equivalent

TABLE 1.3-4 (Sheet 14)

Regulation (10 CFR)	Compliance
50.56	This regulation provides that the Commission will, in the absence of good cause shown to the contrary, issue an operating license upon completion of the construction of a facility in compliance with the terms and conditions of the construction permit. This imposes no independent obligations on the applicant.
50.57(a)	This regulation requires the Commission to make certain findings prior to the issuance of an operating license.
50.57(b)	The license, as issued, will contain appropriate conditions to ensure that items of contruction or modification are completed on a schedule acceptable to the Commission.
50.57(c)	This regulation provides for a low-power testing license.
50.58	This regulation provides for the review and report of the Advisory Committee on Reactor Safeguards.
50.59	This regulation provides for the licensing of certain changes, tests, and experiments at a licensed facility. Technical Specifications and procedures provide implementation of this regulation.
50.70	The Commission has assigned resident inspectors to the SNUPPS plants and space will be provided in conformance with 50.70(b)(1) through (3).
50.71	Records are and will be maintained in accordance with the requirements of sections (a) through (e) of this regulation and the license.
50.80	This regulation provides that licenses may not be transferred without NRC consent. No application for transfer has been made by the SNUPPS utilities.
50.81	This regulation permits the creation of mortgages, pledges, and liens on licensed facilities, subject to certain provisions. The regulation prohibits secured creditors from violating the Atomic Energy Act and the Commission's regulations.
50.82	This regulation provides for the termination of licenses. It does not apply to SNUPPS' because no termination of licenses has been requested.
50.90	This regulation governs applications for amendments to licenses. Future request for license amendments will be made in accordance with these requirements.
50.91	This regulation provides guidance to the NRC in issuing license amendments.

50.67 The FSAR accident analyses, in particular those in Chapter 6.0 and 15.0, demonstrate that offsite and control room doses resulting from postulated accidents would not exceed the criteria in this section of the regulation.

TABLE 1.3-4 (Sheet 26)

Regulation (10 CFR)	Compliance
100.3	This regulation is explanatory and does not impose independent obligations on licensees.
100.10	The factors listed related to both the unit design and the site have been provided in the application. Site specifics, including seismology, meteorology, geology, and hydrology, are presented in Chapter 2.0 of the FSAR. The exclusion area, low population zone, and population center distance are provided and described. The FSAR also describes the characteristics of reactor design and operation.
100.11	Exclusion areas have been established, as described in each FSAR Site Addendum Section 2.1. The low population zone for each unit has been established in accordance with this requirement. ~~The FSAR accident analyses, particularly those in Chapters 6.0 and 15.0, demonstrate that offsite doses resulting from postulated accidents would not exceed the criteria in this section of the regulation.~~
Appendix A	Appendix A to 10 CFR Part 100 provides seismic and geologic siting criteria for nuclear power plants. Site suitability was determined at the construction permit stage.

2.3 METEOROLOGY

2.3.4 SHORT-TERM (ACCIDENT) DIFFUSION ESTIMATES

2.3.4.1 Objective

The objective of this section is to provide short-term atmospheric dispersion factors (χ/Qs) for the postulated accident analyses presented in Chapter 15.0.

2.3.4.2 Calculations

Section 2.3.4.4

2.3.4.2.1 Site Boundary and LPZ

The short-term atmospheric dispersion factors (χ/Qs) are based on onsite meteorological data for the Callaway Plant site. The diffusion equations and assumptions used in the calculations were those outlined in NRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Assessment at Nuclear Power Plants." Table 2.3-1 lists the limiting χ/Qs for the Callaway site. The detailed procedures used in the calculations are given in ~~Section 2.3.4.2~~ of the Site Addendum.

2.3.4.2.2 Control Room Intake

2.3.4.2.2.1 Waste Gas Decay Tank Rupture and Liquid Waste

The basic model employed for the distribution of relative concentrations (χ/Qs) within a building wake at the Callaway control room intakes following an accident is given by Reference 1 to be:

$$\chi / Q = \frac{K_c}{AV} \qquad (1)$$

Where A = reference cross-sectional building area, m^2

V = reference wind speed, m/sec

K_C = nondimensional concentration coefficient

K_C is a function of nondimensional space coordinates x/L, y/L, and z/L, building configuration, wind direction, and source configuration. The K_C field for a given building configuration, source configuration, and wind configuration is considered to be invariant. Accordingly, K_C values determined by wind tunnel tests with a model structure are expected to be the same as those that would be obtained with a geometrically similar building in the full-scale atmosphere in the same wind direction, with a similar leak. The Callaway Plant contiguous building arrangement is shown in Figure 2.3-1. The K_C data used in the analysis for low level release are presented in Figure 2.3-2 and were derived from two sets of tests. One used rectangular prisms (Ref. 2), the other used a model of the EBR-II complex (Ref. 1). Both tests were described and portions of the data presented in Reference 3. The K_C data for the unit vent release from the top of the containment were extracted from Figure 10 of Reference 1 and are presented in Table 2.3-2. The value of A used in conjunction with K_C in Figure 2.3-2 and Table 2.3-2 is the Callaway Plant equivalent of the EBR-II area, A = 1.12 D^2 = 2280 m^2 with the diameter of the reactor D = 45.1 m.

The value of V used in conjunction with Figure 2.3-2 is the mean velocity of the approach flow at an elevation corresponding to the anemometer elevation of the EBR-II model tests. Reference 3 reports this elevation to be 62 feet or 0.77D above the top of the dome. The Callaway Plant equivalent height becomes 63.4 + 0.77 x 45.1 = 98.1m above ground. The V values were obtained by extrapolating wind speeds at anemometer elevations equivalent to 98.1 meters by the power law.

$$V = u_1(98.1/z_1)^n \qquad \cancel{(2)}$$

Where u_1 = mean speed at elevation z_1, m/sec

z_1 = anemometer elevation at a given site, m

n = atmospheric stability exponent

Values of n were arbitrarily assumed for the various stability classes as follows:

Pasquill Stability Class	A	B	C	D	E	F	G
n	0.20	0.25	0.29	0.33	0.40	0.50	0.60

Insert 2.3.4.2.2.2

A cumulative frequency distribution was constructed for the χ/Q values calculated by equations 1 and 2 above, using 3 years combined onsite meteorological data. The corresponding highest 5 percent, 10 percent, 20 percent, and 40 percent χ/Q values are given in Table 2.3-3.

2.3.5 REFERENCES

~~1. Halitsky, J., Golden, J., Halpern, P., (1963): "Wind Tunnel Tests of Gas Diffusion From a Leak in the Shell of a Nuclear Power Reactor and from a Nearby Stack," N. Y. University Department of Met. & Ocean, GSL Rep. 63-2 under USWB Contract Cwb-10321~~

TABLE 2.3-1 LIMITING ATMOSPHERIC DISPERSION FACTOR, ξ/Q(sec/m^3)

	ξ/Q
<u>Site Boundary</u>	
0-2 hr.	~~2.0E-4~~ 2.05E-04
<u>Low Population Zone</u>	
0-8 hr.	~~2.6E-5~~ 3.42E-05
8-24 hr.	~~1.7E-5~~ 2.42E-05
24-96 hr.	~~7.2E-6~~ 1.13E-05
96-720 hr.	~~2.0E-6~~ 3.83E-06

3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

This chapter identifies, describes, and discusses the principal architectural and engineering design features of those structures, components, equipment, and systems which are necessary to assure:

or control room

a. The integrity of the reactor coolant pressure boundary

b. The capability to shut down the reactor and maintain it in a safe shutdown condition

c. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline values of 10 CFR ~~100.~~

50.67, Regulatory Guide 1.183, or 10 CFR 100, as appropriate.

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

This section briefly discusses the extent to which the design criteria for SNUPPS plant structures, systems, and components important to safety comply with Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, "General Design Criteria for Nuclear Power Plants" (GDC). As presented in this section, each criterion is first quoted and then discussed in enough detail to demonstrate SNUPPS compliance with each criterion. For some criteria, additional information may be required for a complete discussion. In such cases, detailed evaluations of compliance with the various general design criteria are incorporated in more appropriate FSAR sections, but are located by reference.

3.1.1 DEFINITION OF SINGLE FAILURE

The single failure criterion is a constraint used in the design of safety systems to improve the reliability of the system to perform its safety function following a design-basis event or design occurrence.

A single failure means an occurrence which results in the loss of the capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electrical systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming that passive components function properly) nor (2) a single failure of a passive component (assuming that active components function properly) results in a loss of the capability of the system to perform its safety functions.

Single failures are random occurrences imposed upon safety systems that are required to respond to a design basis event. They are postulated despite the fact that the systems were designed to remain functional under the adverse condition imposed by the accident. No mechanism for the cause of the single failure need be postulated. Single total effective dose equivalent (TEDE)

CRITERION 19 - CONTROL ROOM

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

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

total effective dose equivalent (TEDE)

DISCUSSION

A separate control room is provided for the control of each unit from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain in a safe manner under accident conditions, including LOCAs. Operator action outside of the control room to mitigate the consequences of an accident is permitted. The control room and its post-accident ventilation systems are designed to satisfy seismic Category I requirements, as discussed in Chapter 3.0. Adequate concrete shielding and radiation protection are provided against direct gamma radiation and inhalation doses postulated to result from a ~~TID-14844~~ release of fission products inside the containment structure. The shielding and the ~~control room~~ standby air-conditioning system allow access to and occupancy of the control rooms under accident conditions without personnel receiving radiation exposures in excess of 5 rem ~~whole body or its equivalent to any part of the body~~ for the duration of the accident. Refer to Chapter 15.0. Fission product removal is provided in the control room recirculation equipment to remove iodine and particulate matter, thereby minimizing the ~~thyroid~~ dose which could result from the accident. The control room habitability features are described in Chapter 6.0.

In the event that the operators are forced to abandon the control room, panel-mounted local instrumentation and controls are provided to achieve and maintain the plant in the hot shutdown condition (see Chapter 7.0). The capability for bringing the plant to a cold shutdown is also provided outside the control room through the use of local controls.

3.1.5 PROTECTION AND REACTIVITY CONTROL SYSTEMS

CRITERION 20 - PROTECTION SYSTEM FUNCTIONS

"The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure."

DISCUSSION

The containment spray system serves to remove radioiodine and other airborne particulate fission products from the containment atmosphere following a LOCA. The system consists of two independent systems, each supplied from separate electrical power busses, as described in Chapter 8.0. Either subsystem alone can provide the fission product removal capacity for which credit is taken in Chapter 15.0, in compliance with Regulatory Guide ~~1.4~~. 1.183

The generation of hydrogen in the containment under post-accident conditions has been evaluated, using the assumptions of Regulatory Guide 1.7 (see Chapter 6.0). A post-accident hydrogen recombiner system is provided with redundancy of vital components so that a single failure does not prevent timely operation of the system. This system is described in Section 6.2.5. A hydrogen purge system is provided as a backup. No single failure causes both subsystems to fail to operate.

CRITERION 42 - INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

"The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems."

DISCUSSION

The containment atmosphere cleanup systems are designed and located so that they can be inspected periodically, as required. The essential equipment of the containment spray system is outside the containment, except for risers, distribution header piping, and spray nozzles in the containment. The hydrogen purge and monitoring components of the hydrogen control system are located outside the containment. The equipment outside the containment may be inspected during normal power operation. Components of the containment spray system and the hydrogen control system located inside the containment can be inspected during refueling shutdowns. See Chapter 6.0 for details on the containment spray system and details of the hydrogen control system.

CRITERION 43 - TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

"The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Certain structures, components, and systems of the nuclear plant are important to safety because they:

a. Assure the integrity of the reactor coolant pressure boundary.

b. Assure the capability to shut down the reactor and maintain it in a safe condition.

c. Assure the capability to prevent or mitigate the consequences of accidents which could result in potential offsite or Control Room exposures comparable to the guideline exposures of 10 CFR ~~100~~ 50.67, Regulatory Guide 1.183, or 10 CFR 100, as appropriate.

d. Contain or may contain radioactive material.

The purpose of this
according to the imp
the facility can be operated without undue risk to the health and safety of the public. Table 3.2-1 delineates each of the items in the plant which fall under the above-mentioned categories and the respective associated classification that the NRC, ANS, and industrial codes committees have developed. Each of the classification categories in Table 3.2-1 is addressed in the following sections.

For identification of system and subsystem boundaries, Table 3.2-1 is supplemented (i.e., referenced to applicable figures) by piping and instrument diagrams which have been marked to clearly show the limits of the seismic Category I and various quality group classifications on a system. The legend for the piping and instrument diagrams is provided in Figure 1.1-1.

Classification of power supplies, instrumentation and controls, valve operators, supports, hangers, and restraints is not delineated in Table 3.2-1 because of the extensive listing required. Generic listings for piping/valves and ductwork/ dampers are included for completeness, since for some systems these are the only items serving a safety function. Containment penetrations are not included in these generic listings as there is a separate subheading for containment penetrations. The classification for all of these unlisted and generically listed items is consistent with the boundaries shown on the piping and instrumentation drawings. A listing of the piping and instrumentation drawings and their associated FSAR figures is found in Table 1.7-2 and in Section 1.7 of each Site Addendum.

3.6.2.3.2 Dynamic Analysis Methods to Verify Integrity and Operability for the Reactor Coolant Loop

3.6.2.3.2.1 General

A LOCA is assumed to occur for a branch line break down to the restraint of the second normally open automatic isolation valve (Case II in Figure 3.6-2) on outgoing* and down to and including the second check valve (Case III in Figure 3.6-2) on incoming lines normally with flow. A pipe break beyond the restraint or second check valve will not result in an uncontrolled loss of reactor coolant if either of the two valves in the line closes.

Accordingly, both of the automatic isolation valves are suitably protected and restrained as close to the valves as possible so that a pipe break beyond the restraint will not jeopardize the integrity and operability of the valves. Further, periodic testing capability of the valves to perform their intended function is essential. This criterion takes credit for only one of the two valves performing its intended function. For normally closed isolation or incoming check valves (Cases I and IV in Figure 3.6-2), a LOCA is assumed to occur for pipe breaks on the reactor side of the valve.

Branch lines connected to the reactor coolant loop (RCL) are defined as "large" for the purpose of this criteria and as having an inside diameter greater than 4 inches up to the largest connecting line, generally the pressurizer surge line. Rupture of these lines results in a rapid blowdown from the RCL, and protection is basically provided by the accumulators and the low head safety injection pumps (residual heat removal pumps).

Branch lines connected to the RCL are defined as "small" if they have an inside diameter equal to or less than 4 inches. This size is such that emergency core cooling system analyses, using realistic assumptions, show that no clad damage is expected for a break area of up to 12.5 square inches, corresponding to 4-inch inside diameter piping.

Engineered safety features are provided for core cooling and boration, pressure reduction, and activity confinement in the event of a LOCA or steam or feedwater line break accident to ensure that the public is protected in accordance with 10 CFR ~~100 guidelines~~. These safety systems have been designed to provide protection for a reactor coolant system pipe rupture of a size up to and including a double-ended severance of a reactor coolant loop.

In order to assure the continued integrity of the vital components and the engineered safety systems, consideration is given to the consequential effects of the pipe break itself to the extent that:

50.67, Regulatory Guide 1.183, or 10 CFR 100 guidelines, as appropriate.

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* It is assumed that motion of the unsupported line containing the isolation valves could cause failure of the operators of both valves to function.

3.7(N) SEISMIC DESIGN

For the OBE loading condition, the nuclear steam supply system is designed to be capable of continued safe operation. The design for the SSE is intended to ensure:

a. That the integrity of the reactor coolant pressure boundary is not compromised;

b. That the capability to shut down the reactor and maintain it in a safe condition is not compromised; and

c. That the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR ~~100~~ is not compromised.

It is necessary to ensure that required critical structures and components do not lose their capability to perform their safety function. Not all critical components have the same functional safety requirements. For example, a safety injection pump must retain its capability to function normally during the SSE. Therefore, the deformation in the pump must be restricted to appropriate limits in order to ensure its ability to function. On the other hand, many components can experience significant permanent deformation without loss of function. Piping and vessels are examples of the latter where the principal

50.67, Regulatory Guide 1.183, or 10 CFR 100, as appropriate.

The seismic requirements for safety-related instrumentation and electrical equipment are covered in Sections 3.10(N) and (B). The safety class definitions, classification lists, operating condition categories, and the methods used for seismic qualification of mechanical equipment are given in Section 3.2.

3.7(N).1 SEISMIC INPUT

3.7(N).1.1 Design Response Spectra

Refer to Section 3.7(B).1.1.

3.7(N).1.2 Design Time History

Refer to Section 3.7(B).1.2.

3.7(N).1.3 Critical Damping Values

The damping values given in Table 3.7(N)-1 are used for the systems analysis of Westinghouse equipment and for the component analysis of the Integrated Head Assembly (IHA) and replaced steam generators (SGs). These are consistent with the damping values recommended in Regulatory Guide 1.61, Rev. 0, except in the case of the primary coolant loop system components and large piping (excluding reactor anchorage surfaces of the buttress are normal to the tangent line of the anchored hoop tendons. Details are shown in Figure 3.8-30.

The concrete shell around the equipment hatch opening is thickened by the method shown in Figures 3.8-31 and 3.8-32.

3.8.1.1.5 Special Reinforcing Requirements

Special reinforcing is required in such areas as the major penetrations. Refer to Figures 3.8-31 through 3.8-35 for typical details in these areas.

3.8.1.2 Applicable Codes, Standards, and Specifications

The following codes, regulations, standards, and specifications are utilized in the reactor building design.

3.8.1.2.1 Regulations

a. 10 CFR 50, "Licensing of Production and Utilization Facilities"

b. 10 CFR 100, "Reactor Site Criteria"

c. 10 CFR 50.67, "Accident Source Term"

3.8.1.2.2 Codes

a. American Concrete Institute, Building Code Requirements for Reinforced Concrete (ACI-318-71)

b. American Institute of Steel Construction (AISC), Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, 7th Edition, adopted February 12, 1969, and Supplement Numbers 1, 2, and 3 (See FSAR Table 3.2-1, Note 19)

c. ASME Boiler and Pressure Vessel Code - 1974 Edition or later

Section II - Material Specifications

Section III, Division 1 - Nuclear Power Plant Components

Section V - Nondestructive Examination

Section VIII - Pressure Vessels

Section IX - Welding and Brazing Qualifications

d. American Welding Society, Structural Welding Code (AWS D1.1-75)(See FSAR Table 3.2-1, Note 19)

3.11(B).1.1.2 Safety-Related System Listing

Safety-related systems are those plant systems necessary to ensure:

a. The integrity of the reactor coolant pressure boundary.

b. The capability to shut down the reactor and maintain it in a safely shutdown condition.

c. The capability to prevent or mitigate the consequences of accidents which could result in offsite exposures comparable to the guidelines of 10 CFR ~~100~~

Systems that perform these type functions are those systems required to achieve or support emergency reactor shutdown, containment isolation, reactor core cooling, containment heat removal, core residual heat removal, and prevention of significant release of radioac 50.67, Regulatory Guide 1.183, or 10 CFR 100, as appropriate. n or support these functions is contained in the Callaway Equipment List (CEL). The specific safety function of each system is described in FSAR system description sections and in the CEL database.

Class 1E powered I&C devices are included in the system that they serve (e.g., EG-FT-0108 is a flow transmitter in the component cooling water system [EG]). The I&C devices can be divided into two categories, NSSS and BOP supplied. Each type can be identified in the fourth column of Table 3.11(B)-3. The BOP supplied devices that are purchased by the Bechtel I&C Group have a specification number that begins with the letter J (e.g., J-301 for EG-FT-0108). The NSSS-supplied devices are identified in the fourth column by the respective Westinghouse EQDP number (e.g., ESE-4).

3.11(B).1.2 <u>Plant Environments</u>

3.11(B).1.2.1 Normal Environments

<u>Pressure, Temperature, Humidity, and Radiation</u>

Normal operating environmental conditions are defined as conditions existing during routine plant operations. These environmental conditions, as listed in Table 3.11(B)-1, represent the normal maximum and minimum conditions expected during routine plant operations.

<u>Dust</u>

In the NUREG-0588 review, dust was considered and was determined to be an insignificant factor in equipment qualification because outside air sources and ventilation units are typically equipped with filters which remove airborne dust. Also concrete coating, plant housekeeping, dust seals, and equipment maintenance requirements provide assurance that dust will not degrade equipment performance.

3.11(B).1.2.2 Accident Environments - Inside Containment

Accident environmental conditions are defined as those deviating from the normal operating environmental conditions. These conditions are specified in Table 3.11(B)-2.

In the NUREG-0588 review, Callaway LOCA/HELB/MSLB pressure, temperature, humidity, radiation, chemical spray, and submergence environmental conditions were evaluated. Where required, plant-unique environmental conditions were developed using the Category I criteria of NUREG-0588. The development of these conditions is described below. The post-accident parameters used in the equipment review are provided in summary form in Table 3.11(B)-2 and as used in the review, in Figures 3.11(B)-1 through 84. HELB P/T curves are also located in Reference 24.

<u>Radiation</u>

Using the guidance of NUREG-0588, post-LOCA radiation environments were determined in all areas of the containment. The original fission product release data used in this analysis were obtained from Westinghouse. The isotopic inventory provided by Westinghouse was for an equilibrium cycle Callaway core. The data were calculated at the end of cycle life and, therefore, represent maximums suitable for post-accident evaluations. This source term is referred to as the licensing basis EQ source term, ap have seen changes in fuel type (fr GE+), power level (from 3425 MWt to s discussed in Section 4.2.1). The do ed by 5% to account for these effects. In addition, the airborne gamma doses were increased by another 3% to account for the replacement of the active spray additive system with a passive system of baskets adjacent to the containment recirculation sumps containing trisodium phosphate. The following discussion refers to the initial calculations performed with the licensing basis EQ source term and a 50% cesium release fraction.

It is noted that the offsite and control room doses discussed in Section 15.6.5 were calculated using the elemental iodine spray removal model discussed in Section 6.5A.3.

The accident scenario assumed that a LOCA event occurred causing core damage. The entire source of 100 percent noble gas inventory, 50 percent of the core halogen inventory, 50 percent of the cesium, and 1 percent of the other solids was released to the containment. This release was conservatively assumed to occur at time zero. For the liquid source, 50 percent of the halogens, 50 percent of the cesium, and 1 percent of the remaining fission product solids were assumed to go directly to the sump and were diluted by the volume of the refueling water storage tank (RWST) and the liquid volume of the reactor coolant system. For the airborne source, 100 percent of the noble gases and 50 percent of core halogens were assumed to be released to the free volume of the containment. The simultaneous release of 50 percent of the halogens to the atmosphere and to the sump introduced additional conservatism.

Credit was taken for mechanistic removal of the airborne iodine via containment spray and plateout. The spray removal lambdas for elemental and particulate iodine ($25.7\ hr^{-1}$ + $0.73\ hr^{-1}$) were taken from the calculated values listed in Table 6.5-2. The plate-out removal lambda ($15.8\ hr^{-1}$) was calculated using methodology outlined in NUREG/CR-0009. The surface area available for plateout was assumed to be equivalent to the heat sink area used in the containment pressure analysis given in Table 6.2.1-4. In addition, two of the four hydrogen mixing fans were assumed to be operating, at 42,500 cfm each, to provide mixing between the sprayed (86 percent) and unsprayed (14 percent) regions of the containment. These removal processes were assumed to persist until the elemental and particulate iodine in the sprayed region were reduced by factors of 200 and 10,000, respectively.

These decontamination factors (DFs) were taken from Reference 22. The spray removal rate for elemental iodine was calculated in Section 6.5A.2 to be $25.7\ hr^{-1}$. This spray removal rate plus the plateout removal rate ($25.7\ hr^{-1}$ + $1.58\ hr^{-1}$) were assumed to be effective in the sprayed region until an elemental iodine decontamination factor (DF) of 200 was reached in the EQ dose calculations. Only the plateout removal rate was assumed to be effective in the unsprayed region until an elemental iodine DF of 2 was reached in the EQ dose calculations. The spray removal rate for particulate iodine was calculated to be $0.73\ hr^{-1}$ in Section 6.5A.1 and was assumed to be effective in the sprayed region until a particulate iodine DF of 10,000 was reached in the EQ dose calculations.

~~It is noted that the offsite and control room doses discussed in Section 15.6.5 were calculated using an elemental iodine spray removal rate of 10 hr-1 and a particulate iodine spray removal rate of 0.45 hr-1, until a DF of 28.7 was reached for elemental species and a DF of 50 was reached for particulate species. No plateout removal lambda was used in the Section 15.6.5 dose calculations since credit was taken for the instantaneous plateout of half of the iodines released to the containment atmosphere (i.e. 25% of the core iodines).~~

With the replacement of the spray additive system with trisodium phosphate baskets, the minimum equilibrium sump fluid pH is reduced to 7.1. This reduced pH results in a reduced spray partition coefficient (H, from Equation 6.5A-15 on page 6.5A-7) of 1100 per Reference 22. ~~Using Equation 6.5A-15, the resulting elemental iodine DF was calculated to be 28.7 for the analysis of offsite and control room doses discussed in Section 15.6.5.~~ Per Reference 23, the particulate iodine spray removal rate, calculated using Equation 6.5A-1 on page 6.5A-2, can conservatively be based on an assumed E/D of 10 per meter initially, changing to 1 per meter after a DF of 50. After the particulate iodine spray removal rate is reduced, there is no DF limit. ~~However, for simplicity and conservatism, removal was assumed to stop after a DF of 50 was reached in the analysis of offsite and control room doses.~~ With consideration given to these reduced DF values for elemental and particulate iodines, airborne gamma doses listed in Table and Control Room

During and Following an Accident." The response has been included in Appendix 7A. All Category I instruments are included in the NUREG-0588 program.

3.11(B).5.2 Equipment Operability

1.183

For the NUREG-0588 review, a post-DBA maximum operability requirement of 6 months (180 days) was utilized. Equipment was evaluated against this period for operability unless a shorter operability duration was justified. This value was selected as a conservative bounding time for termination of accident effects within the containment. The containment pressure-temperature analysis, as reflected in Figures 3.11(B)-3 and 6, indicates that containment conditions return to normal or below normal operating conditions within 30 days. It should also be noted that Regulatory Guide ~~1.4~~ provides criteria for evaluating the offsite radiological consequences of a LOCA event for a maximum of 30 days following the accident.

Margins of 1 hour or more for equipment with required operability times of less than 10 hours have generally been used for the Callaway equipment qualification review. However, margins of less than 1 hour have been used when adequate technical justification could be provided. Union Electric concurs with the AIF position on the 1-hour time margin, as stated in a letter to Mr. Harold Denton dated January 4, 1982, in that an arbitrary time margin of 1 hour appears inappropriate and should not be required when adequate technical justification for a shorter period exists.

3.11(B).5.3 Margins

The discussions in Section 3.11(B).1 show that post-accident environmental parameters were conservatively and uniquely determined using plant-specific data. Hence, the guideline generic techniques discussed in NUREG-0588 are not applicable.

The values for margin identified in Section 6.3.1.5 of IEEE-323-1974 were used as acceptance criteria during the NUREG-0588 review. The only regular exception to the IEEE-323-1974 margins was for radiation. As identified in Item 1.4 of NUREG-0588, additional margin need not be added to the radiation parameters if the methods identified in Appendix D of NUREG-0588 are utilized. The methods used to determine the Callaway radiation parameters are consistent with the Appendix D methodology. Hence, the radiation margins required by Section 6.3.1.5 of IEEE-323-1974 were not necessary.

3.11(B).5.4 Aging

During the NUREG-0588 review, two general observations were made concerning equipment aging:

1. Some IEEE-323-1974 equipment underwent accelerated thermal aging based on the Arrhenius method. This approach was considered acceptable.

APPENDIX 3A - CONFORMANCE TO NRC REGULATORY GUIDES

This appendix briefly discusses the extent to which the standard plant conforms to NRC published regulatory guides, Division 1. The Standard Plant FSAR Appendix 3A may refer to the Addendum Appendix 3A or the Union Electric Company Operational Quality Assurance Manual (OQAM) for the specific regulatory commitment for certain regulatory guides. However, in cases where a reference is not made to the Addendum Appendix 3A or the OQAM, the commitment is as stated in the Standard Plant Appendix 3A and the same regulatory position is not repeated in the Addendum Appendix 3A or the OQAM. The statement of specific regulatory commitment for the following regulatory guides is located as indicated:

Callaway FSAR, Standard Plant - Regulatory Guides 1.1, 1.2, 1.3,~~1.4,~~ 1.5, 1.6, 1.7, 1.9, 1.10, 1.11, 1.12, 1.13, 1.14, 1.15, 1.18, 1.20, 1.22, 1.24,~~1.25,~~ 1.26, 1.29, 1.31, 1.32, 1.34, 1.35, 1.36, 1.40, 1.41, 1.42, 1.43, 1.44, 1.45, 1.46, 1.47, 1.48, 1.49, 1.50, 1.51, 1.52, 1.53, 1.54, 1.55, 1.56, 1.57, 1.59, 1.60, 1.61, 1.62, 1.63, 1.65, 1.66, 1.67, 1.68, 1.68.1, 1.68.2, 1.69, 1.70, 1.71, 1.72, 1.73, 1.75, 1.76, 1.77, 1.78, 1.79, 1.80, 1.81, 1.82, 1.83, 1.84, 1.85, 1.87, 1.89, 1.90, 1.92, 1.93, 1.95, 1.96, 1.97, 1.98, 1.99, 1.100, 1.101, 1.102*, 1.103, 1.104, 1.105, 1.106, 1.107, 1.108, 1.110, 1.112, 1.115, 1.117, 1.118, 1.119, 1.120, 1.121, 1.122, 1.124, 1.126, 1.128, 1.129, 1.130, 1.131, 1.133, 1.136, 1.137, 1.139, 1.140, 1.141, 1.142, 1.143, 1.147, 1.150, 1.152, 1.155, 1.158, 1.160, 1.163, 1.181, 1.182, 1.187, 1.195, and 1.205.

1.183

Callaway FSAR, Site Addendum - Regulatory Guides 1.17, 1.21, 1.23, 1.27, 1.59, 1.86, 1.91, 1.102*, 1.109, 1.111, 1.113, 1.114, 1.125, 1.127, 1.132, 1.134, 1.138, and 1.145.

Union Electric Operational Quality Assurance Manual - Regulatory Guides 1.8, 1.28, 1.30, 1.33, 1.37, 1.38, 1.39, 1.58, 1.64, 1.74, 1.88, 1.94, 1.116, 1.123, 1.144, and 1.146.

Exceptions to the guides are identified, and justification is presented or referenced. In the discussion of each guide, the sections or tables of the FSAR, where more detailed information is presented, are referenced. The referenced tables provide a position-by-position comparison to each regulatory position of section C of the regulatory guides. All statements within the Regulatory Position Section (C) of the Regulatory Guides are considered requirements unless a specific exception or clarification has been committed to by Union Electric. This is true regardless of the qualifier (i.e., "shall" or "should") which prefaces the statement. As regards to standards endorsed by the Regulatory Guide, unless further qualified within the Regulatory Guide, "shall" statements denote requirements while "should" statements denote recommendations.

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* Refer to both the Callaway FSAR Standard Plant and the Callaway FSAR Site Addendum for the complete statement of regulatory commitment.

post-irradiation fracture toughness data have been obtained. Evaluation of the data obtained to date on material irradiated to fluences between 2.2 and 4.5 x 10^{19} n/cm^2 indicates that the reference toughness curve, as contained in the American Society of Mechanical Engineers (ASME) Code, Section III, remains a conservative lower bound for toughness values for pressure vessel steels.

Details of progress and results obtained in the HSST program are available in the HSST program progress reports issued by Oak Ridge National Laboratory.

Regulatory Position C.2 is followed, inasmuch as no significant changes have been made in approved core or reactor designs.

Regulatory Position C.3 is followed, since the vessel design does not preclude the use of an engineering solution to assure adequate recovery of the fracture toughness properties of the vessel material. If additional margin is needed, the reactor vessel can be annealed at any point in its service life. This solution is already feasible, in principle, and could be performed with the vessel in place.

NOTE: Regulatory Guide 1.2 (Safety Guide 2) has been withdrawn by the NRC Staff letter to Regulatory Guide Distribution List, June 17, 1991. The guide has been superseded by 10 CFR 50, Section 50.61, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events. 10 CFR 50, Section 50.61 establishes screening criteria to effectively limit the extent of irradiation embrittlement permitted for reactor pressure vessel materials. The pressurized thermal shock requirements are sufficient to address thermal shock concerns. The withdrawal of Regulatory Guide 1.2 (Safety Guide 2) does not alter prior or existing licensing commitments based on its use.

REGULATORY GUIDE 1.3 REVISION 2 DATED 6/74

Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors

DISCUSSION:

The recommendations of this regulatory guide are not applicable to a PWR.

REGULATORY GUIDE 1.4 REVISION 2 DATED 6/74

Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors

DISCUSSION: Use of this regulatory guide in Chapter 15 has been replaced by Regulatory Guide 1.183 for Alternative Source Term (AST) application

~~The recommendations of this regulatory guide are met as described in Table 15.6-7.~~

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 15.7-1.

REGULATORY GUIDE 1.25 REVISION 0 DATED 3/72

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 (Safety Guide 25)

DISCUSSION:

Use of this regulatory guide in Chapter 15 has been replaced by Regulatory Guide 1.183 for alternative source term application

~~The recommendations of this regulatory guide are met as described in Table 15.7-2~~.

REGULATORY GUIDE 1.26 REVISION 3 DATED 2/76

Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 3.2-4. As described in Section 3.2, Westinghouse utilizes the safety classes defined in ANSI N18.2a-1975.

REGULATORY GUIDE 1.27

Ultimate Heat Sink for Nuclear Power Plants

DISCUSSION:

Refer to Appendix 3A in the Site Addendum.

REGULATORY GUIDE 1.28

Quality Assurance Program Requirements (Design and Construction)

DISCUSSION:

Refer to the Union Electric Company Operational Quality Assurance Manual.

REGULATORY GUIDE 1.29 REVISION 3 DATED 9/78

Seismic Design Classification

DISCUSSION:

DISCUSSION:

UE complies with the requirements of this Regulatory Guide.

REGULATORY GUIDE 1.163 REVISION 0 DATED 9/95

Performance-Based Containment Leak-Test Program

DISCUSSION:

UE complies with the recommendations of this Regulatory Guide as discussed in the Leak Rate Test Program (ESP-SM-1001).

REGULATORY GUIDE 1.181 Initial Issue DATED 9/99

Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e) (Endorses NEI 98-03)

REGULATORY GUIDE 1.183 Initial Issue DATED 7/00

Alternative radiological source terms for evaluating design basis accidents at nuclear power reactors

DISCUSSION

The recommendations of this regulatory guide are met as described in Chapter 15

REGULATORY GUIDE 1.182 Initial Issue DATED 5/00

Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants

DISCUSSION:

AmerenUE complies with the requirements of this Regulatory Guide.

REGULATORY GUIDE 1.187 Initial Issue DATED 11/00

Guidance for Implementation of 10 CFR 50.59, Changes Tests and Experiments.

DISCUSSION:

AmerenUE complies with Regulatory Guide 1.187 with the following clarifications to NEI 96-07 Guidelines for 10 CFR 50.59 Implementation, dated November 2000:

1. With regard to Regulatory Position C.1 of Regulatory Guide 1.187, AmerenUE substitutes the word, "Implementation" for "Evaluations" to reflect title of NEI 96-07, dated November 2000.

Training and qualification of personnel performing Safety Analysis calculations will be accomplished in accordance with AmerenUE's Engineering Support Personnel training program.

4. 2.4 Comparison Calculations

Comparison and benchmark calculations will be performed in accordance with approved procedural controls. Computer codes used for safety analysis will be controlled in accordance with AmerenUE's software control procedures.

5. 2.5 Quality Assurance and Change Control

Safety Analysis calculations will be performed in accordance with the AmerenUE OQAP, which implements 10CFR50, Appendix B Criterion III. Computer codes used for safety analysis will be controlled in accordance with AmerenUE's software control procedures.

REGULATORY GUIDE 1.195 REVISION 0 DATED 5/03

Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors

DISCUSSION:

The recommendations of this regulatory guide are met as described in the analysis of FSAR design basis accidents and their radiological consequences.

REGULATORY GUIDE 1.205 REVISION 01 DATED 12/2009

Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met. Refer to the FSAR Section 9.5.1.

REGULATORY GUIDE 1.194 Initial Issue DATED 6/03
Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants

DISCUSSION

The recommendations of this regulatory guide are met as described in Site Addendum 2, Section 2.3.4.4.

4.2 FUEL SYSTEM DESIGN

The plant design conditions are divided into four categories in accordance with their anticipated frequency of occurrence and risk to the public: Condition I - Normal Operation; Condition II - Incidents of Moderate Frequency; Condition III - Infrequent Incidents; and Condition IV - Limiting Faults. Chapter 15.0 describes bases and plant operation and events involving each condition.

The reactor is designed so that its components meet the following performance and safety criteria:

a. The mechanical design of the reactor core components and their physical arrangement, together with corrective actions of the reactor control, protection, and emergency cooling systems (when applicable) ensure that:

   1. Fuel damage* is not expected during Condition I and Condition II events. It is not possible, however, to preclude a very small number of rod failures. These are within the capability of the plant cleanup system and are consistent with plant design bases.**

   2. The reactor can be brought to a safe state following a Condition III event with only a small fraction of fuel rods damaged** although sufficient fuel damage might occur to preclude immediate resumption of operation.

   3. The reactor can be brought to a safe state and the core can be kept subcritical with acceptable heat transfer geometry following transients arising from Condition IV events.

b. The fuel assemblies are designed to withstand loads induced during shipping, handling, and core loading without exceeding the criteria of Section 4.2.1.5. Fuel assemblies can withstand loads introduced by a postulated reactor vessel head drop as evaluated in Section 9.1.4.3 for Westinghouse fuel.

c. The fuel assemblies are designed to accept control rod insertions in order to provide the required reactivity control for power operations and reactivity shutdown conditions (if in such core locations).

guidelines

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* Fuel damage as used here is defined as penetration of the fission product barrier (i.e., the fuel rod cladding).

** In any case, the fraction of fuel rods damaged must be limited so as to meet the dose ~~guideline~~ of 10 CFR ~~100~~.

50.67 or Regulatory Guide 1.183

4.3 NUCLEAR DESIGN

4.3.1 DESIGN BASES

This section describes the design bases and functional requirements used in the nuclear design of the fuel and reactivity control system and relates these design bases to the General Design Criteria (GDC) presented in 10 CFR 50, Appendix A. Where applicable, supplemental criteria such as the "Final Acceptance Criteria for Emergency Core Cooling Systems" are addressed. Chapter 15.0.9 addresses the Fuel System Nuclear Design Bases that affect the fuel fission product inventory used in the Accident Dose Assessment. Before discussing the nuclear design bases, it is appropriate to briefly review the four major categories ascribed to conditions of plant operation.

The full spectrum of plant conditions is divided into four categories, in accordance with the anticipated frequency of occurrence and risk to the public:

a. Condition I - Normal Operation
b. Condition II - Incidents of Moderate Frequency
c. Condition III - Infrequent Faults
d. Condition IV - Limiting Faults

In general, the Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Condition II incidents are accommodated with, at most, a shutdown of the reactor with the plant capable of returning to operation after corrective action. Fuel damage (fuel damage as used here is defined as penetration of the fission product barrier, i.e., the fuel rod cladding) is not expected during Condition I and Condition II events. It is not possible, however, to preclude a very small number of rod failures. These are within the capability of the CVCS and are consistent with the plant design basis.

Condition III incidents do not cause more than a small fraction of the fuel elements in the reactor to be damaged, although sufficient fuel element damage might occur to preclude immediate resumption of operation. The release of radioactive material due to Condition III incidents is not sufficient to interrupt or restrict public use of these areas beyond the exclusion radius. Furthermore, a Condition III incident does not by itself generate a Condition IV fault or result in a consequential loss of function of the reactor coolant or reactor containment barriers.

Condition IV occurrences are faults that are not expected to occur but are defined as limiting faults which must be designed against. Condition IV faults do not cause a release of radioactive material that results in exceeding the limits of 10 CFR ~~100~~ 50.67 or Regulatory Guide 1.183.

The core design power distribution limits related to fuel integrity are met for Condition I occurrences through conservative design and maintained by the action of the control system. The requirements for Condition II occurrences are met by providing an adequate protection system which monitors reactor parameters. The control and

TABLE OF CONTENTS (Continued)

CHAPTER 6.0

ENGINEERED SAFETY FEATURES

Engineered safety features (ESF) are those safety-related systems and components designed to directly mitigate the consequences of a design basis accident by:

a. Protecting the fuel cladding

b. Ensuring the containment integrity

c. Limiting fission product releases to the environment and Control Room within the guideline values of 10 CFR, Part ~~100~~ 50.67 and Regulatory Guide 1.183

The limiting design basis accidents which are di[illegible]hapter 15.0 and Section 6.3 are:

a. Loss-of-coolant accident (LOCA)

b. Main steam line break (MSLB)

c. Steam generator tube rupture

d. Fuel handling accident

The engineered safety features consist of the following systems:

a. Containment (Section 6.2.1)

b. Containment heat removal (Section 6.2.2)

c. Containment isolation (Sections 6.2.4 and 6.2.6)

d. Containment combustible gas control (Section 6.2.5)

e. Emergency core cooling (Section 6.3)

f. Fission product removal and control systems (Section 6.5)

g. Emergency HVAC and filtration (Section 9.4)

h. Control room habitability (Section 6.4)

i. Auxiliary feedwater (Section 10.4.9)

The containment is provided to contain radioactivity following a LOCA.

6.2 CONTAINMENT SYSTEMS

The containment systems include the containment, the containment heat removal systems, the containment isolation system, and the containment combustible gas control system.

The design basis accident (DBA) is defined as the most severe of a spectrum of hypothetical loss-of-coolant accidents (LOCA). The ability of the containment systems to mitigate the consequences of a DBA depends upon the high reliability of these systems. This section provides the design criteria and evaluations to demonstrate that these systems function within the specified limits throughout the unit operating lifetime.

6.2.1 CONTAINMENT FUNCTIONAL DESIGN

A physical description of the containment and the design criteria relating to construction techniques, static loads, and seismic loads is provided in Section 3.8. This section pertains to those aspects of containment design, testing, and evaluation that relate to the accident mitigation function.

6.2.1.1 Containment Structure

6.2.1.1.1 Design Bases

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the DBA without exceeding the design leakage rate, as required by 10 CFR 50, Appendix A, General Design Criterion 50, and that, in conjunction with the other containment systems and the other engineered safety features, the release of radioactive material subsequent to a DBA does not result in doses in excess of the guideline values specified in 10 CFR ~~100.~~ 50.67 and Regulatory Guide 1.183 The radiological consequences of the DBA are presented in Section 15.6.

a. Assumed Accident Conditions

For the purpose of determining the design pressure requirements for the containment structure and the containment internal structures, the following simultaneous occurrences are assumed:

1. The postulated reactor coolant system pipe rupture, as listed in Table 6.2.1-1, is assumed to be concurrent with the loss of offsite power and the worst single active failure. No two pipe breaks are assumed to occur simultaneously or consecutively. For design loadings on the systems used to mitigate the consequences of a postulated reactor coolant system pipe rupture, a safe shutdown earthquake is assumed.

The component cooling water system is described in Section 9.2.2, the essential service water system is described in Section 9.2.1, and the ultimate heat sink is described in Section 9.2.5.

Single failures in systems which remove energy from the containment are considered to be consistent with the single failures assumed in the development of the mass and energy release data. The energy removal capability of the containment air coolers, the containment spray system, and the residual heat removal system consider the parameters provided in Table 6.2.1-3.

50.67 and Regulatory Guide 1.183

f. Bases for Containment Depressurization Rate

To meet the containment safety design basis of limiting the release of radioactive material subsequent to a DBA so that the doses are within the guideline values specified in 10 CFR ~~100~~, the containment pressure is reduced to less than 50 percent of the containment design pressure within 24 hours after an accident. Chapter 15.0 contains the assumptions used in the analysis of the offsite radiological consequences of the accident.

g. Bases for Minimum Containment Pressure Used in ECCS Performance Studies

The minimum containment pressure transient used in the analysis of the emergency core cooling system's capability is based on the conservative overestimated heat removal capability and pressure reduction capability of the containment structures and the containment systems and on the conservative reactor coolant system thermal analysis provided in Section 15.6. The determination and evaluation of the minimum containment pressure transient are provided in Section 6.2.1.5.

6.2.1.1.2 Design Features

The principal containment and containment subcompartment design parameters are provided in Table 6.2.1-2. General arrangement drawings for the reactor containment are provided in Figures 1.2-9 through 1.2-18. Simplified arrangement drawings illustrating the nodalization model used for the containment subcompartment analyses are provided in Figures 6.2.1-43 through 6.2.1-46, 6.2.1-51 through 6.2.1-55, and 6.2.1-76.

a. Missile and Pipe Whip Protection

Missile shield considerations are described in Section 3.5. The structural design of the containment and the containment subcompartments is discussed in Section 3.8. The designed structural strength considers the effects of pipe whip and jet forces, as discussed in Section 3.6.

The analysis has shown that even assuming a stuck RCCA with or without offsite power, and assuming a single failure in the engineered safeguards, the core remains in place and intact. Radiation doses will not exceed 10 CFR ~~100~~ guidelines.

50.67 and Regulatory Guide 1.183

DECREASE IN HEAT REMOVED BY THE SECONDARY SYSTEM

Feedwater System Pipe Break

A major feedwater line rupture is defined as a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell side fluid inventory in the steam generators. If the break is postulated in a feedwater line between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. Further, a break in this location could preclude the subsequent addition of auxiliary feedwater to the affected steam generator. (A break upstream of the feedwater line check valve would affect the NSSS only as a loss of feedwater. This case is covered by the evaluation in Sections 15.2.6 and 15.2.7).

Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either an RCS cooldown (by excessive energy discharge through the break) or an RCS heatup. Potential RCS cooldown resulting from a secondary pipe rupture is evaluated in Section 15.1.5. Therefore, only the RCS heatup effects are evaluated for a feedwater line rupture.

A feedwater line rupture reduces the ability to remove heat generated by the core from the RCS for the following reasons:

a. Feedwater flow to the steam generators is reduced. Since feedwater is subcooled, its loss may cause reactor coolant temperatures to increase prior to reactor trip.

b. Fluid in the steam generator may be discharged through the break, and would then not be available for decay heat removal after trip.

c. The break may be large enough to prevent the addition of any main feedwater after trip.

An auxiliary feedwater system functions to ensure the availability of adequate feedwater so that:

a. No substantial overpressurization of the RCS occurs (less than 110 percent of design pressures); and

b. Sufficient liquid in the RCS is maintained so that the core remains inplace and geometrically intact with no loss of core cooling capability.

operators to achieve and/or maintain the plant in a safe shutdown condition. The following safety design bases are met:

SAFETY DESIGN BASIS ONE - The habitability systems are housed within a structure capable of withstanding the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, and external missiles (GDC-2).

SAFETY DESIGN BASIS TWO - The habitability systems are designed to remain functional after an SSE and to perform their intended function following a postulated hazard, such as a fire, internal missiles, or pipe break (GDC-3 and 4).

SAFETY DESIGN BASIS THREE - Habitability system redundancy is provided so that safety functions can be performed, assuming a single active component failure coincident with a loss of offsite power.

SAFETY DESIGN BASIS FOUR - The habitability systems are designed so that the active components are capable of being tested during plant operation. Provisions are made to allow for inservice inspection of appropriate components of the control room air-conditioning system.

SAFETY DESIGN BASIS FIVE - The habitability systems are designed and fabricated according to codes consistent with the quality group classification assigned by Regulatory Guide 1.26 and the seismic category assigned by Regulatory Guide 1.29. The power supply and control functions are in accordance with Regulatory Guide 1.32.

SAFETY DESIGN BASIS SIX - The capability to isolate all nonsafety-related HVAC system penetrations of the control building boundary is provided, if required, so that the occupation and habitability of the control room will not be compromised.

SAFETY DESIGN BASIS SEVEN - The radiation exposure of control room personnel throughout the duration of any one of the postulated DBAs discussed in Chapter 15.0 does not exceed the guideline values of GDC-19.

SAFETY DESIGN BASIS EIGHT - Throughout the duration of any one of the postulated hazardous chemical releases discussed in Section 2.2 of the Site Addendum or DBAs discussed in Chapter 15.0 of the FSAR, the habitability systems maintain the control room atmosphere at environmental conditions suitable for occupancy per GDC-19. The habitability systems comply with Regulatory Guides 1.78 and 1.95.

SAFETY DESIGN BASIS NINE - The control room ventilation system is capable of automatic transfer from its normal operational mode to its emergency mode upon detection of airborne radiation ~~resulting~~ in exposure of control room personnel in excess of GDC-19 limits.

an event or the release of

order to prevent

SAFETY EVALUATION SIX - Section 9.4.1.2.3 describes the provisions made to assure the isolation of the control room.

SAFETY EVALUATION SEVEN - The direct radiation exposure rate of a control room occupant throughout the duration of any one of the postulated DBAs discussed in Chapter 15.0 ~~does not exceed 0.5 mr/hr whole-body, and thus~~ will not exceed GDC-19 requirements. A detailed discussion of the dose calculation model for control room operators is discussed in Appendix 15A. Control room shielding design, based on the most limiting design basis LOCA fission product release, is discussed in Section 12.3.

will not exceed requirements. Inclusion of the resulting contribution with the Control Room operator inhalation, immersion, and transit dose

SAFETY EVALUATION EIG ... lated hazardous chemical releases ... n Chapter 15.0, the habitability system ... tions below those established by Regul a ... iance with Requlatory Guides 1.78 and 1.95 is provided in Tables 6.4-1 and 6.4-2, respectively.

SAFETY EVALUATION NINE - Upon detection of high radiation in the induction trunk, the control room ventilation system is capable of automatic transfer from normal to emergency mode to minimize the exposure of control room personnel.

6.4.5 TESTS AND INSPECTIONS

Testing and inspection of control room HVAC systems are described in Section 9.4.1.4.

The emergency mode of the control room HVAC system will undergo an acceptance test to verify that the system will maintain a 1/8-inch w.g. positive pressure in the emergency zone. Testing complies with Regulatory Guide 1.95, as described in Table 6.4-2.

The control room is classified as Type B per Regulatory Guide 1.78. Since the air exchange rate exceeds 0.06 air exchanges per hour for the control room, periodic testing of the control room pressurization system is not required per the exclusion provisions of Regulatory Guides 1.78 and 1.95. This periodic testing is not required for the Callaway plant based on the adequacy of a 400 cfm (nominal with tolerance of (+) 40 cfm, (-) 40 cfm) pressurization flow rate (Reference 1).

6.4.6 INSTRUMENTATION REQUIREMENTS

Safety-related instrumentation and isolation signals are discussed in Sections 9.4.1.2.3 and 7.3.

Indication of all fan operational status is provided in the control room.

An indication of the position of all isolation dampers is provided in the control room.

All instrumentation associated with filtration units complies with Regulatory Guide 1.52, as described in Table 9.4-2.

10 CFR ~~100~~ 50.67 and Regulatory Guide 1.183. The safety evaluations which demonstrate the design and construction of the ESF filtration systems are provided in Sections 9.4.2 and 9.4.3.

SAFETY EVALUATION TWO - The results of the analyses described in Chapter 15.0 demonstrate that the control building HVAC systems reduce and control fission product release to the control room following a LOCA, such that radiation exposures of control room personnel are within the requirements of GDC-19. The safety evaluations which demonstrate the design and construction of these control building HVAC systems are provided in Sections 9.4.1 and 6.4.

6.5.1.4 Tests and Inspections

Tests and inspections for ESF filter systems are described in Sections 9.4.1.4, 9.4.2.4, and 9.4.3.4.

6.5.1.5 Instrumentation Requirements

Instrumentation and controls are provided to facilitate automatic operation and remote control of the system and to provide continuous indication of system parameters. Further descriptions are provided in Sections 9.4.1.5, 9.4.2.5, and 9.4.3.5.

6.5.1.6 Materials

The materials used for ESF filtration systems were chosen considering the environmental conditions and are commensurate with acceptable construction practices.

6.5.2 CONTAINMENT SPRAY SYSTEM

The containment spray system (CSS) is an ESF, the functions of which are to reduce pressure and temperature in the containment atmosphere following a postulated LOCA or MSLB inside containment and to remove radioactive fission products from the containment atmosphere. These functions are performed by spraying a chemical solution into the containment atmosphere through a large number of nozzles on spray headers located in the containment dome. Reduction of pressure and temperature in the containment with the CSS is discussed in Section 6.2.2.1.

Radioiodine in its various forms is the fission product of primary concern in the evaluation of a LOCA. It is absorbed by the containment spray from the containment atmosphere. To enhance this iodine absorption capacity of the spray, the spray solution is adjusted to an alkaline pH which promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms tending to plate out on containment structures or to be retained in the containment recirculation sumps.

The physical characteristics of the CSS are discussed in Section 6.2.2.1. Discussed herein is the containment spray system's fission product removal capability following a LOCA.

6.5.2.1 Design Bases

6.5.2.1.1 Safety Design Bases

SAFETY DESIGN BASIS ONE - The CSS is designed to provide an equilibrium sump solution pH of greater than or equal to 7.1 following the complete dissolution of the trisodium phosphate stored in baskets adjacent to the containment recirculation sumps.

SAFETY DESIGN BASIS TWO - The CSS is capable of reducing the iodine and particulate fission product inventories in the containment atmosphere such that the offsite radiation exposures resulting from a design basis LOCA are within the plant siting dose guidelines of 10 CFR ~~100~~ 50.67.

Additional safety design bases are included in Section 6.2.2.1, in which the capability of the spray system to remove heat from the containment atmosphere is discussed.

6.5.2.1.2 Power Generation Design Basis

The CSS has no power generation design basis.

6.5.2.2 System Design

6.5.2.2.1 General Description

The spray additive tank has been retired in place and associated lines have been capped, as shown schematically in Figure 6.2.2-1.

Initially, water from the refueling water storage tank (RWST) is used for containment spraying followed by water from the containment recirculation sumps.

Those parts of the system in contact with containment spray fluids, are stainless steel or an equivalent corrosion-resistant material.

The trisodium phosphate (TSP-C) baskets constructed of stainless steel mounted to carbon steel supports contain sufficient TSP-C to bring the equilibrium sump fluid to a minimum pH of 7.1 upon mixing with the borated water from the refueling water storage tank, the accumulators, and reactor coolant. This assures continued iodine retention effectiveness of the sump water during the recirculation phase.

The spray header design, including the number of nozzles per header, nozzle spacing, and nozzle orientation, is provided in Section 6.2.2.1 and shown in Figures 6.2.2-2 and 6.2.2-4. Each spray header layout is oriented to provide more than 90-percent area coverage at the operating deck of the reactor building.

Total containment free volume, unsprayed containment free volume, specific unsprayed regions and volumes, and post-accident ventilation between sprayed and unsprayed

On actuation, approximately 5 percent of each spray pump's discharge flow is recirculated.

When the refueling water storage tank has reached its specified low-low-2 level limit, recirculation spray flow is manually initiated. The operator can remotely initiate recirculation flow by use of either or both of the spray pumps. Sections 6.2.2.1.5 and 6.5.2.5 address the instrumentation and information displays available to the operator, in order for manual switchover of the CSS to take place.

System flow rates and the duration of operational modes are presented in Section 6.2.2.1.2.3.

Design operation of the CSS is such that LOCA iodine removal requirements are fulfilled during the injection phase and the amount of TSP-C provided is sufficient to ensure long-term iodine retention. Following a large break LOCA, the containment spray during the injection phase will be a boric acid solution having a pH of about 4.5. The desired pH level is greater than 7.0 to assure iodine retention in the sumps, to limit corrosion and the associated production of hydrogen, and to limit chloride induced stress-corrosion cracking of austenitic stainless steels. To adjust the sump solution pH into the desired range, a minimum of 9000 pounds of trisodium phosphate dodecahydrate (NA_3PO_4 • 12 H_2O • 1/4 NaOH) is stored in two baskets, one adjacent to each containment recirculation sump, at an elevation to assure TSP-C disolution. This amount of trisodium phosphate is sufficient to assure that the equilibrium sump solution pH will be greater than or equal to 7.1. The containment iodine removal credit assumed in the calculation of offsite doses following a LOCA is provided in Table 15.6-6.

6.5.2.3 Safety Evaluation

The safety evaluations are numbered to correspond to the safety design bases.

SAFETY EVALUATION ONE - The system's capability to reduce the airborne fission product inventory is based on the surface area of the spray solution for removal during injection and on sump solution pH for retention during recirculation, and on the system's capability to provide spray for essentially all regions of the containment, considering post-accident conditions.

During injection, the effectiveness of the spray against elemental iodine vapor is chiefly determined by the rate at which fresh solution surface area is introduced into the containment atmosphere, as discussed in Reference 3. The first-order spray removal coefficient calculated per Reference ~~3~~, as discussed in Section 6.5A.3, is 37 hr^{-1}. Thus, the elemental iodine removal coefficient of ~~10~~ hr^{-1} used in Section 15.6.5 is conservative. The minimum equilibrium sump pH of 7.1 assures iodine retention in the recirculated spray liquid.

4 20

at least 4 hours

The system is designed to provide a spray solution during the recirculation phase with a minimum equilibrium pH of 7.1. The mass of TSP-C in the baskets results in this minimum pH level in the sumps.

The worst case concentration during the injection phase would be greater than or equal to 4.0 but less than 7.0 when water from the refueling water storage tank is sprayed directly to the containment. The injection phase is the only time that this pH = 4.0 condition could exist. The injection phase is short (1 hour) relative to the entire spray duration (~~approximately 24 hours~~). During the spray recirculation phase, ~~the equilibrium pH range is 7.1-8.1.~~ This spray is directed through the same spray headers and, therefore, should rinse all of the previously sprayed components (for a period of ~~approximately 23 hours~~).

the sump pH increases to and remains higher than the 7.1 equilibrium value.

The minimum equilibrium sump pH c Specification minimum of 9000 lbm of TSP-C in the baskets and the maximum sump solution boric acid concentration of 2500 ppm boron. With the Technical Specification maximum of 14,300 lbm of TSP-C in the baskets and the minimum sump solution boric acid concentration of 1971 ppm boron, the maximum equilibrium sump pH would be less than 8.1.

approximately 3 hours

The previously evaluated upper bound for containment spray pH of 11.0 will continue to be cited, consistent with Section 3.11(B).1.2.2, for the purpose of performing EQ reviews.

Another issue that has been reviewed is the unlikely, but possible, event in which an initially concentrated solution of TSP-C occupies the stagnant volume of an inoperable sump. This situation would not last for long since, as the recirculated sump fluid is cooled in the RHR heat exchangers, sufficient buoyancy-driven circulation within containment will result to displace the stagnant solution and eventually yield a uniform, equilibrium solution.

SAFETY EVALUATION TWO - The spray ~~iodine~~ removal analysis is based on the assumptions that:

a. Only one out of two spray pumps is operating

b. The ECCS is operating at its maximum capacity

The spray system is assumed to spray approximately 85 percent of the total containment net free volume. This volume consists of those areas directly sprayed plus those volumes which have good communication with the directly sprayed volumes. The remaining 15 percent of the containment free volume has restricted communication with the sprayed volumes and is assumed to be unsprayed. A description of the unsprayed volumes is presented in Table 6.5-2.

The performance of the spray system was evaluated at the containment post-LOCA calculated saturation temperature corresponding to the calculated peak pressures and containment design pressure provided in Table 6.2.1-2. The net spray flow rate of ~~3,131~~ 3,086 gpm (see Table 6.5-2) per train ,which bounds both injection and recirculation spray design flow rates, was used in the calculations described in Appendix 6.5A.

Based on Regulatory Guide ~~1.4~~ 1.183, three species of airborne iodine are postulated to exist in the containment atmosphere following a LOCA. These are elemental, particulate, and organic species.

It has been assumed in these evaluations of spray removal effectiveness that organic ... e containment spray. A limited credit for the removal ... tal iodine has been taken in the offsite and control room dose calculation, assuming that the spray removal rate is ~~0.45~~ 6.46 hr^{-1} until a decontamination factor of 50 is attained for particulates (and 0.646 hr^{-1} thereafter) and that spray removal rate is ~~10~~ 20 hr^{-1} until a decontamination factor (DF) of ~~28.7~~ 200 is attained for elemental iodine. These assumptions underestimate the actual amounts of iodine removed and thus result in calculated accident doses higher than could realistically be expected.

Utilizing the dose analysis input parameters indicated above, in Table 6.5-2, and in Appendix 15A, the dose analysis of Section 15.6.5 demonstrates that offsite radiation exposures resulting from a design basis LOCA are within the plant ~~siting~~ dose guidelines of 10 CFR ~~100~~ 50.67.

Appendix 6.5A provides the model used to calculate the ~~iodine~~ removal coefficients provided in Table 6.5-2.

6.5.2.4 Tests and Inspections

CSS tests and inspections are discussed in Section 6.2.2.1.4, including spray nozzle tests and inspections.

6.5.2.5 Instrumentation Requirements

Containment spray instrumentation is discussed in Section 6.2.2.1.5.

6.5.2.6 Materials

The chemical compositions of the containment spray fluid entering the spray header during the injection phase of containment spray and the containment spray fluid in the system during the recirculation phase of containment spray (containment recirculation sump solution) are provided in Table 6.5-5.

None of the materials used is subject to decomposition by the radiation or thermal environment.

The corrosion of materials in the NSSS and the containment building, resulting from the spray solution used for iodine absorption, has been tested by the Reactor Division at ORNL (Ref. 2). The spray solutions provided in Table 6.5-5 result in negligible corrosion, based on these studies.

TSP-C does not undergo radiolytic decomposition in the post-LOCA environment. Sodium has a low neutron absorption cross section and will not undergo significant activation.

With respect to the potential for decomposition, TSP-C is stable to at least 158°F. Temperatures 158°F may result in the loss of H_2O from the TSP-C but will not affect its caustic properties.

6.5.3 FISSION PRODUCT CONTROL SYSTEMS

6.5.3.1 Primary Containment

50.67 or Regulatory Guide 1.183

The containment consists of a prestressed post-tensioned, reinforced concrete structure with cylindrical walls, hemispherical dome, and base slab lined with a welded quarter-inch carbon steel liner plate, which forms a continuous, leaktight membrane. Details of the containment structural design are discussed in Section 3.8. Layout drawings of the containment structure and the related items are given in the general arrangement drawings of Section 1.2.

The containment walls, liner plate, penetrations, and isolation valves function to limit the release of radioactive materials, subsequent to postulated accidents, such that the resulting offsite doses are less than the guideline values of 10 CFR ~~100~~. Containment parameters affecting fission product release accident analyses are given in Appendix 15A.

sometime after 4 hours.

Long-term containment pressure response to the design basis LOCA is shown in Figure 6.2.1-1. Relative to this time period, the CSS is operated to reduce iodine concentrations and containment atmospheric temperature and pressure commencing with system initiation, at approximately 60 seconds, as shown in Table 6.2.2-3 and ending ~~when containment pressure has returned to normal~~. For the purpose of post-LOCA dose calculations discussed in Chapter 15.0, two dose models have been assumed, the 0-2 hour case and the 0-30 day case, as shown in Appendix 15A.

The containment minipurge system may be operated for personnel access to the containment when the reactor is at power, as discussed in Section 9.4.6.

Redundant, safety-related hydrogen recombiners are provided in the containment as the primary means of controlling postaccident hydrogen concentrations. A hydrogen purge system is provided for backup hydrogen control. See Section 6.2.5.3 (Safety Evaluation Eight).

Containment combustible gas control systems are discussed in detail in Section 6.2.5.

6.5.3.2 Secondary Containment

This section is not applicable to SNUPPS.

6.5.4 ICE CONDENSER AS A FISSION PRODUCT CLEANUP SYSTEM

This section is not applicable to SNUPPS.

6.5.5 REFERENCES

1. Spraying Systems Company Topical Report No. SSCO-15215-1C-304SS-6.3-NP, April 1977, "Containment Spray Nozzles for Nuclear Power Plants"

2. "Design Considerations of Reactor Containment Spray Systems, The Corrosion of Materials in Spray Solutions," ORNL-TM-2412Part III, December 1969

3. NUREG-0800, Standard Review Plan Section 6.5.2, Revision 2, "Containment Spray as a Fission Product Cleanup System," December 1988.

Insert 6.5.5

TABLE 6.5-1 ESF FILTRATION SYSTEMS INPUT PARAMETERS TO CHAPTER 15.0 ACCIDENT ANALYSIS

~~Emergency exhaust filter~~ Replace with Insert T6.5-1 ~~ent) 90~~

~~Emergency exhaust system flowrate (SCFM) 9,000~~

~~Control room filter adsorber unit efficiency (percent) 95~~

~~Control room air conditioning system flowrate (SCFM) per train~~

~~Filtered intake from control building 440~~

~~Filtered recirculation from control room 1,360~~

TABLE 6.5-2 INPUT PARAMETERS AND RESULTS OF SPRAY ~~IODINE~~ REMOVAL ANALYSIS

Core power rating	3,565 MWt
Total containment free volume ,maximum	~~2.50~~ 2.70 x 10^6 ft^3 **
Unsprayed containment free volume	<15.0 percent
Area coverage at the operating deck	
Design	>90 percent
Calculated	>93 percent
Mixing rate between sprayed and unsprayed volumes	~~85,000 cfm~~ 13,500 cfm ***
Dose model	One region
~~Minimum vertical distance to operating deck from lowest spray header~~ Average fall height of spray drops to operating deck	~~118 feet 2 in.~~ 131.4 feet
Net spray flow rate per train, injection Phase	~~3,131~~ 3086 gpm
Number of spray pumps operating	1
Spray solution pH	4.0 - ~~7.0~~ 6.59 (injection phase) ≥ 7.1 (recirculation phase at equilibrium)
Elemental iodine absorption coefficient, λs, used In LOCA offsite and control room dose calculations	~~10~~ 20 hr^{-1} (1)
Calculated λs	25.7 hr^{-1} (2) 37 hr^{-1}
Particulate iodine absorption coefficient, λp, used in LOCA offsite and control room dose calculations	~~0.45~~ 6.46 initially 0.646 (for DF > 50) hr^{-1} (3)
Calculated λp	0.73 hr^{-1} (4)
Spray drop size, design	See Figure 6.5-2
~~Schmidt number (See Section 6.5A.2)~~	~~11.58~~
~~Gas diffusivity (See Section 6.5A.2)~~	~~0.064 /sec~~
~~Partition coefficient (See Section 6.5A.2)~~	~~5,000~~

TABLE 6.5-2 (Sheet 2)

Gas phase mass transfer coefficient (See Section 6.5A.3) 9.5 ft/min

Terminal mass-mean drop velocity (See Section 6.5A.3) 790 ft/min

Partition coefficient (See Section 6.5A.3) 1100

200

(1) Until DF = ~~28.7~~.

(2) ~~λs of 25.7 hr^{-1} was calculated in Section 6.5A.2 and used in the EQ dose calculations discussed in Section 3.11(B).1.2.2 λs of 37 hr^{-1} was calculated in Section 6.5A.3 but 10 hr^{-1} was used in the offsite and control room dose calculations discussed in Section 15.6.5.~~

(3) Until DF = 50.

(4) λp of 0.73 hr^{-1} was calculated in Section 6.5A.1 and used in the EQ dose calculations.

λs of 25.7 hr-1 was calculated in Section 6.5A.2 and used in the EQ dose calculations discussed in Section 3.11(B).1.2.2
λs of 20 hr^{-1} was calculated in Section 6.5A.3 and was used in the offsite and control room dose calculations discussed in Section 15.6.5

** The maximum volume is conservative for the calculation of containment spray removal coefficients in Appendix 6.5A.3 and 6.5A.4 since the equation used includes the containment volume as the denominator.
*** Per Regulatory Guide 1.183, 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 unsprayed air volume is 405,000ft^3, this results in a mixing rate of 405,000/30 minutes = 13,500 cfm

APPENDIX 6.5A - ~~IODINE~~ REMOVAL MODELS FOR THE CONTAINMENT SPRAY SYSTEM

6.5A.1 PARTICULATE IODINE MODEL

FOR EQ DOSE CALCULATIONS

The spray washout model for aerosol particles is represented in equation form as follows:

$$\lambda P = \frac{3hEF}{2dV} \qquad (6.5A\text{-}1)$$

Where:

λP	=	spray removal constant for particles
h	=	drop fall height
E	=	total collection efficiency for a single drop
F	=	spray volumetric flow rate
d	=	mean drop diameter
V	=	volume of sprayed region

The capture of particles by falling drops results from Brownian diffusion, diffusiophoresis, interception, and impaction. Early in the injection phase, particles are removed mainly by impaction. Following injection, when the larger particles have already been removed, the removal rate is controlled by diffusiophoresis, which is the collection of particulates by steam condensing on the spray drops. The single drop collection efficiency, E, is taken as 0.0015, the minimum value observed in experimental tests (Ref. 1). The minimum collection efficiency, 0.0015, was only attained after the major fraction of airborne particles was removed. For early time periods, the removal rates were much higher than the minimum values ultimately reached. Per Reference 11, it is conservative to assume that E/D is 10 per meter initially (i.e., 1% efficiency for spray drops of one millimeter in diameter), changing abruptly to one per meter after the aerosol mass has been depleted by a DF of 50 (i.e., 98% of the particulate mass is ten times more readily removed than the remaining 2%). Using the 831 micron mean drop diameter identified in Table 6.5-2 and the minimum collection efficiency of 0.0015 from Reference 1, E/D would be 1.8 per meter which is consistent with the value from Reference 11 after a DF of 50 is attained.

The spray removal constant (λP) for particulate iodine has been calculated to be 0.73/hr, based on equation 6.5A-1, and used in Section 3.11(B).1.2.2.

A limited and conservative credit for spray removal of airborne particulates containing iodine has been taken in Section 15.6.5, assuming the spray removal constant is 0.45/hr, until a decontamination factor of 50 is reached, following the postulated LOCA (see Table 6.5-2).

Where:

vf	=	the specific volume of liquid at saturation, ft^3/lb
v	=	the specific volume of the drop before condensation, ft^3/lb
h_{fg}	=	the latent heat of evaporation, Btu/lb
h_g	=	the enthalpy of steam at saturation, Btu/lb
d and d'	=	the drop diameter before and after condensation, cm

Postma and Pasedag (Ref. 6) conclude that condensation will tend to increase the iodine washout rate due to the increased volume of the spray. Their effect has been conservatively ignored.

The drop exposure time calculated is based on the assumption that the drops were sprayed in such a manner that the initial downward velocity of the drops at the spray ring header elevation was zero. The drops fall under the effect of gravity from the spray ring header to the operating deck. The minimum height is given in Table 6.5-2. As the drop size increases, the average exposure time decreases from about 20 to 5 seconds. Incorporating the above parameters into equation 6.5A-16 with the sprayed containment volume, V, and assuming a single spray header flow rate, the value of the spray removal coefficient calculated (25.7 hr^{-1}) is presented in Table6.5-2.

The resulting elemental iodine spray removal constant is greater than 10/hr. A conservative removal constant of 10/hr is assumed ~~and used in the design basis LOCA evaluations presented in Section 15.6.5~~.

6.5A.3 ELEMENTAL IODINE MODEL FOR OFFSITE AND CONTROL ROOM DOSE CALCULATIONS

As discussed in Reference 11, the effectiveness of the spray during the injection phase against elemental iodine vapor is chiefly determined by the rate at which fresh solution surface area is introduced into the containment atmosphere. The rate of solution created per unit gas volume in the containment atmosphere may be estimated as (6F/VD), where F is the spray volumetric flow rate, V is the volume of the sprayed region, and D is the mean diameter of the spray drops. The first-order spray removal constant for elemental iodine, λ_s, may be taken to be:

$$\lambda_s = \frac{6k_g TF}{VD}$$

where k_g is the gas phase mass transfer coefficient and T is the drop fall time (or drop exposure time), which may be estimated by the ratio of the average fall height to the terminal velocity of the average drop. The above expression represents a first-order approximation if a well-mixed droplet model is used for spray absorption efficiency. This expression is valid for λ_s values equal to or greater than 10 per hour but less than 20 per hour. ~~Using this expression and the values contained in Table 6.5-2 a value of 37 hr^{-1} is calculated. A value of 10 per hour will continue to be used in the dose calculations of Section 15.6.5~~.

Insert 6.5A.3

~~Spray removal of elemental iodine continues until the DF of Equation 6.5A-15 is reached. Although the VL term in Equation 6.5A-15 represents the volume of the sumps plus any overflow from the sumps, it is conservative to just use the volume of the sumps for VL since a lower DF will result. The value for the partition coefficient, H, in Equation 6.5A-15 was taken from Figure 6 of Reference 13 using the 323°K plot at 14 hours (representative of the average conditions during a LOCA). The value of 1100 used is considered to be conservative since the sump fluid temperature at 14 hours would be greater than 323°K per Figure 6.2.1-7 and Figure 6 of Reference 13 shows that higher temperatures would be associated with higher partition coefficients. The resulting DF is calculated to be 28.7~~

Delete

Insert 6.5A.4

~~6.5A.4~~ 6.5A.5 REFERENCES

1. Hilliard, R. K., Coleman L. F., "Natural Transport Effects of Fission Product Behavior in the Containment System Experiment," BNWL-1457, Battelle Pacific Northwest Laboratories, Richland, Washington, December 1970.

2. Hilliard, R. K., et al, "Removal of Iodine and Particulates from Containment Atmospheres by Sprays - Containment Systems Experiment Interim Report," BNWL-1244, 1970.

3. Perkins, J. F., "Decay of U235 Fission Products," Physical Science Laboratory, RR-TR-63-11, U.S. Army Missile Command Redstone Arsenal, Alabama, July 25, 1963.

4. Parsley, Jr., L. F., "Design Considerations of Reactor Containment Spray Systems - Part VII," ORNL TM 2412, Part 7, 1970.

5. Ranz, W.E., and Marshall, Jr., W.R., "Evaporation from Drops," Chemical Engineering Progress 48, 141-46, 173-80, 1952.

6. Postma, A. K., and Pasedag, W. F., "A Review of Mathematical Models for Predicting Spray Removal of Fission Products in Reactor Containment Vessels," WASH-1329, U.S. Atomic Energy Commission, June 1974.

7. Griffiths, V., "The Removal of Iodine from the Atmosphere by Sprays," Report No. AHSB(S)R45, United Kingdom Atomic Energy Authority, London, 1963.

8. Eggleton, A. E. J., "A Theoretical Examination of Iodine-Water Partition Coefficient," AERE (R)-4887, 1967.

9. Postma, A. K., Coleman, L. F., and Hilliard, R. K., "Iodine Removal from Containment Atmospheres by Boric Acid Spray," BNP-100, Battelle-Northwest, Richland, Washington, 1970.

10. Coleman, L. F., "Iodine Gas-Liquid Partition," Nuclear Safety Quarterly Report, February, March, April 1970, BNWL-1315-2, Battelle-Northwest, Richland, Washington, p. 2.12-2.19, 1970.

11. NUREG-0800, Standard Review Plan Section 6.5.2, Revision 2, “Containment Spray as a Fission Product Cleanup System,” December 1988.

12. ANSI/ANS-56.5-1979, “PWR and BWR Containment Spray System Design Criteria.”

13. E.C. Beahm, W. E. Shockley, C. F. Weber, S. J. Wisbey, and Y. M. Wang, “Chemistry and Transport of Iodine in Containment,” NUREG/CR-4697, October 1986.

Insert 6.5A.5 here

CHAPTER 7.0

INSTRUMENTATION AND CONTROLS

7.1 INTRODUCTION

This section describes the various plant instrumentation and control systems and the functional performance requirements, design bases, system descriptions, design evaluations, and tests and inspections for each. The information provided in this chapter emphasizes those instruments and associated equipment which constitute the protection system, as defined in IEEE Standard 279-1971, "IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations."

The instrumentation and control systems provide automatic protection and exercise proper control against unsafe and improper reactor operation during steady state and transient power operations (Conditions I and II) and to provide initiating signals to mitigate the consequences of emergency and faulted conditions (Conditions III and IV). ANS conditions are discussed in Chapter 15.0.

Applicable criteria and codes are listed in Table 7.1-2.

7.1.1 IDENTIFICATION OF SAFETY-RELATED SYSTEMS

Safety-related instrumentation and control systems and their supporting systems are those systems required to ensure:

a. The integrity of the or Control Room sure boundary.

b. The capability to shut down the reactor and maintain it in a safe shutdown condition.

c. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR ~~100~~.

The definitions provid 50.67, Regulatory Guide 1.183, or 10 CFR 100, as appropriate. tems into the categories defined for

A listing of these systems, by categories, that are comparable to those of nuclear power plants of similar design is given in Table 7.1-1. Table 7.1-1 also identifies the systems that are different with references to discussions of those differences.

The plant's control and instrumentation systems are grouped into the following categories:

a. Reactor trip system (RTS)

9.1.4.1 Design Bases

9.1.4.1.1 Safety Design Bases

The portions of the FHS that are safety related are the containment isolation features of the fuel transfer tube and the crane structural components which prevent falling of major crane components onto fuel assemblies or safe shutdown equipment.

SAFETY DESIGN BASIS ONE - The FHS is protected from the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, and external missiles (GDC-2).

SAFETY DESIGN BASIS TWO - The FHS is designed to remain intact after an SSE or following the postulated hazards of fire, internal missiles, or pipe breaks (GDC-3 and 4).

SAFETY DESIGN BASIS THREE - The FHS components are capable of being tested during plant operation. Provisions are made to allow for inservice inspection and testing of components at appropriate times.

SAFETY DESIGN BASIS FOUR - The FHS is designed and fabricated to codes consistent with the seismic category assigned by Regulatory Guide 1.29 and industry standard specifications.

SAFETY DESIGN BASIS FIVE - The containment isolation provisions for the system are selected, tested, and located in accordance with the requirements of GDC-54 and 10 CFR 50, Appendix J, Type B testing.

50.67 and Regulatory Guide 1.183

SAFETY DESIGN BASIS SIX - The FHS is designed and arranged so that there are no loads which, if dropped, could result in damage, leading to the release of radioactivity in excess of 10 CFR ~~100~~ guidelines, or impair the capability to safely shut down the plant. Specific administrative controls for handling of the spent fuel pool transfer gates are addressed in Section 9.1.4.3.

This meets the requirements of Regulatory Guide 1.13, as described in Table 9.1-3.

9.1.4.2 System Description

9.1.4.2.1 General Description

The fuel handling system consists of the equipment needed to refuel the reactor core. Basically, this equipment is composed of cranes, handling equipment, and a fuel transfer system.

The associated fuel handling structures are divided into seven areas. In general, these areas are:

SAFETY EVALUATION ONE - The safety-related portions of the FHS are located in the reactor and fuel buildings. These buildings are designed to withstand the effects of earthquakes, tornadoes, hurricanes, floods, external missiles, and other appropriate natural phenomena. Sections 3.3, 3.4, 3.5, 3.7(B), and 3.8 provide the bases for the adequacy of the structural design of these buildings.

SAFETY EVALUATION TWO - The safety-related portions of the FHS are designed to remain intact after an SSE. Section 3.7(B) provides the design loading conditions that were considered. Sections 3.5, 3.6, and 9.5.1 provide the required hazards analysis.

SAFETY EVALUATION THREE - The FHS is initially tested with the program given in Chapter 14.0. Periodic inservice functional testing is done in accordance with Section 9.1.4.4. The fuel transfer tube is inspected in accordance with the technical requirements of ASME Section XI.

SAFETY EVALUATION FOUR - Section 3.2 delineates the seismic category applicable to the safety-related portions of this system.

50.67 and Regulatory Guide 1.183

SAFETY EVALUA ... ovide the safety evaluation for the system containment isolation arrangement and testability.

SAFETY EVALUATION SIX - In the event of a fuel handling accident in the fuel building, the radiological consequences analyzed in Chapter 15.0 demonstrate that the 10 CFR Part ~~100~~ guideline values are not exceeded. The circumstances resulting in a handling accident are limited to the following conditions.

a. Fuel drop from a lifting device

b. Improper operation of the transfer equipment and cranes

c. DELETED

d. Drop of the RV head

The fuel handling equipment is designed to prevent a fuel assembly drop by providing special gripping devices which are locked in a manner which will not allow the release of the fuel assembly during transfer. The special features are described in Section 9.1.4.2.2.

Improper operation of the fuel transfer system is prevented by the location of special limit switches and interlocks which will not allow the movement of fuel assemblies unless they are properly oriented, thus avoiding a fuel handling accident. Further description of these devices is given in Section 9.1.4.2.2.

Limit switches and interlocks located on the fuel handling cranes in conjunction with administrative controls prevent any improper operations which may result in a fuel

TABLE 9.1-3 DESIGN COMPARISION TO REGULATORY POSITIONS OF REGULATORY GUIDE 1.13 REVISION 1, DATED DECEMBER 1975, TITLED "SPENT FUEL STORAGE FACILITY DESIGN BASIS"

Regulatory Guide 1.13 Position	Union Electric
1. The spent fuel storage facility (including its structures and equipment, except as noted in Paragraph 6 below) should be designed to Category I seismic requirements.	1. Complies as described in Section 9.1.2.1.1
2. The facility should be designed (a) to keep tornadic winds and missiles generated by these winds from the fuel storage pool and (b) to keep missiles generated by tornadic winds from contacting fuel within the pool.	2. Complies as described in Section 3.5, and 3.8.
3. Interlocks should be provided to prevent cranes from passing over stored fuel (or near stored fuel in a manner such that if a crane failed the load could tip over on stored fuel) when fuel handling is not in progress. During fuel handling operations, the interlocks may be bypassed and administrative control used to prevent the crane from carrying loads that are not necessary for fuel handling over the stored fuel or other prohibited areas. The facility should be designed to minimize the need for bypassing such interlocks.	3. Complies as described in Section 9.1.4.
4. A controlled leakage building should enclose the fuel pool. The building should be equipped with an appropriate ventilation and filtration system to limit the potential release of radioactive iodine and other radioactive materials. The building need not be designed to withstand extremely high winds, but leakage should be suitably controlled during refueling operations. The design of the ventilation and filtration system should be based on the assumption that the cladding of all of the fuel rods in one fuel bundle might be breached. The inventory of radioactive materials available for leakage from the building should be based on the assumptions given in ~~Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors" (Safety Guide 25)~~.	4. Complies as described in Section 9.4.2 and 15.7.4.

Replace with:
USNRC, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants", July 2000

building ventilation system is designed to provide fresh air ventilation at a minimum rate of 0.1 cfm per square foot of floor area.

POWER GENERATION DESIGN BASIS TWO - The control building exhaust system serves to remove from the control building the hydrogen generated by the batteries during normal operation.

POWER GENERATION DESIGN BASIS THREE - The access control air-conditioning system provides RP access control areas, and the nonvital electric equipment areas of the electrical and mechanical equipment level with an environment suitable for personnel comfort and electrical equipment operation.

POWER GENERATION DESIGN BASIS FOUR - The access control exhaust system collects and processes the effluents from the potentially contaminated regions of the access control area. The exhaust system is designed to meet the requirements of the discharge concentration limits of 10 CFR 20 and the as-low-as-reasonably-achievable dose objective of 10 CFR 50, Appendix I. The access control exhaust system charcoal adsorption train complies with Regulatory Guide 1.140, to the extent discussed in Table 9.4-3.

POWER GENERATION DESIGN BASIS FIVE - The counting room recirculation system provides adequate cooling, humidity control, and filtering of the counting room environment for personnel and equipment.

POWER GENERATION DESIGN BASIS SIX - A supplemental cooling train is provided for each Class 1E electrical equipment air-conditioning system train. During periods when one Class 1E electrical equipment air-conditioning system train is unavailable or removed from service for a limited period of time (such as for online maintenance), the supplemental cooling train provides additional ventilation via forced ventilation flowpaths that enable the other functional Class 1E electrical equipment air conditioning train to provide cooling to the rooms and areas for both trains of Class 1E electrical equipment.

9.4.1.2 System Description

9.4.1.2.1 General Description

The control building HVAC systems are shown in Figure 9.4-1. The systems consist of the control building supply system, control room air-conditioning system with supplemental filtration and pressurization systems, Class 1E electrical equipment air-conditioning system, access control air-conditioning system, counting room recirculation system, control building exhaust system, and the access control exhaust system. The design conditions for these systems are presented in Table 3.11(B)-1. Potential radiation doses in the control room are discussed in Chapter 15.0.

The control building is serviced by an outside-air-supply system which provides fresh cooled or heated air to each of the various levels of the building. Self-contained

The model for control room dose analysis is presented in Appendix 15A.

The indication of the amount of filter loading for all filters associated with the essential and nonessential air handlers is provided at each of the air handlers.

Alarms are provided in the control room to indicate high charcoal bed temperatures in the control room filtration, control room pressurization and access control filtration units and high room temperature in the ESF switchgear and dc switchgear rooms.

An alarm is provided in the control room to indicate high hydrogen concentrations in a battery room.

Alarms are provided in the control room to indicate high carbon monoxide/carbon dioxide concentrations, high radiation, and smoke in the control building intake.

All instrumentation provided with the filtration units is as required by Regulatory Guide 1.52 or 1.140, as applicable.

9.4.2 FUEL BUILDING HVAC

The fuel building ventilation system consists of the fuel building supply system which includes the fuel building heating coil, the fuel building supply air unit, and the fuel handling area cooling coil; the emergency exhaust system, including the emergency exhaust heating coil; the auxiliary/fuel building normal exhaust system; the spent fuel pool cooling pump room coolers; and the unit heaters. Since both the emergency exhaust system and the auxiliary/fuel building normal exhaust system also serve the auxiliary building, their operation in th release building is discussed in Section 9.4.3.

The fuel building supply system provides conditioned outside air for ventilation and cooling or heating, as required, to all areas of the fuel building. The auxiliary/fuel building normal exhaust system exhausts air from the area above the spent fuel pool during normal operation and provides a means of purging smoke following a postulated fire.

In the event of a fuel handling accident, the emergency exhaust system collects and processes the airborne ~~particulates~~ in the fuel building. In the event of a LOCA, the emergency exhaust system processes the atmosphere of the auxiliary building.

The fuel storage pool cooling pump room coolers provide a suitable ambient temperature for the electric motor drives of the safety-related pumps.

The fuel building unit heaters provide supplemental heating for the fuel building, when required.

and is also credited (based upon manual actuation) to insure any release from containment to the auxiliary building via an open personnel hatch is transported to the plant vent for release.

SAFETY EVALUATION FOUR - The fuel storage pool cooling pump room coolers, the emergency exhaust system, and the fuel building HVAC boundary penetration isolation provisions are initially tested with the program given in Chapter 14.0. Periodic inservice functional testing is done in accordance with Section 9.4.2.4.

The exhaust system is also credited (based upon manual actuation) during a fuel handling accident in containment to insure any release from containment to the auxiliary building via an open personnel hatch is transported to the plant vent for release.

, Section XI pump room coolers.

SAFETY EVALUATION FIVE - Section 3.2 delineates the quality group classification and seismic category applicable to the safety-related portion of this system and supporting system. All the power supplies and control function necessary for safe function of the fuel storage pool cooling pump room coolers, emergency exhaust system, and the fuel building HVAC boundary penetration isolation provisions are Class 1E, as described in Chapters 7.0 and 8.0.

SAFETY EVALUATION SIX - Section 9.4.2.2.3 describes the provisions made to assure the isolation of the auxiliary building following a DBA.

SAFETY EVALUATION SEVEN - The emergency exhaust system maintains a negative pressure of no less than 1/4 in. w.g. in the fuel building to prevent unprocessed exfiltration following a fuel handling accident which releases radioactivity. The eme 50.67 and Regulatory Guide 1.183 radioactivity downstream of the filter ads filter adsorber unit limits the radiological consequences of a fuel handling accident to less than 10 CFR ~~100~~ limits.

SAFETY EVALUATION EIGHT - Room coolers are installed in each fuel storage pool cooling pump room and are designed to maintain these rooms below 122°F (50°C), based on maximum heat load within the room.

9.4.2.4 Tests and Inspections

Preoperational testing is described in Chapter 14.0.

Filters and adsorbers for the emergency exhaust system are tested in the manufacturer's shop, after initial installation and subsequent to each filter or adsorber change. After installation, interim tests and inspections will be performed after every 720 hours of operation and once per 18 months in accordance with the requirements of Regulatory Guide 1.52 and ASTM D3803-1989 as discussed in Table 9.4-2, to detect any deterioration of components that may develop under service or standby conditions.

Prefilters will not undergo factory or inplace testing since no credit is taken for removal of particulates in meeting permissible dose rates.

HEPA filters will be factory tested with monodispersed DOP aerosol to demonstrate a minimum particulate removal efficiency of no less than 99.97 percent for 0.3 micron

Depending on the equipment in the compartments, the access varies from Zones B through E. Corridors are shielded to allow Zone B access, and operator areas for valve compartments are limited to Zone C access.

Removable sections of block shield walls and concrete plugs are utilized to replace worn-out equipment and spent filter cartridges, respectively. Partial shield walls are placed between equipment in compartments with more than one piece of equipment to permit maintenance access.

12.3.2.2.5 Turbine Building Shielding Design

Radiation shielding is not required for any process equipment located in the turbine building. All areas in the turbine building are classified Zone A.

12.3.2.2.6 Control Room Shielding Design

The design basis LOCA dictates the shielding requirements for the control room. Shielding is provided to permit access and occupancy of the control room under LOCA conditions with radiation doses limited to 5 rem ~~whole body~~ TEDE from all contributing modes of exposure for the duration of the accident, in accordance with GDC-19. A complete discussion of control room habitability during a LOCA is provided in Section 6.4. Figure 12.3-3 provides an isometric view of the control room shielding.

12.3.2.2.7 Diesel Generator Building Shielding Design

There are no radiation sources in the diesel generator building. Therefore, no shielding is required within the building.

12.3.2.2.8 Miscellaneous Plant Areas and Plant Yard Areas

Sufficient shielding is provided for all plant buildings containing radiation sources so that radiation levels at the accessible outside surfaces of the buildings are maintained below Zone A levels. Plant yard areas which are frequently occupied by plant personnel are fully accessible during normal operation and shutdown. These areas are surrounded by a security fence and closed off from areas accessible to the general public. Access to outside storage tanks which have a contact dose rate greater than 0.5 mrem/hr is restricted by a fence located at a distance at which the dose rate is less than 0.5 mrem/hr.

12.3.2.2.9 Independent Spent Fuel Storage Installation (ISFSI) Shielding Design

The ISFSI is designed for interim dry storage of spent nuclear fuel. The shielding is sufficient to comply with the requirements of 10 CFR 72.104 and 10 CFR 72.106. The dose rate on the VVM closure lid is calculated to be 0.25 mrem/hr neutron and 0.69 mrem/hr gamma for a total dose rate of 0.94 mrem/hr. The dose rate on the outlet duct screen is calculated to be 0.80 mrem/hr neutron and 1.32 mrem/hr gamma for a total

12.3.3.1 Design Objectives

The plant HVAC systems for normal plant operation and anticipated operational occurrences are designed to meet the requirements of 10 CFR 20, "Standards for Protection Against Radiation," and 10 CFR 50, "Licensing of Production and Utilization Facilities."

12.3.3.2 Design Criteria

,10 CFR 50.67 and Regulatory Guide 1.183, as appropriate

Design criteria for the plant HVAC systems include the following:

a. During normal operation and anticipated operational occurrences, the average and maximum airborne radioactivity levels to which plant personnel are exposed in the restricted areas of the plant are ALARA and within the limits specified in 10 CFR 20.

b. During normal operation and anticipated operational occurrences, the dose from concentrations of airborne radioactive material in unrestricted areas beyond the site boundary will be ALARA and within the limits specified in 10 CFR 20 and 10 CFR 50.

c. The plant siting dose guidelines of 10 CFR 100 will be satisfied, following those hypothetical accidents described in Chapter 15.

d. The dose to control room personnel shall not exceed the limits specified in GDC-19, following those hypothetical accidents described in Chapter 15.0 and Section 6.4.

12.3.3.3 Design Guidelines

In order to accomplish the design objectives, the following guidelines are followed, wherever practicable.

12.3.3.3.1 Guidelines to Minimize Airborne Radioactivity

a. Access control and traffic patterns are considered in the basic plant layout to minimize the spread of contamination.

b. Equipment vents and drains are piped directly to a collection device connected to the collection system, instead of allowing any contaminated fluid to flow across the floor to the floor drain.

c. All-welded piping systems are employed on contaminated systems, to the maximum extent practicable, to reduce system leakage. If welded piping systems are not employed, drip trays are provided at the points of potential leakage. Drains from drip trays are piped directly to the collection system.

e. The clear space for doors is a minimum of 3 feet by 7 feet.

f. The filters are designed with replaceable 2 feet by 2 feet units that are clamped in place against compression seals. The filter housing is designed, tested, and proven to be airtight with bulkhead type doors that are closed against compression seals.

, 10 CFR 50.67 and Regulatory Guide 1.183, as appropriate

12.3.4 AREA RA ... ORING INSTRUMENTATION

12.3.4.1 Area Radiation Monitoring

The area radiation monitoring system (ARMS) is provided to supplement the personnel and area radiation survey provisions of the plant radiation protection program described in Section 12.5 and to ensure compliance with the personnel radiation protection guidelines of 10 CFR 20, 10 CRF 50, 10 CFR 70, and Regulatory Guides 8.2, 8.8, and 8.12.

12.3.4.1.1 Design Bases

The principal objectives and criteria of the ARMS are provided below.

12.3.4.1.1.1 Safety Design Bases

The area radiation monitoring system has no function related to the safe shutdown of the plant or the capability to mitigate the consequences of accidents that could result in offsite exposures comparable to the guideline exposure of 10 CFR 100 and, therefore, has no safety design bases. See Appendix 7A for a discussion of Regulatory Guide 1.97.

12.3.4.1.1.2 Power Generation Design Bases

POWER GENERATION DESIGN BASIS ONE - The ARMS functions continuously to immediately alert plant personnel entering or working in nonradiation or low-radiation areas of increasing or abnormally high radiation levels which, if unnoticed, could possibly result in inadvertent overexposures.

POWER GENERATION DESIGN BASIS TWO - The ARMS serves to inform the control room operator of the occurrence and approximate location of an abnormal radiation increase in nonradiation or low-radiation areas.

POWER GENERATION DESIGN BASIS THREE - The ARMS complies with the requirements of 10 CFR 50, Appendix A, General Design Criterion 63 for monitoring fuel and waste storage and handling areas.

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15A-3 Core Inventory (Ci) and WGDTR Fuel and Rod Gap Activties

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Spent fuel cask drop accidents are not applicable to Callaway Plant. The spent fuel cask handling equipment has been upgraded to single-failure-proof status to provide the maximum practical defense in depth in accordance with NUREG-0612 and to allow the use of the spent fuel cask handling equipment and lifting devices to handle heavy loads in the vicinity of spent fuel without the need for load drop analyses. This is supported by NRC Information Notice 99-15 which stated in general that for cask movements with single-failure-proof cranes, cask drops or tipping accidents need not be considered. Since the cask cannot drop, no cask rupture can occur and thus no radioactivity can be released.

15.0.1.4 Condition IV - Limiting Faults

Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. They are the most drastic which must be designed against and represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR ~~100~~ 50.67 or Regulatory Guide 1.183. A single Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault, including those of the emergency core cooling system and the containment. For the purposes of this report the following faults have been classified in this category:

a. Steam system pipe break.

b. Feedwater system pipe break.

c. Reactor coolant pump shaft seizure (locked rotor).

d. Reactor coolant pump shaft break.

e. Spectrum of rod cluster control assembly ejection accidents.

f. Steam generator tube rupture.

g. Loss-of-coolant accidents, resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (large break).

h. Design basis fuel handling accidents.

15.0.2 OPTIMIZATION OF CONTROL SYSTEMS

15.0.2.1 Setpoint Study

A control system setpoint study is performed in order to simulate performance of the reactor control and protection systems. In this study, emphasis is placed on the

The secondary power is obtained from the measurement of steam or feedwater flow, feedwater inlet temperature to the steam generators, and steam pressure. High accuracy instrumentation is provided for use during these measurements. Accuracy tolerances meet or exceed requirements established by the safety analysis.

15.0.8 PLANT SYSTEMS AND COMPONENTS AVAILABLE FOR MITIGATION OF ACCIDENT EFFECTS

The plant is designed to afford protection against the possible effects of natural phenomena, postulated environmental conditions, and dynamic effects of the postulated accidents. In addition, the design incorporates features which minimize the probability and effects of fires and explosions. The Operating Quality Assurance Manual discusses the quality assurance program which has been implemented to assure that the NSSS will satisfactorily perform its assigned safety functions. The incorporation of these features in the plant, coupled with the reliability of the design, ensures that the normally operating systems and components listed in Table 15.0-6 will be available for mitigation of the events discussed in Chapter 15.0. In determining which systems are necessary to mitigate the effects of these postulated events, the classification system of ANSI-N18.2-1973 is utilized. The design of safety-related systems (including protection systems) is consistent with IEEE Standard 379-1972 and Regulatory Guide 1.53, in the application of the single failure criterion.

In the analysis of the Chapter 15.0 events, control system action is considered only if that action results in more severe accident results. No credit is taken for control system operation if that operation mitigates the results of an accident. For some accidents, the analysis is performed both with and without control system operation to determine the worst case. The pressurizer heaters are generally assumed not to be energized for the analysis of the Chapter 15 events. Operation of the pressurizer heaters as a result of normal control action or a single failure will be less conservative or have negligible effects for most analyses. Therefore, unless it is shown that such a control action results in more limiting results or more severe consequences, the control action of the pressurizer heaters is not modeled for the analyses performed in Chapter 15. Any exceptions are noted in the text describing the individual analysis assumptions.

15.0.9 FISSION PRODUCT INVENTORIES

~~The calculation of the core fission product inventory employs the ORIGEN 2 computer code modelling a three region enveloping cycle core with a core power level of 3636 MWt (3565 MWt plus 2% postulated calorimetric error). Of the 96 assemblies in core Region 1, 32 have operated at a specific power of 50.7 MW/MTU for 474 days and 64 have operated at a specific power of 57.0 MW/MTU for 474 days. Of the 88 assemblies in core Region 2, 24 have operated at a specific power of 48.5 MW/MTU for 474 days and at 44.3 MW/MTU for 474 days and 64 have operated at a specific power of 52.7 MW/MTU for 474 days and at 33.8 MW/MTU for 474 days. The 9 assemblies in core Region 3 have operated at a specific power of 50.6 MW/MTU for 474 days, at 38.0 MW/MTU for 474 days, and at 21.1 MW/MTU for 474 days. The average burnups in Regions~~

Insert 15.0.9

~~1, 2, and 3 at the end of a cycle (MWD/MTU) are 26,000, 42,000, and 52,000, respectively. The isotopic yields utilize data for fissioning of U-235, U-238, and Pu-239 and account for the depletion of U-235. Radiological consequences are evaluated with source terms based on the 3636 MWt core rating (Table 15A-3), Callaway-specific meteorology based on three years of combined meteorological data (Table 15A-2), and appropriate dose conversion factors (Table 15A-4).~~

15.0.10 RESIDUAL DECAY HEAT

15.0.10.1 Total Residual Heat

Residual heat in a subcritical core is calculated for the LOCA per the requirements of Appendix K of 10 CFR 50.46, as described in References 11 and 12. These requirements include assuming infinite irradiation time before the core goes subcritical to determine fission product decay energy. For all other accidents, the same models are used, except that fission product decay energy is based on core average exposure at the end of the equilibrium cycle.

15.0.10.2 Distribution of Decay Heat Following Loss-of-Coolant Accident

During a LOCA, the core is rapidly shut down by void formation or RCCA insertion, or both, and a large fraction of the heat generation to be considered comes from fission product decay gamma rays. This heat is not distributed in the same manner as steady state fission power. Local peaking effects which are important for the neutron dependent part of the heat generation do not apply to the gamma ray contribution. The steady state factor of 97.4 percent, which represents the fraction of heat generated within the clad and pellet, drops to 95 percent for the hot rod in a LOCA.

For example, consider the transient resulting from the postulated double ended break of the largest reactor coolant system pipe; 1/2 second after the rupture about 30 percent of the heat generated in the fuel rods is from gamma ray absorption. The gamma power shape is less peaked than the steady state fission power shape, reducing the energy deposited in the hot rod at the expense of adjacent colder rods. A conservative estimate of this effect is a reduction of 10 percent of the gamma ray contribution or 3 percent of the total. Since the water density is considerably reduced at this time, an average of 98 percent of the available heat is deposited in the fuel rods, the remaining 2 percent being absorbed by water, thimbles, sleeves, and grids. The net effect is a factor of 0.95, rather than 0.974, to be applied to the heat production in the hot rod.

15.0.11 COMPUTER CODES UTILIZED

Summaries of some of the principal computer codes used in transient analyses are given below. The codes used in the analyses of each transient have been listed in Table 15.0-2.

local condition heat transfer. Component models include a two region nonequilibrium pressurizer, centrifugal and jet pumps, valves, non-conducting heat exchangers, steam separators, and turbine. An automatic steady state initialization procedure is also available.

Insert 15.0.11.8

References 18 and 23

The ~~RETRAN code is~~ discussed in ~~Reference 18~~.

RETRAN-02 and RETRAN-3D codes are

15.0.11.9 VIPRE

The VIPRE computer program performs thermal-hydraulic calculations. The code calculates coolant density, mass velocity, enthalpy, void fractions, static pressure and DNBR distributions along flow channels within a reactor core.

The VIPRE code is described in Reference 19.

15.0.11.10 ANC

ANC is an advanced nodal code capable of two-dimensional and three-dimensional neutronics calculations. ANC is the reference model for certain safety analysis calculations, power distributions, peaking factors, critical boron concentrations, control rod worths, reactivity coefficients, etc. In addition, three-dimensional ANC validates one-dimensional and two-dimensional results and provides information about radial (x-y) peaking factors as a function of axial position. It can calculate discrete pin powers from nodal information as well.

The ANC code is described in Reference 20.

15.0.12 LIMITING SINGLE FAILURES

The most limiting single failure as described in Section 3.1 of safety-related equipment, where one exists, is identified in each analysis description, and the consequences of this failure are described therein. In some instances, because of redundancy in protection equipment, no single failure which could adversely affect the consequences of the transient has been identified. The failure assumed in each analysis is listed in Table 15.0-7.

15.0.13 OPERATOR ACTIONS

For most of the events analyzed in Chapter 15.0 the plant will be in a safe and stable hot standby condition following the automatic actuation of reactor trip. This condition will, in fact, be similar to plant conditions following any normal, orderly shutdown of the reactor. At this point, the actions taken by the operator would be no different than normal operating procedures. The exact actions taken, and the time at which these actions would occur, will depend on what systems are available (e.g., turbine bypass system, main feedwater system, etc.) and the plans for further plant operation. As a minimum, to maintain the hot stabilized condition, decay heat must be removed via the steam

1) Each Level channel is tested one-at-a-time during the level channel testing with zero time delay as describd in the WCAP.

2) The TTD function and timers discussed in Reference 15 are no longer applicable in Callaway.

3) Section 3.6.2.2 is titled OUTAGE TESTING. The PROM logic modules and EAM testing described under this section may be performed on-line and not restricted to performance during outages.

16. RETRAN-02 -- A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," Electric Power Research Institute, EPRI NP-1850-CCM-A, Rev. 2, 1984.

17. Letter from Cecil O. Thomas (NRC) to Dr. Thomas W. Schnatz, Utility Group for Regulatory Applications (UGRA), "Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, 'RETRAN - A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems,' and EPRI NP-1850-CCM, 'RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems,'" dated September 2, 1984.

18. D.S. Huegel, et. al., WCAP-14882-P-A (Proprietary)/WCAP-15234-A (Non-proprietary), "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analysis," April 1999.

19. Y.X. Sung, et. al., WCAP-14565-P-A (Proprietary)/WCAP-15306-A (Non-proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

20. Y.S. Liu, et. al., WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Compter Code," September 1986.

21. Westinghouse Letter SCP-07-17, "Callaway Plant Engineering Report and Guidelines in Support of End of Cycle 15 T_{avg} Coastdown, Revision 1," dated February 9, 2007.

22. ORNL/TM-2005/39, Oak Ridge National Lab, SCALE 6.1 Package Manual, June 2011.

23. M. P. Paulsen et. al., NP--7450(A), "RETRAN-3D -- A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," , Research Project 889-10, EPRI, Rev. 11, May 2017

CALLAWAY - SP

TABLE 15.0-2 (Sheet 6)

	EVENT	REACTOR COOLANT PUMP HEAT (MWt)	REACTOR VESSEL COOLANT FLOW (gpm)	VESSEL T-AVG (°F)	PRESSURIZER PRESSURE (PSIA)	PRESSURIZER WATER LEVEL (% span)	FEEDWATER TEMP (°F)	EQUIVALENT S/G TUBE PLUGGING LEVEL	FULL POWER STEADY STATE FΔH	F_Q
15.5	Increase in coolant inventory									
	Inadvertent ECCS operation at power (DNB Case)	20	382,630	588.4	2250	65	446	5%	NA	NA
	(Pzr. Filling Case)	20	374,400	567.2	2190	43	446	5%	NA	NA
	CVCS malfunction	See Section 15.5.2 for all asumptions								
15.6	Decrease in coolant inventory									
	Inadvertent RCS depressurization	14	382,630	588.4	2250	60	446	5%	NA	NA
	S/G tube rupture									
	ASD faillure case	14	374,400	592.7	2280	60	446	0%	NA	NA
	Overfill ~~Overall~~ case	14	374,400	567.7	2280	38	390	5%	NA	NA
	Loss of coolant accidents	See Section 15.6.5 for all asumptions								

++++RETRAN Option 1 Film Boiling Correlation

NOTES:
1. Deleted
2. 2250 psia used in offsite dose evaluation

TABLE 15.0-7 SINGLE FAILURES ASSUMED IN ACCIDENT ANALYSES

Event Description	Worst Failure Assumed
Feedwater temperature reduction	One protection train
Excessive feedwater flow	One protection train
Excessive steam flow	(1)(3)
Inadvertent secondary depressurization	(3)
Steam system piping failure	One safety injection train
Steam pressure regulator malfunction	(2)
Loss of external load	One protection train
Turbine trip	One protection train
Inadvertent closure of MSIV	One protection train
Loss of condenser vacuum	One protection train
Loss of ac power	Turbine driven auxiliary feedwater pump
Loss of normal feedwater	Turbine driven auxiliary feedwater pump
Feedwater system pipe break	Turbine driven auxiliary feedwater pump
Partial loss of forced reactor coolant flow	One protection train
Complete loss of forced reactor coolant flow	One protection train
RCP locked rotor	One protection train
RCP shaft break	One protection train
RCCA bank withdrawal from subcritical	One protection train
RCCA bank withdrawal at power	One protection train
Dropped RCCA, dropped RCCA bank	(1)
Statically misaligned RCCA	(3)
Single RCCA withdrawal	One protection train
Inactive RC pump startup	(3)
Flow controller malfunction	(2)
Uncontrolled boron dilution	Standby ECCS charging pump is operating (Modes 1 and 2), one source range NIS channel (Modes 3-5)
Improper fuel loading	(3)
RCCA ejection	One protection train
Inadvertent ECCS operation at power	One protection train
Increase in RCS inventory	One pressurizer level channel
BWR transients	(2)
Inadvertent RCS depressurization	One protection train
Failure of small lines carrying primary coolant outside containment	(3)
SGTR	One SG atmospheric steam dump valve
BWR piping failures	(2)

ASD failure

SGTR, Overfill — Auxiliary Feedwater Flow Control Valve

15.1.5 STEAM SYSTEM PIPING FAILURE

15.1.5.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steamline would result in an initial increase in steam flow that decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. If the most reactive RCCA is assumed stuck in its fully withdrawn position after reactor trip, there is possibility that the core will become critical and return to power. A return to power following a steamline rupture is a potential problem mainly because of the high power peaking factors which exist, assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by 50.67 and Regulatory Guide 1.183 e ECCS.

The analysis of a main steamline rupture is performed to demonstrate that the following criteria are satisfied:

> Assuming a stuck RCCA with or without offsite power, and assuming a single failure in the engineered safety features, the core remains in place and intact. Radiation doses do not exceed the guidelines of 10 CFR ~~100~~.
>
> Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that no DNB occurs for any rupture, assuming the most reactive RCCA stuck in its fully withdrawn position. The DNBR design basis is discussed in Section 4.4.

A major steamline rupture is classified as an ANS Condition IV event. See Section 15.0.1 for a discussion of Condition IV events.

Effects of minor secondary system pipe breaks are bounded by the analysis presented in this section. Minor secondary system pipe breaks are classified as Condition III events, as described in Section 15.0.1.3.

The major rupture of a steamline is the most limiting cooldown transient, and is analyzed at zero power with no decay heat. Decay heat would retard the cooldown, thereby reducing the return to power. A detailed analysis of this transient with the most limiting break size, a double ended rupture, is presented here. The assumptions used in this analysis are discussed in Reference 3. Reference 3 also contains a discussion of the spectrum of break sizes and power levels analyzed.

During startup or shutdown evolutions when safety injection on low pressurizer pressure or low steamline pressure is blocked and steamline isolation on low steamline pressure is blocked below P-11 (pressurizer pressure less than 1970 psig), the high negative steamline pressure rate (HNPR) signal is enabled by P-11 to provide steamline isolation. A series of steamline break sensitivities in Mode 3 conditions has been

Margin to Critical Heat Flux

A DNB analysis was performed for both of these cases. It was found that both cases had a minimum DNBR greater than the safety analysis limit value as discussed in Section 4.4.1.1. The WLOP DNB correlation was used in this analysis (Reference 12). Historically, the W-3 DNB correlation had been used; see Reference 5 for the justification discussing the use of the W-3 correlation for low pressure applications, accepted by the NRC in Reference 6.

15.1.5.3 Radiological Consequences

;however, the Reactor Coolant Pumps are assumed to remain on for the steam release input to radiological consequence analysis.

15.1.5.3.1 Method Of Analysis

15.1.5.3.1.1 Physical Model

The radiological consequences of a MSLB inside the containment are less severe than the one outside the containment because the radioactivity released will be held up inside the containment, allowing decay and plateout of the radionuclides. To evaluate the radiological consequences due to a postulated MSLB (outside the containment), it is assumed that there is a complete severance of a main steamline outside the containment.

It is also assumed that there is a simultaneous loss of offsite power, resulting in reactor coolant pump coastdown. The ECCS is actuated and the reactor trips.

The main steam isolation valves, their bypass valves, and the steamline drain valves isolate the steam generators and the main steamlines upon a signal initiated by the engineered safety features actuation system under the conditions of high negative steamline pressure rates, low steamline pressure, or high containment pressures. The main steam isolation valves are installed in the main steamlines from each steam generator downstream from the safety and atmospheric relief valves outside the containment. The break in the main steamline is assumed to occur outside of the containment. The affected steam generator (steam generator connected to a broken steamline) blows down completely. The steam is vented directly to the atmosphere.

Each of the steam generators incorporates integral flow restrictors, which are designed to limit the rate of steam blowdown from the steam generators following a rupture of the main steamline. This, in turn, reduces the cooling rate of the reactor coolant system thereby reducing the return to power.

In case of loss of offsite power, the remaining steam generators are available for dissipation of core decay heat by venting steam to the atmosphere via the atmospheric relief and safety valves. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently so that the RHR system can be utilized to cool the reactor. The MSSVs release steam at their individual set pressures and are not capable of supporting a controlled plant cooldown to RHR entry conditions. Although the MSSVs would be available following a MSLB for SG over-pressurization protection if needed, they do not have a safety function to mitigate a MSLB or to cool down the plant. Plant cooldown to RHR entry conditions is supported by the ASDs. Section 10.3.3 SAFETY EVALUATION SEVEN provides more details.

15.1.5.3.1.2 Assumptions and Conditions

The major assumptions and parameters assumed in the analysis are itemized in Tables 15.1-3 and 15A-1.

Tables 15B-1 and 15B-4 provide a comparison of the analysis to the guidelines in Regulatory Guide 1.183.

The assumptions used to determine the concentrations of radioactive isotopes within the secondary system for this accident are as follows:

a. The initial secondary side radio-iodine concentrations are assumed tobe 10% of the initial Case 1 primary sideconcentrations.

b. A primary-to-secondary leakage rate of 1 gpm is assumed to exist and is assumed to be in the ~~affected~~ steam generator.

faulted

c. The reactor coolant initial iodine activity is determined by two methods, and both cases are analyzed. These are:

Insert 15.1.5.3.1.2A

~~Case 1 - The Case 1 initial radio-iodine concentrations are the same as the Case 1 concentrations used for the Steam Generator Tube Rupture accident sequence. Refer to Table 15.6-4.~~

~~Case 2 - The Case 2 initial radio-iodine concentrations are the same as the Case 2 concentrations used for the Steam Generator Tube Rupture accident sequence. Refer to Table 15.6-4.~~

Insert 15.1.5.3.1.2B

d. The initial reactor coolant concentrations of noble gas correspond to ~~1-percent fuel defects.~~ 225 μCi/gm DOSE EQUIVALENTXE-133~~)~~.

e. Partition factors used to determine the secondary system activities are given in Table 15.1-3.

for iodines and alkali metals

The following specific assumptions and parameters are used to calculate the activity release:

a. Offsite power is lost, resulting in reactor coolant pump coastdown.

;however, the Reactor Coolant Pumps are assumed to remain on for the steam release input to radiological consequence analysis.

b. No condenser air removal system release and no normal operating steam generator blowdown is assumed to occur during the course of the accident.

Approximately eight (7.899)

c. ~~Eight~~ hours after the occurrence of the accident, the residual heat-removal system (RHRS) starts operation to cool down the plant.

cooled to 212°F such that there would be no flashing of the leaked fluid
22

d. After the accident, the primary-to-secondary leakage continues for ~~8~~ hours, at which time the reactor coolant system is ~~depressurized~~.

faulted

e. The ~~affected~~ steam generator (steam generator connected to the broken steamline) is allowed to blow down completely.

on the intact steam generators
approximately 8

f. Steam release to the atmosphere and the associated activity release from the safety and relief valves ~~and the broken steamline~~ is terminated ~~8~~ hours after the accident, when the RHRS is activated to complete cooldown.

and alkali metal
faulted

g. The amount of noble gas activity released is equal to the amount present in the reactor coolant, which leaks to the secondary during the accident. The amount of iodine activity released is based on the activity present in the secondary system and the amount of leaked reactor coolant which is entrained in the steam that is discharged to the environment via the safety and relief valves and the broken steamline. Partition factors used for the

intact

~~unaffected~~ steam generators after the accident occurs are given in Table 15.1-3. An iodine partition factor of 1 is used for the ~~affected~~ steam generator. After the break, all primary-to-secondary leakage is assumed to be through the faulted steam generator.

h. The activity released from the broken steamline and the safety and relief valves during the ~~8-hour duration of the~~ accident is immediately vented to the atmosphere.

15.1.5.3.1.3 Mathematical Models Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are based on the assumptions listed above.

b. The atmospheric dispersion factors used in the analysis were calculated based on the onsite meteorological measurement programs described in Section 2.3 of the Site Addendum.

TEDE
,

c. The ~~thyroid inhalation dose and total-body gamma immersion~~ doses to a receptor at the exclusion area boundary ~~and~~ outer boundary of the low-population zone were analyzed, using the models described in Appendix 15A.

, and in the control room

15.1.5.3.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

For evaluating the radiological consequences due to a postulated MSLB, the activity released from the ~~affected~~ steam generator (steam generator connected to the broken steamline) is released directly to the environment. The ~~unaffected~~ steam generators are

faulted
intact

assumed to continually discharge steam and entrained activity via the safety and relief valves up to the time initiation of the RHRS can be accomplished.

Since the activity is released directly to the environment with no credit for plateout or retention, the results of the analysis are based on the most direct leakage pathway available. Therefore, the resultant radiological consequences represent the most conservative estimate of the potential integrated dose due to the postulated MSLB.

15.1.5.3.2 Identification of Uncertainties and Conservatisms in the Analysis

iodine and noble gas

a. Reactor coolant activities are based on an initial radio-iodine spectrum that would conservatively bound those found in either open or tight type fuel defects. Tight fuel defects tend to produce limiting results for thyroid does, while open fuel defects tend to produce limiting results for whole body dose. The assumed concentrations of longer-lived isotopes represent the values that would be reached in the presence of tight fuel defects. The assumed concentractions of shorter-lived isotopes represent the values that would be reached in the presence of open fuel defects. Since the assumed iodine spectrum represents bounding values for different types of fuel defects, the initial radio-iodine inventory would exceed the Technical Specification limit of 1.0 μCi/gm. Additionally, large spiking factors are assumed in the analysis.

b. A 1-gpm steam generator primary-to-secondary leakage is assumed, which is significantly greater than that anticipated during normal operation. Furthermore, it was conservatively assumed that all 1 gpm leakage is to the faulted ~~affected~~ steam generator only.

c. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that the assumed meteorological conditions would be present during the course of the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meteorological conditions assumed, are conservative.

Insert 15.1.5.3.2

15.1.5.3.3 Conclusions

15.1.5.3.3.1 Filter Loadings

The only ESF filtration system considered in the analysis which limits the consequences of the MSLB is the control room filtration system. Activity loadings on the control room charcoal filter are based on flow rate through the filter, the concentration of activity at the filter inlet, and the filter efficiency.

Activity in the control room filter as a function of time has been evaluated for the more limiting LOCA analysis, as discussed in Section 15.6.5.4.3.1. Since the control room filters are capable of accommodating the potential design basis LOCA fission product iodine loadings, more than adequate design margin is available with respect to postulated MSLB releases.

15.1.5.3.3.2 Dose to Receptor at the Exclusion Area Boundary, ~~and~~ Low-Population Zone Outer Boundary and in the Control Room

The potential radiological consequences resulting from the occurrence of a postulated MSLB have been conservatively analyzed, using ~~asusmptions~~ assumptions and models described. The TEDE doses ~~total-body gamma doses due to immersion from direct radiation and the thyroid dose due to inhalation~~ have been analyzed for the ~~0-2 hour dose~~ worst 2-hour time period at the exclusion area boundary, ~~and~~ for the duration of the accident (0 to ~~8~~ 22 hrs) at the low-population zone outer boundary, and in the Control Room. The results are listed in Table 15.1-4. The resultant doses are well within the guideline values of 10 CFR ~~100~~ 50.67 and Regulatory Guide 1.183 for offsite and the GDC 19 limit for the Control Room.

15.1.5.4 Conclusions

The analysis has shown that the criteria stated earlier in Section 15.1.5.1 are satisfied.

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable and not precluded by the criteria, the above analysis shows that the DNB design basis is met for any rupture, assuming the most reactive RCCA stuck in its fully withdrawn position.

A safety evaluation was performed to determine the impact of a potential increase in the stroke time of the feedwater isolation valves beyond the value assumed in the analyses (15 seconds) due to the installation of new valve actuators. It was concluded that the results presented in this section for the zero power steamline break event are not adversely affected by this plant modification. As such, the reported results and conclusions remain valid.

15.1.5.5 Steam Line Break with Coincident Control Rod Withdrawal

This accident is no longer applicable to Callaway since automatic rod withdrawal is no longer available.

15.1.5.6 Steam System Piping Failure at Full Power

15.1.5.6.1 Identification of Causes and Accident Description

A Steamline Rupture - Full Power Core Response transient is defined as a “break” that results in an increase in steam flow from one or more steam generators. A Steamline Rupture can result from:

- An inadvertent opening of a steam generator dump, safety or relief valve

TABLE 15.1-3 PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A MAIN STEAM LINE BREAK

I.	Source Data:	
a.	Core power level, Mwt	~~3636~~ 34
b.	Steam generator tube leakage, gpm	1
c.	Reactor coolant initial iodine activity:	
	1) ~~Case1~~	~~The MSLB Case 1 initial radio-iodine inventory is the same as the Case 1 SGTR initial radio-iodine inventory. Refer to Table 15.6-4~~.
	Insert T15.1-3A	
	2) ~~Case 2~~	~~The MSLB Case 2 initial radio-iodine inventory is the same as the Case 2 SGTR initial radio-iodine inventory. Refer to Table 15.6-4.~~
d.	Reactor coolant initial noble gas activity:	
	1) Case1	Based on ~~1-percent fuel defects as provided in Table 15A-5 (~~225 µCi/gm DOSE EQUIVALENT XE-133~~)~~
	2) Case 2	Based on ~~1-percent fuel defects as provided in Table 15A-5 (~~225 µCi/gm DOSE EQUIVALENT XE-133~~)~~
	Insert T15.1-3B	
f.	Secondary system initial iodine activity	10% of Case 1 primary side activity
g.	Iodine partition factors	
	1) ~~Affected~~ Faulted steam generator	1.0
	2) ~~Unaffected~~ Intact steam generator	0.01
i.	Reactor coolant mass, lbs	~~5.50E+5~~ 5.51E+5
	Insert T15.1-3C	
j.	Steam generator mass	
	1) ~~Affected~~ Faulted steam generator, lbs	1.555E+5
	2) Each ~~unaffected~~ Intact steam generator, lbs	1.555E+5
II.	Atmospheric Dispersion Factors	See Table 15A-2

TABLE 15.1-4 RADIOLOGICAL CONSEQUENCES OF A MAIN STEAM LINE BREAK

	Doses (rem) (TEDE)
CASE 1, accident initiated iodine spike	
Control Room (30 days)	2.81
Exclusion area boundary (~~0-2 hr~~ 6-8 hours)	5.54E-1
~~Thyroid~~	~~4.9E00~~
~~Whole body~~	~~1.3E-01~~
Low-population zone outer boundary (duration)	4.97E-1
~~Thyroid~~	~~6.1E00~~
~~Whole body~~	~~8.8E-02~~
CASE 2, pre-accident iodine spike	
Exclusion area boundary (0-2 hr)	1.11E-1
~~Thyroid~~	~~4.0E00~~
~~Whole body~~	~~2.5E-02~~
Low population zone outer boundary (duration)	1.03E-1
~~Thyroid~~	~~1.3E00~~
~~Whole body~~	~~1.3E-02~~
Control Room (30 days)	2.16

As such, the assumptions used in this analysis are designed to minimize the energy removal capability of the system and to maximize the possibility of water relief from the coolant system by maximizing the coolant system expansion, as noted in the assumptions listed above.

The assumptions used in the analysis are essentially identical to the loss of normal feedwater flow incident (Section 15.2.7), except that power is assumed to be lost to the reactor coolant pumps at the time of reactor trip.

Results

The transient response of the RCS following a loss of ac power with pressurizer PORVs unavailable is shown in Figures 15.2-9 through 15.2-11. The calculated sequence of events for this transient is listed in Table 15.2-1. The first few seconds after the loss of power to the reactor coolant pumps will closely resemble a simulation of the complete loss of flow incident (see Section 15.3.2); i.e., core damage due to rapidly increasing core temperatures is prevented by promptly tripping the reactor. After the reactor trip, stored and residual decay heat must be removed to prevent damage to either the RCS or the core.

The RETRAN code results show that the natural circulation flow available is sufficient to provide adequate core decay heat removal following reactor trip and RCP coast-down. A separate case was run with high head ECCS charging pumps initiated on a loss of offsite power signal (see assumption (l) above). This case did result in the filling of the pressurizer. However, this occurred sufficiently late in the transient such that operator action to unblock both pressurizer power-operated relief valves could be credited to preclude water relief through the pressurizer safety valves. This action was assumed to occur 9 minutes following the loss of offsite power while pressurizer filling occurred well after this time. This case is analyzed similar to the Inadvertent ECCS at Power event, discussed in Section 15.5.1, where operator action is required to unblock the pressurizer power-operated relief valves thereby precluding water relief through the pressurizer safety valves.

15.2.6.3 Radiological Consequences

15.2.6.3.1 Method of Analysis

15.2.6.3.1.1 Physical Model

The dose calculation for loss of ac power ~~is based on the sequence of events described in Table 15.2-1. It is~~ assumed that heat removal from the nuclear steam supply system is achieved by venting the steam for ~~8~~ 7.3 hours.

The reactor coolant is assumed to be contaminated by radioactive fission products introduced through fuel cladding defects. The secondary system is contaminated by the inleakage of reactor coolant through postulated steam generator tube leaks.

The radioactivity in the vented steam is dispersed in the atmosphere without any reduction due to plateout, fallout, filtering, etc.

15.2.6.3.1.2 Assumptions and Conditions

The major assumptions and parameters assumed in the analysis are found in Tables 15.2-2 and 15.A-1. The assumptions used to determine the activity released are as follows: Insert 15.2.6.3.1.2A

~~a. The reactor coolant initial iodine activity assumed is the Technical Specification limit of 1.0 μCi/gm I-131 dose equivalent (adjusted consistent with Table 15.6-4 item I.c.1).~~

Insert 15.2.6.3.1.2B

d. The initial secondary system iodine activity assumed is ~~1/10~~ 10% of the initial reactor coolant iodine activity.

f. A 1-gpm steam generator primary-to-secondary leakage is assumed for the duration of steam venting.

g For noble gases, the activity released is taken to be the activity introduced by reactor coola Insert 15.2.6.3.1.2C in the steam system.

h. ~~The partition factor for iodine in the steam generators is taken as 0.01 for secondary side releases and 1.0 for iodine in primary-to-secondary leakage. This assumption is conservatively based on a leak in the upper tubes which are assumed to be uncovered for the accident duration.~~

j. The atmospheric dispersion factors are given in Table 15A-2.

15.2.6.3.1.3 Mathematical Models Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

i. The alkali metal activity present in the primary-to-secondary leakage is assumed to be homogeneously mixed in the Steam Generator inventory.

g the

~~b.~~ The atmospheric dispersion factors used in the analysis were calculated using the onsite meteorological measurement programs described in Section 2.3 of the Site Addendum. and in the control room

~~c.~~ ~~The thyroid inhalation and total-body immersion~~ TEDE doses to a receptor at the exclusion area boundary ~~or~~ , outer boundary of the low population zone and in the control room were analyzed using the models described in Appendix 15A.

15.2.6.3.1.4 Identification of Leakage Pathways and Resultant Leakage Activities

Normal activity paths from the secondary system, such as the condenser air removal system and steam generator blowdown, cease during loss of ac power. The steam is released to the atmosphere through the:

a. Power-operated atmospheric relief valves

b. Reactor coolant activities based on extreme iodine spiking effects are conservatively high.

Since all these paths are taken as direct to the atmosphere without any form of decontamination, they are all radiologically equivalent and need not be distinguished.

15.2.6.3.2 Identification of Uncertainties in, and Conservative Aspects of, the Analysis

iodine and noble gas

The principal uncertainties in the ~~dose calculation arise~~ from the uncertainties in the accident circumstances, particularly the extent of steam contamination, the weather at the time, and delay before preferred ac power is restored. Each of these uncertainties ~~is~~ s handled by making ~~very~~ conservative or worst-case assumptions.

are

a. Reactor coolant activities are based on the Technical Specification limit, which ~~is~~ significantly higher than the activities associated with normal operating conditions~~, based on 0.12-percent failed fuel~~.

c. A 1-gpm steam generator primary-to-secondary leakage is assumed, which is significantly greater than that anticipated during normal operation.

d. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that the assumed meteorological conditions would be present during the course of the accident for any extended period of time. Therefore, the evaluated radiological consequences, based on the meteorological conditions assumed, will be conservative.

e. Steam dump to the condenser is not available

15.2.6.3.3 Conclusions

15.2.6.3.3.1 Filter Loadings

No filter serves to limit the release of radioactivity in this accident. There is no significant activity buildup in any ESF filter as a consequence of loss of ac power.

15.2.6.3.3.2 Doses to Receptor at Exclusion Area Boundary, ~~and~~ Low Population Zone Outer Boundary and in the Control Room

The maximum doses to an individual who spends the ~~first~~ worst 2 hours after loss of ac power at the exclusion area boundary, and the maximum doses for a long-term exposure (8 and in the control room for 30 days

hours or longer) at the outer boundary of the low-population zone, are given in Table 15.2-3. These doses are very small compared with the guideline values of 10 CFR ~~100~~.

15.2.6.4 Conclusions

Results of the analysis show that, for the loss of non-emergency ac power to plant auxiliaries event, all safety criteria are met. Auxiliary feedwater capacity is sufficient to prevent water relief through the pressurizer relief and safety valves; this assures that the RCS is not overpressurized.

50.67 for offsite dose and the GDC 19 limit for the Control Room

Analysis of the natural circulation capability g
term heat removal capability exists followin nt
fuel or clad damage.

15.2.7 LOSS OF NORMAL FEEDWATER FLOW

15.2.7.1 Identification of Causes and Accident Description

A loss of normal feedwater, from pump failures, valve malfunctions, or loss of offsite ac power, or feedwater control system failure, results in a reduction in the capability of the secondary system to remove the heat generated in the reactor core. If an alternative supply of feedwater were not supplied to the plant, core residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer would occur, resulting in a substantial loss of water from the RCS. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables never approach a DNB condition.

The reactor trip on low-low water level in one or more steam generators provides the necessary protection against a loss of normal feedwater.

The following occur upon loss of normal feedwater (assuming main feedwater pump failures or valve malfunctions):

a. As the steam system pressure rises following the trip, the steam generator power-operated relief valves are automatically opened to the atmosphere. Steam dump to the condenser is assumed not to be available. If the steam flow through the power-operated relief valves is not available, the steam generator safety valves may lift to dissipate the stored thermal energy of the reactor coolant system and fuel plus the residual decay heat produced in the reactor.

b. As the no-load temperature is approached, the steam generator power-operated relief valves (or safety valves, if the power-operated relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot shutdown condition.

a. Core Power level, MWt 3636 (includes 2% uncertainty)

TABLE 15.2-2 PARAMETERS USED IN EVALUATING RADIOLOGICAL CONSEQUENCES OF LOSS OF NONEMERGENCY AC POWER

I.	Source Data		
	b.	Steam generator tube leakage, gpm	1
	~~b.~~	~~Reactor coolant initial iodine activity dose equivalent~~ Insert T15.2-2A	~~1.0 µCi/gm of I-131 (adjusted consistent with Table 15.6-4 item I.c.1)~~
	d.	Secondary system initial iodine activity dose equivalent	Equivalent to ~~1/10~~ 10% of the initial RCS iodine activity
	e.	Reactor coolant initial noble gas activity	~~Based on 1 percent fuel defects, as provided in Table 11.1-5~~ 225 µCi/gm of Xe-133 dose equivalent
	f.	~~Iodine partition factor in the steam generators for secondary side releases~~ Portion of primary to secondary leakage flashed directly to the environment	~~0.01~~ 5% for the first 2.667 hours
	g.	Iodine partition factor ~~in the steam generators for primary-to-secondary leakage~~ for liquid primary-to-secondary leakage in the Steam Generators	1.0
	h.	Iodine and alkali metal partition factor for steam release from Steam Generator	0.01
	i.	Each steam generator water mass, lb	~~1.032E+5~~ 92.5E+3
II.	Atmospheric Dispersion Factors		See Table 15A-2

CASE 1, Accident initiated iodine spike

P

TABLE 15.2-3 RADIOLOGICAL CONSEQUENCES OF LOSS OF NONEMERGENCY AC POWER

	TEDE Doses (rem)
Exclusion area boundary (~~0-2 hr~~) 0.7 - 2.7 hrs	1.10E-02
~~Thyroid~~	~~6.2E-02~~
~~Whole body~~	~~4.9E-04~~
Low-population zone, outer boundary (duration) 30 day	5.56E-03
~~Thyroid~~	~~2.9E-02~~
~~Whole body~~	~~1.4E-04~~
Control Room (30 days)	1.50

CASE 2, Pre-accident iodine spike

Exclusion area boundary (0.6 - 2.6 hrs)	5.31E-03
Low-population zone, outer boundary (duration)	2.95E-03
Control Room (30 days)	0.82

The transient results with and without offsite power available are shown in Figures 15.3-9 through 15.3-12. The results of these calculations are also summarized in Table 15.3-2. The peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits. Also, the peak clad temperature is considerably less than 2,700°F. The clad temperature was conservatively calculated, assuming that DNB occurs at the initiation of the transient.

The calculated sequence of events is shown on Table 15.3-1. Figure 15.3-9 shows that the core flow rapidly reaches a new equilibrium value. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

15.3.3.3 Radiological Consequences

15.3.3.3.1 Method of Analysis

The release of radioactivity is assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated

15.3.3.3.1.1 Physical Model

The instantaneous seizure of a reactor coolant pump rotor results in a reactor trip on a low coolant flow signal. With the coincident loss of offsite power, the condensers are not available, so the excess heat is removed from the secondary system by steam relief through the steam generator safety and relief valves. ~~Steam generator tube leakage is assumed to continue until the pressures in the reactor coolant and secondary systems are equalized~~. The reactor coolant will contain the gap activities of the fraction of the fuel which undergoes DNB ~~in addition to its assumed equilibrium activity~~.

Tables 15B-1 and 15B-6 provide a comparison of the analysis to the guidelines in Regulatory Guide 1.183.

The major assumptions and parameters used in the analysis are itemized in Tables 15.3-3 and 15A-1 and summarized below.

Insert 15.3.3.3.1.1

~~The assumption used to determine the initial concentrations of isotopes in the reactor coolant and secondary coolant prior to the accident are as follows:~~

~~a. The reactor coolant iodine activity is based on the dose equivalent of 1.0 μCi/gm of I-131 (adjusted consistent with Table 15.6-4 item I.c.1).~~

~~b. The noble gas activity in the reactor coolant is based on 1-percent fuel defects.~~

~~c. The initial secondary system iodine activity assumed is 1/10 of the initial reactor coolant iodine activity.~~

is assumed to experience DNB such that its

sed to calculate the activity released.

a. 5 percent of the core gap activity is released to the reactor coolant at the beginning of the accident.

, with adjustment for high burnup fuel and radial power peaking as presented in Table 15.3-3. None of the fuel is predicted to experience Fuel Centerline Melt shutdown cooling is in operation and releases from the steam generators have been terminated. The

b. Offsite power is lost.

c. Following the incident, steam is released to the environment for heat removal.

7.26

d. Primary-to-secondary leakage continues after the accident for a period of ~~8~~ hours. At that time, ~~reactor coolant and secondary system pressures are equalized. Until the pressure equalizes, the~~ leakage rate is assumed to be constant and equal to ~~the rate existing prior to the incident of~~ 1 gpm (500 lbs/hr).

e. Fission products released from the fuel-cladding gap of the damaged fuel rods are assumed to be instantaneously and homogeneously mixed with the reactor coolant.

Insert 15.3.3.3.1.2

f. The noble gas activity released is equal to the amount present in the reactor coolant which leaks into the secondary system after the accident.

~~g.~~ ~~The partition factor for iodine in the steam generators is taken as 0.01 for secondary side releases and 0.161 for iodine in primary-to-secondary leakage. This assumption is conservatively based on a leak in the upper tubes which are assumed to be uncovered for the accident duration.~~

i. The activity released from the steam generators is immediately vented to the environment.

or the control room

k. No credit is taken for radioactive decay or ground deposition during radioactivity transport to offsite locations.

l. Breathing rates, short-term accident atmospheric dispersion factors corresponding to ground level releases, and dose conversion factors are given in Tables 15A-1, 15A-2, and 15A-4.

j. The alkali particulates are conservatively combined with, and treated as, halogens for transport trough the steam generators.

Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A.

b. The atmospheric dispersion factors used in the analysis were calculated based on the onsite meteorological measurement programs described in Section 2.3 of the Site Addendum, and are provided in Table 15A-2.

TEDE

, and in the control room

c. The ~~thyroid inhalation and total-body immersion~~ doses to a receptor at the exclusion area boundary ~~or~~ outer boundary of the low-population zone were analyzed using the models described in Appendix 15A.

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15.3.3.3.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

The leakage pathways are:

. and Main Steam Safety Valves (MSSVs)

a. Direct steam relief to the atmosphere through the S/G PORVs.

b. Primary-to-secondary steam generator tube leakage and subsequent steam relief to the atmosphere through the S/G PORVs.

and MSSVs

15.3.3.3.2 Identification of Uncertainties and Conservatisms in the Analysis

a. The ~~initial~~ reactor coolant and secondary coolant iodine activities are based on the assumptions stated in Section 15.3.3.3.1.2.

1 gpm

b. A ~~1 gpm~~ steam generator primary-to-secondary leakage, which is significantly greater than that anticipated during normal operation, is assumed.

c. The coincident loss of offsite power with the occurrence of a reactor coolant pump locked rotor is a highly conservative assumption. In the event of the availability of offsite power, the condenser steam dump valves will open, permitting steam dump to the condenser. Thus there is no direct release to the environment.

d. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that the meteorological conditions assumed will be present during the course of the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meteorological conditions assumed, are conservative.

15.3.3.3.3 Conclusions

15.3.3.3.3.1 Filter Loadings

The only ESF filtration system considered in the analysis which limits the consequences of the reactor coolant pump locked rotor accident is the control room filtration system. Activity loadings on the control room charcoal filter are based on the flow rate through the filter, the concentration of activity at the filter inlet, and the filter efficiency.

The activity in the control room filter as a function of time has been evaluated for the loss-of-coolant accident, Section 15.6.5. Since the control room filters are capable of

, and in the Control Room

accommodating the potential design-basis loss-of-coolant accident fission product iodine loadings, more than adequate design margin is available with respect to postulated reactor coolant pump locked rotor accident releases.

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15.3.3.3.3.2 Doses to Receptor at the Exclusion Area Boundary ~~and~~ Low-Population Zone Outer Boundary

TEDE doses

The potential radiological consequences resulting from the occurrence of a postulated reactor coolant pump locked rotor have been conservatively analyzed, using assumptions and models described in previous sections.

worst 2

,

The ~~total-body doses due to immersion from direct radiation and the thyroid dose due to inhalation~~ have been analyzed for the ~~0-2~~ hour dose at the exclusion area boundary ~~and~~ for the duration of the accident ~~(0 to 8 hours)~~ at the low-population zone outer boundary. The results are listed in Table 15.3-4. The resultant doses are ~~well~~ within the guideline values of 10 CFR ~~100~~.

Regulatory Guide 1.183 for offsite locations and the full GDC 19 limit in the control room

15.3.3.4 Conclusions

a. Since the peak RCS pressure reached during this transient is less than that which would cause stresses to exceed the faulted condition stress limits, the integrity of the primary coolant system is not endangered.

, and in the control room for 30 days

b. Since the peak clad surface temperature calculated for the hot spot during the transient remains considerably less than 2,700°F, the core will remain in place and intact with no loss of core cooling capability.

15.3.4 REACTOR COOLANT PUMP SHAFT BREAK

15.3.4.1 Identification of Causes and Accident Description

The accident is postulated as an instantaneous failure of a reactor coolant pump shaft, as discussed in Section 5.4. Flow through the affected reactor coolant loop is rapidly reduced, though the initial rate of reduction of coolant flow is greater for the reactor coolant pump rotor seizure event. Reactor trip is initiated on a low flow signal in the affected loop.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced - first, because the reduced flow results in a decreased tube side film coefficient and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray

Insert T15.3-3

TABLE 15.3-3 PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LOCKED ROTOR ACCIDENT

I.	Source Data		
	a.	Power level, MWt	3,636 (includes 2% uncertainty)
	b.	Steam generator tube leakage, gpm	1
	~~c.~~	~~Reactor coolant initial iodine activity~~	~~Dose equivalent of 1.0 μCi/gm of I-131 (adjusted consistent with Table 15.6-4 item I.c.1)~~
	~~d.~~	~~Reactor coolant initial noble gas activity~~	~~Based on 1-percent fuel defects, as provided in Table 15A-5~~
	~~e.~~	~~Secondary system initial iodine activity~~	~~Equivalent to 1/10 of the initial RCS activity~~
	~~f.~~	~~Activity released to reactor coolant from failed fuel~~	
		~~1. Noble gas, percent of gap inventory~~	~~5~~
		~~2. Iodine, percent of gap inventory~~	~~5~~
		~~3. Gap inventory~~	~~Table 15A-3~~
	~~g.~~	~~Iodine partition factor in the steam generators for secondary side releases~~	~~0.01~~
	~~h.~~	~~Iodine partition factor in the steam generators for primary-to-secondary leakage~~	~~0.161~~
	j.	Reactor coolant mass, lbs	~~5.5E+5~~ 5.51E+5
	k.	Steam generator mass, per generator	~~9.25E+5~~ 9.25E+4
II.	Atmospheric Dispersion Factors		See Table 15A-2

TABLE 15.3-4 RADIOLOGICAL CONSEQUENCES OF A LOCKED ROTOR ACCIDENT

	Doses (rem)
Exclusion Area Boundary ~~(0-2 hr)~~ 0.48 - 2.48 hrs	4.0E-01
~~Thyroid~~	~~2.0E01~~
~~Whole body~~	~~3.8E-01~~
Low Population Zone Outer Boundary (duration)	2.1E-01
~~Thyroid~~	~~8.1E00~~
~~Whole body~~	~~8.8E-02~~
Control Room (0-30 days)	1.34

hotter side of the rod. Calculations have indicated that this bowing would result in a negative reactivity effect at the hot spot since Westinghouse cores are under-moderated, and bowing will tend to increase the under-moderation at the hot spot. Since the 17 x 17 fuel design is also under-moderated, the same effect would be observed. In practice, no significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced. Boiling in the hot spot region would produce a net flow of coolant away from that region. However, the heat from the fuel is released to the water relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator to fuel ratio, and a large reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

15.4.8.3 Radiological Consequences

15.4.8.3.1 Method of Analysis

15.4.8.3.1.1 Physical Model

Prior to the accident, it is assumed that the plant has been operating with simultaneous fuel defects and steam generator tube leakage for a time sufficient to establish equilibrium levels of activity in the reactor coolant and secondary systems.

The RCCA ejection results in reactivity being inserted to the core which causes the local power to rise. In a conservative analysis, it is assumed that partial cladding failure and fuel melting occurs. The fuel pellet and gap activities are assumed to be immediately and uniformly released within the reactor coolant. Two release paths to the environment exist which are analyzed separately and conservatively, as if all the activity is available for release from each path.

The activity released to the containment from the reactor coolant through the ruptured control rod mechanism pressure housing is assumed to be mixed instantaneously throughout the containment and is available for leakage to the atmosphere. The only removal processes considered in the containment are ~~iodine plateout~~, radioactive decay, and leakage from the containment.

The model for the activity available for release to the atmosphere from the S/G relief valves assumes that the release consists of the activity in the secondary system plus that fraction of the activity leaking from the reactor coolant through the steam generator tubes. ~~The leakage of reactor coolant to the secondary side of the steam generator continues until the pressures in the reactor coolant and secondary systems equalize.~~

The release of radioactivity is assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

and MSSVs

~~Thereafter, no mass transfer from the reactor coolant system to the secondary system due to the steam generator tube leakage is assumed.~~ Thus, in the case of coincident loss of offsite power, activity is released to the atmosphere from steam relief through the S/G PORVs.

1.183. Tables 15B-1 and 15B-7 provide a comparison of the analysis to the guidelines of Regulatory Guide 1.183.

15.4.8.3.1.2 Assumptions and C

The major assumptions and parameters used in the analysis are itemized in Tables 15.4-3 and 15A-1 and summarized below. The assumptions are consistent with Regulatory Guide ~~1.77~~.

The assumption used to determine the initial concentrations of isotopes in the reactor coolant and secondary coolant prior to the accident are as follows:

a. The initial reactor coolant iodine activity corresponds to an isotope mixture that bounds Technical Specification allowable conditions for both tight and open fuel defects. The initial isotopic mix is based on the relative concentrations from Table 11.1-5. The concentrations are then changed to achieve a Dose Equivalent I-131 (DEI) of 1.0 μCi/gm, while maintaining the isotopic ratios from Table 11.1-5. ~~This provides conservative values for the longer lived iodines which contribute the majority of the calculated thyroid dose.~~ The initial concentration of the shorter lived iodines are then increased to bound the concentrations which would be observed in the presence of open fuel defects. ~~The shorter lived iodine isotopes are not major contributors of thyroid dose, but may provide a noticeable contribution to calculated whole body dose.~~ The initial reactor coolant iodine activity assumed for this sequence, as provided in Table 15A-5, bounds allowable plant conditions for open or tight fuel defects, and the contributions ~~of the longer and shorter lived isotopes to whole body and thyroid consequences~~.

to the TEDE.

b. The noble gas activity in the reactor coolant and secondary system is based on 1-percent fuel defects.

The noble gas concentrations for both Kr and Xe are then scaled to achieve a Dose Equivalent Xe-133 equal to the Technical Specification limit of 225 μCi/gm.

c. The initial secondary side iodine activity to ~~1/10th~~ of the initial assumed primary side iodine activity.

10%

The [...] following a RCCA ejection accident.

a. ~~10 percent of the fuel rod gap activity, except for Kr-85 which is 30 percent, is additionally released to the reactor coolant.~~

Insert 15.4.8.3.1.2A

b. 0.25 percent of the fuel is assumed to melt.

Insert 15.4.8.3.1.2B

residual heat removal operation to take over decay heat removal at 7.29 hours

c. Following the incident until ~~primary and secondary side pressures equalize~~, steam is released to the environment.

leakage

d. The 1-gpm primary-to-secondary ~~leak~~ to the steam generators is assumed.

e. All noble gas activity in the reactor coolant which is transported to the secondar to be immediat

for the primary to secondary release pathway case and instantaneously and homogeneously mixed within containment for the containment leakage pathway case

Insert 15.4.8.3.1.2C

f. Fission p fuel rods are assumed to be instantaneously and homogeneously mixed with the reactor coolant.

~~g. A partition factor of 0.1 between the vapor and liquid phases for radioiodine in the steam generators is used for secondary side releases and 0.1 for iodine in primary-to-secondary leakage.~~

i. The activity released from the steam generators is immediately relieved to the environment.

j. The containment is assumed to leak at 0.2 volume percent/day during the first 24 hours immediately following the accident and 0.1 volume percent/ day thereafter.

or control room

k. No credit is taken for radioactive decay or ground deposition during radioactivity transport to offsite location. s

l. Short-term accident atmospheric dispersion factors corresponding to ground level releases, breathing rates, and dose conversion factors are given in Tables 15A-2, 15A-1, and 15A-4, respectively.

m. Offsite power is assumed lost.

15.4.8.3.1.3 Mathematical Models Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the ac

in that the condenser is not available to receive the steam released for decay heat removal in the fuel. However, as bias applied to increase the heat load (and so steam release) all four reactor coolant pumps continue to operate.

b. The atmospheri n based on the o Section 2.3 of th

TEDE   and in the control room

c. The ~~thyroid inhalation and total-body immersion~~ doses to a receptor at the exclusion area boundary ~~of~~ outer boundary of the low-population zone were analyzed, using the models described in Appendix 15A.

,

15.4.8.3.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

The leakage pathways are:

and MSSVs

a. Direct steam relief to the atmosphere through the S/G PORVs

b. Primary-to-secondary steam generator tube leakage and subsequent steam relief to the atmosphere through the S/G PORVs. and MSSVs

c. The resultant activity released to the containment is assumed available for leakage directly to the environment.

15.4.8.3.2 Identification of Uncertainties and Conservative Elements in the Analysis

~~a.~~ ~~Reactor coolant and secondary coolant activities are many times greater than assumed for normal operation conditions.~~

a. A 1-gpm steam generator primary-to-secondary leakage, which is significantly greater than that anticipated during normal operation and

d. The alkali particulates are conservatively combined with, and treated as, halogens for transport trough the steam generators.

b. The coincident loss of offsite power with the occurrence of a RCCA ejection accident is a highly conservative assumption. In the event of the availability of offsite power, the condenser steam dump valves will open, permitting steam dump to the condenser. Thus there is no direct release via that path to the environment.

none   recognized

c. It is assumed that ~~50 percent~~ of the iodines released to the containment atmosphere is adsorbed (i.e. plate out) onto the internal surfaces of the containment or adheres to internal components. However, it is ~~estimated~~ that the removal of airborne iodines by various physical phenomena such as adsorption, adherence, and settling could reduce the resultant doses ~~by a factor of 3 to 10~~.

d. The activity released to the containment atmosphere is assumed to leak to the environment at the containment leakage rate of 0.2-volume percent/day for the first 24 hours and 0.1-volume percent/day thereafter. The initial containment leakage rate is based on the peak calculated internal containment pressure anticipated after a LOCA. The pressures associated with a RCCA ejection accident are considerably lower than that calculated for a LOCA. The pressure inside the containment also decreases considerably with time, with an expected decrease in leakage rates. Taking into account that the containment leak rate is a function of pressure, the resultant doses could be reduced by a a significant factor ~~factor of 5 to 10 (Ref. 12)~~.

e. . The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that the meteorological conditions assumed will be present during the course of the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meteorological conditions assumed, are conservative.

15.4.8.3.3 Conclusions

15.4.8.3.3.1 Filter Loadings

The only ESF filtration system considered in the analysis which limits the consequences of the RCCA ejection accident is the control room filtration system. Activity loadings on the control room charcoal filter are based on the flow rate through the filter, the concentration of activity at the filter inlet, and the filter efficiency.

The activity in the control room filter as a function of time has been evaluated for the loss-of-coolant accident, Section 15.6.5. Since the control room filters are capable of accommodating the potential design-basis loss-of-coolant accident fission product iodine loadings, more than adequate design margin is available with respect to postulated RCCA ejection accident releases.

15.4.8.3.3.2 Doses to Receptor at the Exclusion Area Boundary , ~~and~~ Low-Population Zone Outer Boundary and in the Control Room

The potential radiological consequences resulting from the occurrence of a postulated RCCA ejection accident have been conservatively analyzed, using assumptions and models described in previous sections.

The TEDE doses ~~total body doses due to immersion from direct radiation and the thyroid dose due to inhalation~~ have been analyzed for the worst 2 ~~0-2~~ hour dose at the exclusion area boundary , ~~and~~ for the duration of the accident at the low-population zone outer boundary and in the control room for 30 days. The results are listed in Table 15.4-4. The resultant doses are well within the guideline values of ~~10 CFR 100~~ Regulatory Guide 1.183 for offsite locations and the full GDC 19 limit in the control room.

15.4.8

Even on a conservative basis, the analyses indicate that the described fuel and clad limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the RCS. The analyses have demonstrated

TABLE 15.4-3 PARAMETERS USED IN EVALUATING THE RCCA EJECTION ACCIDENT

I.	Source Data			
	a.	Core power level, MWT		3636 (includes 2% uncertainty)
	b.	Core inventories		Table 15A-3
	c.	Steam generator tube leakage, gpm		1
	d.	Reactor coolant initial noble gas activity		225 µCi/gm of Xe-133 dose equivalent ~~Based on 1 percent fuel defects, as provided in Table 15A-5~~
	g.	Radial peaking factor		1.65
	e.	Reactor coolant initial iodine activity		See Section 15.4.8.3.1.2.a.
	f.	Secondary system initial iodine activity		See Section 15.4.8.3.1.2.c.
	h.	Extent of core damage		10 percent of fuel rods experience cladding failure; 0.25 percent of fuel experiences melting
	i.	Activity released ~~to reactor coolant~~, percent		
		1. Cladding failure		
			(a) Noble gas gap activity	100
			(b) Iodine gap activity	25 for Case 1 to containment 50 for Case 2 to reactor coolant
		2. Fuel melting		
		Insert T15.4-3	(a) Noble gas gap activity	100
			(b) Iodine fuel activity	~~50~~
	j.	Iodine partition factor in the steam generators for secondary side ~~releases~~ bulk boiling		~~0.1~~ 0.01
	~~j.~~	~~Iodine partition factor in the steam generators for primary-to-secondary leakage~~		~~0.1~~
	m.	Reactor coolant mass, lbs		~~5.50 + 5~~ 5.51E+5
	n.	Total secondary side ~~fluid~~ steam mass released to the environment, lbs		~~4.24E + 5~~ 1.66E6

TABLE 15.4-3 (Sheet 2)

II.	Atmospheric Dispersion Factors		Table 15A-2
III.	Activity Release Data		
	a.	Containment volume, ft^3	~~2.5 + 6~~ 2.7E+6
	b.	Containment leak rate, volume percent/day	
		1. 0-24 hours	0.20
		2. 1-30 days	0.10
	c.	Percent of containment leakage that is unfiltered	100
	d.	Plateout of iodine within containment, percent	~~50~~ 0
	e.	Offsite power	Lost
	f.	Mass of primary fluid leaked to the secondary lbs	~~167~~ 3648
	g.	Duration of primary-to-secondary leakage, sec	~~1200~~ 26,244 (7.29 hours)

TEDE

TABLE 15.4-4 RADIOLOGICAL CONSEQUENCES OF A ROD-EJECTION ACCIDENT

	Doses (rem)
Control Room (30 days)	4.05
CASE 1, Containment Leakage Release	
Exclusion Area Boundary (0-2 hr)	1.43
~~Thyroid~~	~~1.3E01~~
~~Whole body~~	~~6.6E-02~~
Low Population Zone Outer Boundary (duration)	3.24
~~Thyroid~~	~~1.3E01~~
~~Whole body~~	~~2.3E-02~~
CASE 2, Steam Generator Atmospheric Steam Dump Release	
Exclusion Area Boundary (~~0-2 hr~~ 0.6 - 2.6 hrs)	1.34
~~Thyroid~~	~~4.9E00~~
~~Whole body~~	~~1.6E-01~~
Low Population Zone Outer Boundary (duration)	7.2E-01
~~Thyroid~~	~~4.9E-01~~
~~Whole body~~	~~1.6E-02~~
Control Room (30 days)	4.13

of the reactor makeup system, it would not result in engineered safety features system actuation. Frequent operation of the automatic reactor makeup system will provide the operator some indication of the loss of reactor coolant.

15.6.2.1 Radiological Consequences

c. The alkali metal activity in the reactor coolant is based on 1% fuel defects, as provided in Table 11.1-5

15.6.2.1.1 Method of Analysis

15.6.2.1.1.1 Physical Model

The volatile fractions of the spilled reactor coolant are assumed to be available for immediate release to the environment.

225 μCi/gm of Xe-133 dose equivalent

15.6.2.1.1.2 Assumptions and Conditions

The major assumptions and parameters used in the analysis are provided in Table 15.6-2 and summarized below:

a. The reactor coolant initial iodine activity is based on the dose equivalent of 1.0 μCi/gm of I-131 ~~(adjusted consistent with Table 15.6-4 item I.c.1)~~. Although no reactor trip or primary side depressurization is expected, an accident-initiated iodine spiking factor of 500 is assumed in Table 15.6-2 to conservatively address scenarios including a reactor trip.

b. The initial noble gas activity in the reactor coolant is based on ~~1 percent fuel defects.~~

d. A total of 39,958 pounds of reactor coolant is spilled (based on a release for 30 minutes followed by a 10-second valve closure) onto the auxiliary building floor.

e. All of the noble gases in the spilled reactor coolant are released to the environment.

f. ~~Ten~~ percent of the spill is assumed to flash. All of the iodine activity in the flashed fraction of the spill is assumed to be released.

Twenty

g. No credit is taken for mixing and holdup of the releases within the auxiliary building, nor are the auxiliary building normal exhaust filters credited with reducing the release. That is, the release is modeled as being direct to the environment.

15.6.2.1.1.3 Mathematical Models Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A.

b. The atmospheric dispersion factors were calculated based on the onsite meteorological measurement programs described in Section 2.3 of the Site Addendum, and are provided in Table 15A-2.

c. The ~~thyroid inhalation and total body immersion~~ TEDE doses to a receptor at the exclusion area boundary, ~~or~~ outer boundary of the low-population zone, and in the control room were analyzed, using the models described in Appendix 15A.

15.6.2.1.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

The reactor coolant spilled in the auxiliary building will collect in the floor drain sumps. From there, it will be pumped to the radwaste treatment system. Therefore, the only release paths that present a radiological hazard involve the volatile fraction of spilled coolant.

Normally, gases released in the auxiliary building mix with the building atmosphere and are gradually exhausted through the filtered building ventilation system. The charcoal filters normally remove a very large fraction of the airborne iodine in the building atmosphere. However, the ventilation system is not designed to mitigate the consequences of an accident (e.g., it might not survive an earthquake more severe than the operating-basis earthquake), nor can the possibility of unplanned leakages from the auxiliary building be eliminated; hence, no credit is taken for these effects reducing the released activity.

The evaporated radionuclides are assumed to be available immediately to the outside atmosphere.

15.6.2.1.2 Identification of Uncertainties and Conservatisms in the Analysis

The principal uncertainties in the calculation of doses following a letdown line rupture arise from the unknown extent of reactor coolant contamination by radionuclides, the quantity of coolant spilled, the fraction of the spilled activity that escapes the auxiliary building, and the environmental conditions at the time. Each of these uncertainties is treated by taking worst-case or extremely conservative assumptions.

The extent of coolant contamination assumed greatly exceeds the levels expected in practice. The rupture is postulated in a seismic Category I, ASME Section III, Class 2 piping system. It is assumed that the leak goes undetected for 30 minutes. It is expected that considerable holdup and filtration occurs in the auxiliary building, but no credit is assumed.

The purpose of all these conservatisms is to place an upper bound on doses.

15.6.2.1.3 Conclusions

15.6.2.1.3.1 Filter Loadings

No filter is credited with the collection of radionuclides in this accident analysis. The buildup on these filters (auxiliary building and control building charcoal filters) that may be expected due to the adsorption of some of the iodine is very small compared with the design capacity of these filters.

,

15.6.2.1.3.2 Dose to Receptor at the Exclusion Area Boundary ~~and~~ Low-Population Zone Outer Boundary

and in the Control Room

TEDE

The radiological consequences resulting from the occurrence of a postulated letdown line rupture have been conservatively analyzed, using assumptions and models described in previous sections.

worst 2

The ~~thyroid inhalation total-body immersion~~ doses have been analyzed for the ~~0-2~~ hour dose at the exclusion area boundary ~~and~~ for the duration of the accident at the low-population zone outer boundary. The results are listed in Table 15.6-3. The resultant doses are well within the guideline values of 10 CFR ~~100~~.

,

15.6.3 STEAM GENERATOR TUBE FAILURE

The accident examined is the complete severance of a single steam generator tube. This event is considered an ANS Condition IV event, a limiting fault (see Section 15.0.1). The accident is assumed to take place at 50.67 and Regulatory Guide 1.183 aminated with corresponding to co unt of defective , and in the control room for 30 days accident leads to an increase in the contamination of the secondary system due to the leakage of radioactive coolant from the RCS. In the event of a coincident loss of offsite power or failure of the steam dump system, discharge of activity to the atmosphere takes place via the steam generator safety and/or power-operated atmospheric steam dump valves.

In view of the fact that the steam generator tube material is Inconel-600 and is a highly ductile material, it is considered that the assumption of a complete severance is somewhat conservative. The more probable mode of tube failure would be one or more minor leaks of undetermined origin. Activity in the steam and power conversion system is subject to continual surveillance, and an accumulation of minor leaks which exceed the limits established in the Technical Specifications is not permitted during plant operation.

In order to select the reference worst case, a spectrum of steam generator tube rupture (SGTR) events was anayzed. The letters of Reference 3 provide a detailed description of the selection process. Based on the selection process, two major SGTR accident scenarios are identified as the major concerns for radioactive releases to the environment.

15.6.3.1.2 Analysis of Effects and Consequences

<u>Method of Analysis</u>

-3D

Mass and energy balance calculations are performed using RETRAN (Section 15.0.11.8) to determine primary-to-secondary mass release and to determine the amount of steam vented from each of the steam generators from the occurrence of the tube rupture until after the second primary-secondary pressure equalization. RETRAN provides time-dependent values of RCS mass, break flow, flashed fraction, steam generator liquid mass, and steam generator atmospheric steam dump valve flow for the calculation of radiological consequences. Conservatively high values of break flow rate and flashed fraction are assumed for ~~the first hour of~~ the transient to maximize radiological consequences. ~~Supplementary mass and energy balance calculations, with conservative assumptions, are performed for the period from pressure equalization until 8 hours after the accident, beyond the time of RHR initiation.~~

In estimating the mass transfer from the RCS through the broken tube, the following assumptions are made:

a. Reactor trip and safety injection occur coincidentally as a result of low pressurizer pressure. Overtemperature ΔT trip is not considered. This allows more break flow. Loss of offsite power occurs at reactor trip.

b. The tube rupture is a double-ended guillotine break of a single hot leg tube at the tube sheet of the steam generator. This break location maximizes the flashed fraction of the RCS break flow.

c. As listed on Table 15.0-4, the low pressurizer pressure safety analysis limit (SAL) for reactor trip is 1845 psig. This reactor trip SAL is lower than the actual setpoint of 1885 psig, which thereby delays the trip and results in increased break flow. Safety injection is assumed concurrent with reactor trip which decreases the time for initiation of safety injection, again resulting in increased break flow. Safety injection occurs 15 seconds after the SI signal. The actual SI setpoint is 1849 psig with a lower SAL in Table 15.0-4. This minimum expected delay results in an early rise in RCS pressure due to SI and results in increased break flow.

d. Break flow is characterized by resistance-limited flow. An additional 5% uncertainty is added to the flow.

e. The assumption of a loss of offsite power at reactor trip prevents steam dump to the condenser and steam is discharged to the atmosphere via the ASDs. With the condenser unavailable for retention of any leaked radioactivity, offsite doses are maximized.

f. Pressurizer heaters and spray are not modelled.

g. MSIV isolation is modeled at reactor trip and the assumed loss of offsite power, although it could be sig ... ed on the expected operator response. Early isolat ... es the failed open ASD to have a greater impact on the ... ator pressure, which maximizes steam flow and bre ... mline isolation ... maximizes the mass transferred ... team generator to the atmosphere.

This encompasses a 2.3 second signal delay and a 2.0 second valve stroke time.

Feedwater

4.3

h. Prior to reactor trip, the normal feedwater matches the steam flow in the intact steam generators. For the ruptured steam generator, the total feed flow (including the break flow) matches the steam flow. ~~The feedwater~~ isolation ~~signal~~ occurs ~~2.3~~ seconds after reactor trip causes rod motion ~~and the feedwater isolation valves stroke closed within 2.0 seconds~~. These are the minimum expected delay and stroke time, respectively, which tend to decrease heat removal from the RCS resulting in higher RCS temperatures and pressures. This results in maximum flashed fraction and break flow.

62%

i. The initial steam generator liquid level is ~~48.4~~% of the narrow range span. ~~This is the minimum expected level, minimizing the amount of secondary inventory available for decay heat removal. This increases the flashed fraction (the amount of leaked reactor coolant that is vaporized on the secondary side).~~ Auxiliary feedwater (AFW) flow is maintained to achieve a narrow range level of at least ~~45~~% in all steam generators. AFW is initiated 60 seconds after reactor trip and attains a flow rate of 250 gpm to all steam generators. 50% maximum expected delay for AFW initiation maximizes break flow and maintains high RCS temperatures. This minimum expected AFW flow to the ruptured steam generator results in decreased RCS heat rem ... CS temperatures, and thereby maximizes the fla ... reactor coolant.

until the cooldown to RHR conditions is initiated when it is lowered to 1% to maximize the steam released during the long term cooldown.

j. The ruptured steam gener ... 184.7 psia. This is 4% higher than the nominal st ... release of pressure from the ruptured steam gener ... sed valve discharge flow and integrated break flow. ... red steam generator fails open for 20 minutes, begi ... , shortly after reactor trip.

k. The ... perature is 4.3°F above the nominal valu ... on of the RCS break flow.

applying a 1.2133 multiplier to

l. Core residual heat generation is based on the 1979 version of ANS 5.1. ANSI/ANS-5.1-1979 is a conservative representation of the decay energy release rates. Long-term operation at the initial power level preceding the trip is assumed.

m. The narrow range level in all steam generators must be greater than 4% and the ruptured steam generator pressure must be greater than 430 psig prior to initiating RCS cooldown. The cooldown is initiated 10 minutes after the failed ASD is isolated.

n. RCS depressurization is assumed to begin 3 minutes after completion of cooldown. When the ruptured steam generator pressure is higher than the RCS pressure, the pressurizer PORVs are closed.

o. Safety injection is terminated 5 minutes after completion of RCS depressurization.

Other initial conditions, given in Table 15.0-2, are chosen to maximize RCS temperatures, decay heat, flashed fraction of RCS leakage, and break flow, thereby maximizing radioactivity transfer to the secondary and, consequently, offsite doses.

The ASDs on the intact SGs are conservatively assumed to open only 90%.

...ve assumptions, suitably conservative for this case are made to maximize offsite

...reactor trip, steam is dumped to the condenser from both the ruptured and intact ...generators. After the condenser is lost, following assumed loss of offsite power at reactor trip, steam from all steam generators is released to the atmosphere.

that of

~~Following isolation of the ruptured steam generator, one of the ASDs on the three intact steam generators i~~ of the intact loops ~~umed to fail closed~~. ~~This additional failure is beyond single failure criteria.~~ The effect of this assumption is to conservatively increase the time it takes to reduce the RCS temperature to below the ruptured steam generator saturation temperature using atmospheric steam dump from the intact steam generators. ~~From 2 to 5 hours,~~ steam is assumed to be relieved from the intact steam generators to reduce the ~~RCS~~ temperature ~~and pressure~~ to RHRS conditions. ~~The ruptured steam generator is depressurized to the RHRS cut-in pressure using the emergency recovery procedures.~~ After ~~5~~ hours, further plant cooldown is carried out with the RHRS. The 0 to 2 hour and 2 to ~~8~~ hour steam releases from the intact steam generators required to remove decay heat, metal heat, reactor coolant pump heat, and stored fluid energy in the RCS and steam generators are determined based on these assumptions.

5.86

5.86

Key Recovery Sequence

The ruptured loop has no cooling to drive circulation in the loop and therefore remains at a higher temperature than the intact loops. After the intact loops have reached RHR cut-in temperature the ruptured loop ASD block valve is reopened to depressurize the ruptured SG to RHR cut-in pressures and to induce circulation and cool the ruptured loop to RHR-cut in conditions. This method was chosen for cooling the ruptured SG as it is the most conservative with respect to steam release

The recovery sequence to be followed cons...

a. Identification of the ruptured s...

b. Isolation of the ruptured stea... ASD block valve;

c. After the operators have put an end to the rupture flow by lowering the primary pressure to below that of the ruptured SG,

d. Controlled depressurization of the RCS to a value equal to the ruptured steam generator pressure;

intact loops to RHR cut-in temperature

e. Subsequent termination of safety injection flow; and

f. Further cooldown ~~and depressurization of the RCS to conditions suitable for RHR initiation~~.

g. Depressurization of the ruptured SG and the RCS and cooldown of the ruptured loop until RHR cut-in conditions are reached.

<u>Results</u>

In Table 15.6-1, the sequence of events is presented. These events include postulated operator action times. Loss of offsite power is assumed to occur at reactor trip.

220,243

The previously discussed assumptions lead to an estimate of ~~486,000~~ pounds for the total amount of reactor coolant transferred to the secondary side of the ruptured steam generator as a result of a tube rupture accident. The steam releases to the ~~condenser and~~ atmosphere from both the ruptured and intact steam generators are given in Table 15.6-4.

The following is a list of figures of pert

This value is conservatively biased to 300,000 pounds for use as the basis for offsite and control room dose.

<u>Number</u>

15.6-3a Pressurizer and Steam Generator (Ruptured and Intact Generators) Pressure Transients for Steam Generator Tube Rupture Event

15.6-3b Reactor Coolant System Temperature (Ruptured Loop) Transient for Steam Generator Tube Rupture Event

15.6-3c Reactor Coolant System Temperature (Intact Loops) Transient for Steam Generator Tube Rupture Event

15.6-3d Steam Flow Rate (Intact Generators) Transient for Steam Generator Tube Rupture Event

15.6-3e Steam Flow Rate (Ruptured Generator) Transient for Steam Generator Tube Rupture Event

15.6-3f Steam Generator Temperature (Ruptured and Intact Generators) Transients for Steam Generator Tube Rupture Event

15.6-3g Steam Generator Atmospheric Relief Valve Flow Rate (Ruptured Generator) Transient for Steam Generator Tube Rupture Event

15.6-3h Steam Generator Atmospheric Relief Valve Flow Rate (Intact Generators) Transient for Steam Generator Tube Rupture Event

Tables 15B-1 and 15B-5 provide a comparison of the analysis to the guidelines in Regulatory Guide 1.183.

15.6.3.1.3 Radiological Consequences

Method of Analysis

The evaluation of the radiological consequences due to a postulated steam generator tube rupture (SGTR) with a stuck open atmospheric steam dump valve on the ruptured steam generator assumes a complete severance of a single steam generator tube while the reactor is operating at full rated power and a coincident loss of offsite power. Occurrence of the accident leads to an increase in contamination of the secondary system d[...]ough the tube break. A reactor trip occurs automatic[...]er pressure. The reactor trip will automatically trip the tu[...]

The intact loop ASDs are assumed to open only 90%.

Steam generator blowdown will automatically be terminated by the SGBSIS (AFAS) signal (refer to Section 10.4.8) which is initiated by the safety injection signal. The assumed coincident loss of offsite power will cause closure of the condenser steam dump valves to protect the condenser. The steam generator pressure will then increase rapidly, resulting in steam discharge as well as activity release through the steam generator atmospheric steam dump valves. ~~An atmospheric steam dump valve on one of the unaffected steam generators is conservatively assumed not to open and will therefore be unavailable to support the RCS cooldown.~~ This assumption has the effect of increasing the time it takes to reduce the RCS temperature to below the ruptured steam generator saturation temperature. ~~This additional failure is beyond the required single failure criteria.~~ Venting from the affected steam generator, i.e., the steam generator which experiences the tube rupture, will continue until the manual block valve is closed, isolating the stuck open atmospheric steam dump valve on the ruptured steam generator. At this time, the affected steam generator is effectively isolated. The remaining unaffected steam generators remove core decay heat by venting steam through the atmospheric steam dump valves until the ~~controlled cooldown is terminated~~.

alkali metals,

The analysis of the radiological consequences of an SGTR considers the most severe release of secondary activity, as well as reactor activity leaked from the tube break. The inventory of iodine and noble gas fission product activity available for release to the environment depends on the primary-to-secondary break flow and coolant leakage rates, ~~the percentage of defective fuel in the core~~, flashed fraction of reactor coolant, and the mass of steam discharged to the environment. Conservative assumptions were made for all these parameters.

activity initially in the primary and secondary systems

The major assumptions and parameters assumed in the analysis are itemized in Tables 15.6-4 and 15A-1 and are summarized below.

The assumpt

and seconda

RHR conditions are reached in the intact loops. At this point the ruptured loop ASD block valve is reopened to induce natural circulation and cooling in the ruptured loop. Steam release through the ASDs of all loops is terminated when RHR cut-in temperature is reached in the ruptured loop.

a.

two cases:

Case 1 - The initial reactor coolant iodine activity corresponds to an isotope mixture that bounds Technical Specification allowable conditions for both tight and open fuel defects. The isotopic mix is based on the initial RCS concentrations from Table 15A-5. This table provides conservative values for the iodine isotopic spectrum that bound the RCS concentrations which could be expected with either tight or open fuel defects. Case 1 then includes an accident initiated, spiked release rate that increases by a factor of 335 during the accident sequence.

Case 2 - The initial reator coolant iodine activity corresponds to an assumed pre-accident iodine spike which results in concentrations that are a factor of 60 higher than those used in Case 1.

b. The noble gas activity in the reactor coolant, as provided in Table 15A-5 (225 mCi/gm DOSE EQUIVALENT XE-133) .

c. The initial secondary side radio-iodine concentrations are assumed to be 10% of the initial Case 1 primary side concentrations.

The following assumptions and parameters are used to calculate the activity released and the offsite doses following an SGTR:

a. Break flow to the ruptured steam generator is conservatively assigned values that bound calculated break flow rate values. The assumed values bound the break flow rates calculated by the RETRAN code. Break flow rate values are discussed in Table 15.6-4 (225 μCi/gm DOSE EQUIVALENT XE-133).

b. The fraction of reactor coolant that flashes to steam after reaching the secondary side, as assumed in the accident analysis, varies over time. Key events which trigger changes in the assumed flashed fraction are reactor trip and closure of the manual block valve to isolate the failed open SG atmospheric steam dump valve. Flashed fraction values assumed in the radiological analysis are described in Table 15.6-4.

c. A 1-gpm primary-to-secondary leak is assumed to occur to the unaffected steam generat occurs for the first 1.39 hours. Primary

approximately

d. All noble gas activity in the reactor coolant which is transported to the secondary system via the tube rupture and the primary-to-secondary leakage is assumed to be immediately released to the environment.

e. At 80 minutes after the accident, it is assumed that the RCS and steam generator pressures are equalized and below the steam generator atmospheric relief valve set pressure. Break flow to the ruptured steam generator ~~and primary~~ to-secondary leakage to the intact steam generators ~~are~~ conservatively assumed to continue until ~~8~~ hours after the tube rupture.

f. The iodine partition faction between the liquid and steam in the steam generator is assumed to be 0.01.

is

5.86

g. The steam releases from the steam generators to the atmosphere are given in Table 15.6-4.

At 5.86

h. Offsite power is lost.

i. ~~Five~~ hours after the accident, the RHR system is assumed to be in operation to cool down the plant. Thus, no additional steam release is assumed.

j. Radioactive decay prior to the release of activity is considered. No decay during transit or ground deposition is considered.

k. Short-term accident atmospheric dispersion factor, breathing rates, and dose conversion factors are provided in Tables 15A-2, 15A-1, and 15A-4, respectively.

Mathematical Models Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are based on the assumptions listed above.

and in the control room

b. The atmospheric di[illegible] analysis were calculated based on the onsite meteorological measurements program, as described in Section 2.3 of the Site Addendum, and are provided in Table 15A-2.

TEDE

c. The ~~thyroid inhalation immersion~~ doses to a receptor at the exclusion area boundary ~~and~~ outer boundary of the low-population zone were analyzed, using the models described in Appendix 15A.

,

Identification of Leakage Pathways and Resultant Leakage Activity

The

For the purposes of evaluating the radiological consequences due to a postulated SGTR, the activity released from the affected steam generator, prior to isolation, is released directly to the environment by the atmospheric steam dump valve. ~~Two of the~~ unaffected steam generators are assumed to continually discharge steam and entrained activity via the atmospheric steam dump valves up to the time initiation of the RHR system can be accomplished. Since the activity is released directly to the environment with no credit for plateout or retention, the results of the analysis are based on the most direct leakage pathway available. Therefore, the resultant radiological consequences represent the most conservative estimate of the potential integrated dose due to the postulated SGTR.

Identification of Uncertainties and Conservatisms in the Analysis

a. Reactor coolant activities based on extreme iodine spiking effects are orders of magnitude greater than that assumed for normal operating conditions.

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b. A 1-gpm steam generator primary-to-secondary leakage, with a conservatively high density, is assumed which is significantly greater than that anticipated during normal operation. This leakage continues ~~for 8 hours, even though~~ RHR operation is assumed to begin at ~~5~~ hours.

until

c. Tube rupture of the steam generator is assumed to be a double-ended severance of a single steam generator tube. This is a conservative assumption, since the steam generator tubes are constructed of highly ductile materials. The more probable mode of tube failure is one or more the leaked primary fluids which immediately flashes to steam after arriving in the secondary side. Release via the steaming pathway is terminated by the SG atmospheric steam dump block valve closure at 30 minutes. Release via the flash pathway is conservatively continued following block valve closure. Release via this pathway is continued until the RETRAN results indicate that no further flashing will occur.

~~k.~~ ~~The steam release from the intact steam generator ASDs during the 5 hour cooldown to RHR cut-in conditions is conservatively assumed to occur in its entirety during the 0-2 hour period of the transient.~~

k. TEDE ~~Whole body~~ doses from the intact steam generator ASDs during the cooldown to RHR cut-in conditions are calculated using conservative primary side activities.

Table 15.6-5 lists the offsite doses for the SGTR with a stuck-open ASD.

15.6.3.2 STEAM GENERATOR TUBE RUPTURE WITH FAILURE OF FAULTED STEAM GENERATOR AFW CONTROL VALVE

15.6.3.2.1 Identification of Causes and Accident Description

As discussed in Reference 3, an SGTR case demonstrating the effects of steam generator overfill was performed. In this case the analysis assumes the failure of the AFW control valve on the discharge side of the motor-driven AFW pump feeding the ruptured steam generator. The ASD on the ruptured steam generator is not assumed to fail open. The ASD never opens and all liquid relief is considered through a main steam safety valve (MSSV). The AFW control valve is assumed failed in the wide-open position to maximize the flow to the ruptured steam generator. Failure of this valve coupled with the contribution from the turbine-driven AFW pump provides a greater potential for overfilling the ruptured steam generator. For this special overfill scenario, reactor trip and safety injection actuation were conservatively assumed at SGTR initiation (time zero) to maximize the AFW addition to the ruptured steam generator. Some of the assumptions which differ from the analysis described in Section 15.6.3.1.1 do so because the trip time sensitivity has been eliminated. The effect of these revised assumptions is an increase in break flow and ruptured steam generator AFW flow, which results in overfill and water relief.

The analysis scenario is outlined below. This analysis is consistent with the overfill scenario presented in Reference 3, but has been updated to match the current plant configuration. This includes revised (longer) operator action times that reflect recent simulator studies of this SGTR scenario.

An SGTR occurs while the plant is at 100% thermal power and while at steady state. Concurrent with the SGTR a reactor trip occurs and a safety injection signal is generated. A loss of offsite power (LOOP) is assumed coincident with the reactor trip.

Following reactor trip, safety injection actuation, and the loss of offsite power, the feedwater flow stops and the Main Steam Isolation Valves close. The secondary pressure rises and approaches the setpoints of the secondary ASDs and MSSVs. In response to the reactor trip and LOOP, auxiliary feedwater is delivered to the secondary. It is assumed that the AFW control valve fails full open on the ruptured SG and delivers excessive AFW to the ruptured steam generator. The excessive AFW flow quickly rebounds the ruptured steam generator water level and dr i toward overfill.

In accordance with the emergency operating procedures (EOPs), the ruptured SG is isolated by ensuring that the MSIV, ASD, and blowdown isolation valves are closed on the ruptured loop. The final isolation step requires AFW termination to the SG. After isolation, the primary and ruptured secondary pressure rise in response to reduced heat removal. Following isolation of the ruptured steam generator, operators begin cooldown of the primary via the intact steam generators' ASDs. The ASDs are conservatively assumed to open 90%. ~~An atmospheric steam dump valve on one of the unaffected steam generators is conservatively assumed not to open and will therefore be unavailable to support the RCS cooldown.~~ This assumption has the effect of increasing the time it takes to reduce the RCS temperature to below the ruptured steam generator saturation temperture. ~~This additional failure is beyond the required single failure criteria.~~ Eventually proper subcooling limits are obtained and primary depressurization is initiated using a primary power operated relief valve (PORV). Primary depressurization is performed until primary and secondary pressures equalize. This stops break flow momentarily. In accordance with EOP procedures, the safety injection flow is terminated fairly soon after the depressurization step. Unfortunately, safety injection flow, in the interim, has re-pressurized the primary and a primary/ secondary pressure difference still exists. After SI termination, it is assumed that the operators minimize the primary/secondary pressure difference by opening a PORV. Any primary rise after this step is moderate and a function of decay heat.

Primary and secondary equilibrium does not occur before the ruptured steam generator overfills and water fills the steamline up to the MSIV. When the steam generator and steamline go water solid a pressure spike (on the secondary) occurs as the primary side (driven by SI) drives the secondary pressure toward equilibrium. Thus a safety valve opens and contaminated water is dumped to the atmosphere. Water continues to be relieved from the ruptured SG MSSV until equilibrium is reached between the primary and secondary pressures, effectively terminating flow into the ruptured steam generator. To assure continued relief, an active failure of the SV is assumed to occur, i.e., after water relief the valve remains partially open (5%). Eventually, water relief depletes the secondary mass and creates a steam void. This steam void grows until water is no longer able to pass out the safety valve.

It is assumed that steam relief continues until RHR cut-in, since steam relief continues to shrink the ruptured SG mass via cooling and mass depletion. Following break flow termination it is assumed that the operators transition to the cooldown procedures and initiate cooldown via intact SG atmospheric steam dump. Cooldown to RHR cut-in conditions requires approximately ~~4 hours from initiation of intact SG atmospheric steam dump.~~ 6.4 hours from the start of the event.

15.6.3.2.2 Analysis of Ef

Method of Analysis

Mass and energy balance calculations are performed using RETRAN-3D Section 15.0.11.8 to determine the plant response to the SGTR and calculate the break flow, break flow flashing, secondary releases, and system masses for the calculation of the radiological consequences.

In the calculation of the plant response for this scenario the following assumptions are made:

a. Single failure: The ruptured steam generator's auxiliary feedwater control valve fails in the full open position.

b. Additional active failure: The ruptured steam generator's safety valve fails partially open (5% effective area) after water relief.

c. The atmospheric steam dump (ASD) valve on the ruptured SG is assumed inoperable in the closed position for the duration of the accident sequence.

d. The tube rupture is modeled as a double-ended-guillotine break of a single tube at the cold leg tube sheet. An additional 5% uncertainty is added to the flow predicted for resistance limited flow.

e. Initial conditions

- Core power = 3565 MWt
- Pressurizer pressure = 2280 psia. This is the nominal pressure plus error allowance. The higher pressure maximizes the break flow.
- Pressurizer level = 38% of narrow range span (NRS)
- Vessel average temperature = 570.7^{o}F - 3^{o}F = 567.7^{o}F. This is the minimum expected vessel average temperature. The lower temperature increases the density of the reactor coolant and thus increases the leakage.
- RCS flow = thermal design flow = 374,400 gpm
- Feedwater temperature = ~~390~~ 387.2^{o}F.

- Steam generator level = ~~57.5~~% NRS. ~~This is the nominal level plus uncertainty to maximize the initial inventory~~.
- Steam ger 80.9 ~~ube~~ plugging = 5%.

This level maximized the initial inventory

f. Reactor trip occurs at time zero.

g. Loss of offsite power (LOOP) occurs at reactor trip (i.e., at time zero)

h. MSIV isolation is modeled at reactor trip and the assumed loss of offsite power, although it could be significantly delayed based on the expected operator response. Early isolation of the MSIV allows the ruptured SG to depressurize due to the addition of the (maximum) AFW flow, while the intact SG pressure stays relatively high. This results in increased break flow to the ruptured SG, which is conservative. It also leads to higher AFW flow to the ruptured SG. If the MSIV would be left open, the ruptured and intact SGs would tend to be at the same pressure, which would be closer to that of the intact SGs (which are lumped together in the RETRAN model). Also, with the MSIV open, overfilling the ruptured SG would not necessarily lead to water relief, since the water could go to the intact SGs. The secondary pressure would not spike and the safety valve would not lift.

i. The MSIV closes in 1.5 seconds. As noted above, early isolation is considered to be more limiting.

j. The main feedwater isolation valve (MFIV) closure is modeled as a step function after a 17 second delay. The SI signal generated at reactor trip initiates the isolation. A safety e the impact of a potential increase in t n valves beyond the value assume the isolation of new valve actuators. presented in this section for a ste or faulted steam generator AFW control valve are not adversely affected by this plant modification. As such, the reported results and conclusions remain valid.

for 3840 seconds until the manual cooldown to RHR conditions is initiated. Then ramped over 1000 seconds to 1.2133

0.9706

k. Decay heat = ~~0.8~~ x 1979 ANS 2σ model

l. The following maximum AFW flow rates are modeled prior to partial/full isolation of AFW flow to the ruptured SG:

- The AFW flow to the ruptured driven AFW pump flow to the r SG pressure of 1235.7 psia is used as a base. As the intact SG

Decay heat = 1.2133 after 4840 seconds.

pressure drops the flow to the ruptured SG is reduced. This model is reflected in the table below:

Ruptured SG Pressure (psia)	AFW to Ruptured SG (gpm)		Intact SG Pressure (psia)	Reduction in AFW to Ruptured SG (gpm)
414.7	1312.6		414.7	72.6
614.7	1209.4		614.7	55.4
814.7	1099.1		814.7	37.8
1014.7	976.8		1014.7	20.0
1139.7	889.9		1139.7	8.6
1235.7	818.1		1235.7	0.0

- The AFW flow to the intact SGs (total for the 3) before isolation of the turbine driven AFW pump flow to the ruptured steam generator is provided in the table below.

Intact SG Pressure (psia)	AFW to Intact SGs (gpm)
214.7	1687.0
414.7	1569.9
614.7	1448.7
814.7	1321.1
1014.7	1202.3
1139.7	1085.9
1235.7	1008.4

m. The following maximum AFW flow rates are modeled after partial/full isolation of AFW flow to the ruptured SG:

- The AFW flow to the ruptured SG after isolation of the turbine driven AFW pump flow to the ruptured steam generator is provided in the table below:

Ruptured SG Pressure (psia)	AFW to Ruptured SG (gpm)
414.7	767.8
614.7	709.3
814.7	648.0
1014.7	583.2
1139.7	534.4
1235.7	494.9

- The AFW flow to the intact SGs (total for the 3) after isolation of the turbine driven AFW pump flow to ruptured steam generator, and after complete isolation of AFW to the ruptured SG, is provided in the table below:

Intact SG Pressure (psia)	AFW to Intact SGs (gpm)
214.7	1758.0
414.7	1649.5
614.7	1534.0
814.7	1417.8
1014.7	1288.1
1139.7	1197.5
1235.7	1122.1

n. AFW flow is initiated 5 seconds after reactor trip, with a 30-second ramp up to full flow. Quicker initiation of AFW flow provides more limiting results for this accident sequence.

1300 1192.7

o. Safety Injection modeling: High and intermediate injection pumps assumed with maximum expected flow. Injection starts 15 seconds after the SI signal (which is generated at the start of the event). Quicker initiation of AFW flow provides more limiting results for this accident sequence.

p. ~~Only two of the intact SG ASDs are credited in the RCS cooldown. This conservatively assumes an additional failure beyond single failure criteria.~~

q. Operator actions modeled:

- Isolation of turbine-driven AFW flow to the ruptured SG at 10 minutes from the start of the event.

The intact loop ASDs are assumed to only open 90%. This results in conservatively increasing the cooldown time.

minutes from the

- Initiate cooldown by dumping steam from the lumped intact loop SG ASD after 30 minutes from reactor trip (which is at the start of the event).

- The cooldown is terminated when the core outlet temperature reaches the target temperature specified in the EOPs as a function of the ruptured SG pressure.

- Initiate RCS depressurization using the pressurizer power-operated relief valves 3 minutes after the end of the RCS cooldown.

- The depressurization is terminated when the pressurizer pressure and the faulted SG pressure are equal.

- SI flow is terminated 5 minutes after the depressurization is completed.

- Depressurize using pressurizer power-operated relief valve 15 minutes after SI termination to terminate break flow.

212°F

includes

- Cooldown to RHR cut-in is initiated after break flow is terminated. The RETRAN analysis ~~does not include~~ the complete cooldown to ~~RHR conditions~~. The ~~initial part of the~~ cooldown is shown to ~~demonstrate~~ that once the cooldown is initiated the pressure differential (and break flow) is minimal.

demonstrate

r. The break flow flashing fraction is conservatively determined assuming all break flow is at the ruptured loop hot leg temperature.

At 30 minutes the operators initiate RCS cooldown by opening ~~two of~~ the intact SG ASDs. This cooldown continues until the subcooling margin appropriate to allow the primary depressurization is reached. The cooldown is completed approximately ~~45~~ 42 minutes into the event.

At approximately ~~48~~ 45 minutes operators depressurize the primary using pressurizer power operated relief valves (PORVs) until primary-secondary pressure equilibrium is reached, at approximately ~~49~~ 46 minutes. Safety injection flow is terminated 5 minutes later. A secondary RCS depressurization is initiated at approximately ~~68~~ 66 minutes from the start of the event, leading to break flow termination. Cooldown to RHR conditions using ~~two of~~ the intact SG ASDs is assumed to be initiated at approximately ~~69~~ 64 minutes from the start of the event.

Eventually, the steam void resulting from continued water relief from the assumed stuck open MSSV on the ruptured steam generator grows to the extent that the valve no longer passes water. This occurs at approximately ~~89~~ 84 minutes from the start of the event.

The following is a list of figures of pertinent time dependent parameters:

Number	Title
15.6-33a	Pressurizer and Steam Generator (Ruptured and Intact Generators) Pressure Transients for SGTR Event with Overfill
15.6-33b	Reactor Coolant System Temperature (Ruptured Loop) Transient for SGTR Event with Overfill

Number	Title
15.6-33c	Reactor Coolant System Temperature (Intact Loops) Transient for SGTR Event with Overfill
15.6-33d	Reactor Coolant System and Steam Generator (Ruptured and Intact Generators) Water Mass Transient for SGTR Event with Overfill
15.6-33e	Ruptured Steam Generator Break Flow Flashing Fraction Transient for SGTR Event with Overfill
15.6-33f	Steam Generator Temperature (Ruptured and Intact Generators) Transient for SGTR Event with Overfill
15.6-33g	Steam Generator Atmospheric Release Flow Rate (Ruptured Generator) Transient for SGTR Event with Overfill
15.6-33h	Steam Generator Atmospheric Release Flow Rate (Intact Generators) Transient for SGTR Event with Overfill

Tables 15B-1 and 15B-5 provide a comparison of the analysis to the guidelines in Regulatory Guide 1.183.

15.6.3.2.3 Radiological Consequences

The analysis of the radiological consequences of the SGTR with overfill and water release is performed in a manner consistent with that presented in Section 15.6.3.1.3 for the SGTR with the postulated stuck open ARV. The assumptions are outlined below. Unless otherwise noted, these assumptions are consistent with the Section 15.6.3.1.3 analysis assumptions.

a. Short-term accident atmospheric dispersion factors and breathing rates are provided in Tables 15A-2 and 15A-1, respectively.

b. Dose conversion factors are listed in Table 15A-4.

c. The initial reactor coolant system (RCS) iodine and noble gas concentrations are defined as in the Section 15.6.3.1.3 dose calculations.

d. Spike modeling

- The accident-initiated iodine spike is modeled as in the Section 15.6.3.1.3 dose calculations.

- The pre-accident iodine spike case spike is modeled as in the Section 15.6.3.1.3 dose calculations.

e. Initial secondary activity is 10% of the primary side activity modeled for the accident-initiated iodine spike.

f. Water/Steam Iodine Partitioning: Fluid released from the steam generators as steam retains a portion of the activity present in the fluid. The partition factor is 0.01. All activity contained in break flow that flashes to steam upon entering the SG is released without partitioning

g. Activity released with water from ruptured SG = 50%. Activity contained in water released from the ruptured SG after overfill is not subject to partitioning. However, only 50% of the activity contained in the water is assumed to become airborne. No additional activity release due to evaporation is modeled. These assumptions were made in the analysis approved in Reference 3.

h. ~~Break flow rate for iodine doses:~~

- ~~The Section 15.6.3.1.3 dose analysis conservatively modeled a constant break flow rate. For the analysis of doses for the overfill case the transient break flow rate from the RETRAN analysis presented in Figure 15.6-33i is used, up until the time when water relief stops. This is consistent with the analysis approved in Reference 3.~~

- ~~The calculation of iodine doses until RHR conditions are reached conservatively assumes a break flow of 4 lbm/sec until 5 hours after break flow termination. This is consistent with the analysis approved in Reference 3. This portion of the analysis assumes that RHR conditions are achieved within 5 hours of break flow termination, even thought the intact SG releases and the noble gas releases assume 8 hours.~~

h. The ~~noble gas~~ doses are calculated in Section 15.6.3.1.3 assuming a constant break flow rate of ~~65~~ 60 lbm/sec for the first ~~hour~~ 4,000 seconds of the transient ~~and 10 lbm/sec thereafter for 8 hours. For the analysis of the SGTR with overfill the duration of the 65 lbm/sec break flow is extended until 2 hours~~.

i. Break flow flashing fraction

- The Section 15.6.3.1.3 dose analysis modeled conservative bounding values for the flashing fraction. For the analysis of doses for the overfill case the transient flashing fraction from the RETRAN analysis presented in Figure 15.6-33e ~~is used. This is consistent with the analysis approved in Reference 3. This analysis models the release of all the activity contained in the flashed break flow. This conservative assumption is consistent with Section 15.6.3.1.3 which modeled the direct release of activity in flashed break flow even after the ruptured SG's failed open atmospheric steam dump (ASD) valve was isolated~~.

and secondary side water level in Figure 15.6-33j are used. Figure 15.6-33j shows that the tube bundle in the ruptured SG effectively remains covered for the duration of the accident. On this basis, credit is taken for scrubbing of the flashed fraction

Primary to secondary leakage flashing is conservatively modeled as 4% between 6,000 and 9,000 seconds to account for the intact steam generators being partially uncovered during this time period. There is no flashing modeled outside of this time period consistent with Reference 3.

~~Primary to secondary leakage flashing is not modeled. This is consistent with the analysis approved in Reference 3.~~ The leak is small and it is assumed that any steam bubbles formed by flashing leakage would collapse before reaching the top of the water level.

j. The ruptured SG releases are modeled using the RETRAN analysis results presented in Figure 15.6-33g. ~~In the calculation of doses for the cooldown to RHR conditions it is assumed that a steam flow rate of 8 lbm/sec is maintained due to the failed open safety valve. Thus, 144,000 lbm of steam is released from the ruptured SG in the 5 hours from break flow termination until RHR c~~

The total steam released from the ruptured SG is 292,353 lbm over 9.4 hours.

k. The intact SG releases are modeled using the RETRAN analysis results presented in Figure 15.6-33h. ~~In the calculation of doses for the cooldown to RHR conditions it is assumed that 1.25E6 lbm of steam is released from the intact SGs from the time of break flow termination. This value was conservatively calculated for the case with the failed open ASD presented in Section 15.6.3.1.3 and remains conservative when applied to the analysis of the SGTR with overfill.~~

l. The reactor coolant system, ruptured steam generator and intact steam generators' masses are modeled using the RETRAN analysis results from Figure 15.6-33d. ~~This is consistent with the analysis approved in Reference 3.~~ The analysis presented in the Section 15.6.3.1.3 modeled conservative bounding values for the RCS and secondary masses.

Table 15.6-5a lists the offsite doses for the SGTR with overfill and water release.

The total steam released from the intact steam SGs is 977,502 lbm over 6.4 hours, with 396,435 LBM released in the first two hours and 581,117 lbm from the two hours until the RHR cut-in conditions are met.

C o

15.6.3.3.1 Filter Loadings

The only ESF filtration system considered in the analysis which limits the consequences of the steam generator tube rupture is the control room filtration system. Activity loadings on the control room charcoal filter are based on flow rate through the filter, concentration of activity at the filter inlet, and filter efficiency.

Activity in the control room filter as a function of time has been evaluated for the LOCA, Section 15.6.5. Since the control room filters are capable of accommodating the potential design-basis LOCA fission product iodine loadings, more than adequate design margin is available with respect to postulated SGTR accident releases.

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15.6.3.3.2 Doses to Receptor at the Exclusion Area Boundary ~~and~~ Low-Population Zone Outer Boundary

and in the Control Room

T worst 2 radiological consequences resulting from the occurrence of a postulated SGTR have been conser TEDE doses ed, using assumptions and models described in previous sections.

The ~~total-body dose due to immersion and the thyroid dose due to inhalation~~ have been analyzed for the ~~0-2~~ hour period at the exclusion area boundary and for ~~a time period effectively greater than~~ the duration of the accident ~~(0 to 8 hours)~~ at the low-population zone outer boundary. Two potentially limiting failure scenarios have been analyzed.

, and for 30 days in the control room

esents the offsite dose results for the case of an SGTR with a stuck-open ured steam generator. Table 15.6-5a presents the offsite dose results for the case of an SGTR with the postulated failure of the ruptured steam generator AFW flow control valve. For both scenarios, the doses considering a pre-accident iodine spike are within the guideline values of 10 CFR ~~100~~. For both scenarios, the doses considering an accident-initiated iodine spike are within the ~~10% of the~~ guideline values of 10 CFR ~~100~~.

50.67 and Regulatory Guide 1.183

Conclusions

A steam generator tube rupture will cause no subsequent damage to the RCS or the reactor core. An orderly recovery from the accident can be completed, even assuming simultaneous loss of offsite power.

50.67 and Regulatory Guide 1.183

15.6.4 SPECTRUM OF BWR STEAM SYSTEM PIPING FAILURES OUTSIDE OF CONTAINMENT

This section is not applicable to the Callaway Plant.

15.6.5 LOSS-OF-COOLANT ACCIDENTS RESULTING FROM A SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

15.6.5.1 Identification of Causes and Frequency Classification

A loss-of-coolant accident (LOCA) is the result of a pipe rupture of the RCS pressure boundary. For the analyses reported here, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft2. This event is considered an ANS Condition IV event, a limiting fault, in that it is not expected to occur during the lifetime of the plant, but is postulated as a conservative design basis.

For large-break LOCAs, the most limiting single failure is the loss of one train of ECCS injection. The large-break LOCA analyses assume both maximum containment safeguards (to analyze lowest containment pressure conditions) and minimum ECCS safeguards (to analyze the loss of one complete train of emergency core cooling system

15.6-32 4 Inch broken loop (BL) & intact loop (IL) Pumped SI Flow Rate Vs. Time

The peak cladding temperature calculated for the limiting small break LOCA is 1043°F. The maximum local metal-water reaction is 0.02 percent which is below the acceptance criteria limit of 17 percent. The total core metal-water reaction is less than 0.01 percent which is much less than the 1 percent acceptance criteria. These results are below all acceptance criteria limits of 10 CFR 50.46.

15.6.5.4 <u>Radiological Consequences</u>

Insert 15.6.5.4.1

15.6.5.4.1 Method of Analysis

15.6.5.4.1.1 Containment Leakage Contribution

PHYSICAL MODEL - Following a postulated double-ended rupture of a reactor coolant pipe with subsequent blowdown, the ECCS limits the clad temperature to well below the melting point and ensures that the reactor core remains intact and in a coolable geometry, minimizing the release of fission products to the containment. However, to demonstrate that the operation of the Callaway Plant does not represent any undue radiological hazard to the general public, a hypothetical accident involving a significant release of fission products to the containment is evaluated.

It is assumed that ~~100 percent of the noble gases and 50 percent of~~ the ~~iodine~~ equilibrium core saturation fission product inventory is ~~immediately~~ released in phases to the containment atmosphere. ~~Of the iodine released to the containment, 50 percent is assumed to plateout onto the internal surfaces of the containment or adhere to internal components.~~ The ~~remaining iodine and the noble gas~~ activity ~~are~~ released from the core during each of the release phases is assumed to be immediately available for leakage from the containment.

Once the ~~gaseous~~ fission product activity is released to the containm is subject to various mechanisms of removal which operate simultaneously to reduce the amount of activity in the containment. The removal mechanisms include radioactive decay, containment sprays, and containment leakage. For the noble gas fission products, the only removal processes considered in the containment are radioactive decay and containment leakage.

a. Radioactive Decay - Credit for radioactive decay for fission product concentrations located within the containment is assumed throughout the course of the accident. Once the activity is released to the environment, no credit for radioactive decay or deposition is taken.

b. Containment Sprays - The containment spray system is designed to absorb airborne iodine fission products within the containment atmosphere. To enhance the ~~iodine-retention~~ iodine-removal and retention capability of the containment sprays, trisodium phosphate is added to the spray solution via baskets adjacent to the sumps. The spray effectiveness for the retention of iodine is dependent on maintaining a long-term sump pH greater than 7.0.

c. Containment Leakage - The containment leaks at a rate of 0.2 volume percent/day as incorporated as a Technical Specification requirement at peak calculated internal containment pressure for the first 24 hours and at 50 percent of this leak rate for the remaining duration of the accident. The containment leakage is assumed to be directly to the environment.

1.183

ASSUMPTIONS AND CONDITIONS - The major assumptions and parameters assumed in the analysis are itemized in Tables 15A-1 and 15.6-6 and discussed in Section 6.5A.3.

In the evaluation of a LOCA, all the fission product release assumptions of Regulatory Guide ~~1.4~~ have been followed. The following specific assumptions were used in the analysis. ~~Table 15.6-7 provides a comparison of the analysis to the requirements of Regulatory Guide 1.4.~~ The nuclide groups and their release fractions are presented in Table 15.6-6

a. The reactor core equilibrium noble gas and iodine inventories are based on long-term operation at a core power level of 3,636 MWt. (includes 2% uncertainty)

b. ~~One hundred percent of the core equilibrium radioactive noble gas inventory is immediately available for leakage from the containment~~.

~~c.~~ ~~Twenty-five percent of the core equilibrium radioactive iodine inventory is immediately available for leakage from the containment. The other 25% released to the containment atmosphere instantaneously plates out.~~

c. Of the iodine fission product inventory released to the containment, ~~91~~ 4.85 percent is in the form of elemental iodine, ~~5~~ 95 ~~percent is in the form of~~ particulate iodine, and ~~4~~ 0.15 percent is in the form of organic iodine.

d. Credit for ~~iodine~~ removal of iodine and particulates by the containment spray system is taken, starting ~~at time zero~~ 2 minutes after event initiation and continuing ~~until a decontamination factor of 28.7 for the elemental species and 50 for the particulate species has been achieved~~ for four hours, which occurs before a decontamination factor of 200 for the elemental species. Credit for the particulate species removal is continued for the duration of spray but is reduced by a factor of 10 after a decontamination factor of 50 is achieved.

e. The following iodine removal constants for the containment spray system are assumed in the analysis:

Elemental iodine	-	~~10.0~~ 20.0 per hr
Organic iodine	-	0.0 per hr
~~Particulate iodine~~ Particulates	-	~~0.45 per hr~~ 6.46 per hr prior to a DF of 50 0.646 per hr after a DF of 50

f. The following parameters were used in the two-region spray model:

Fraction of containment sprayed - 0.85
Fraction of containment unsprayed - 0.15
Mixing rate (cfm) between sprayed and unsprayed regions - ~~85,000~~ 13,500

Section 6.5 contains a detailed analysis of the sprayed and unsprayed volumes ~~and~~ Table 6.5-2 includes an explanation of the mixing rate between the sprayed and unsprayed regions.

g. The containment is assumed to leak at 0.2 volume percent/day during the first 24 hours immediately following the accident and 0.1 volume percent/ day thereafter.

h. The containment leakage is assumed to be direct unfiltered to the environment.

i. The ~~control building and~~ control room filters will be 95 percent efficient in the removal of all species of iodine. The emergency exhaust ESF filter efficiency is 90% in the assumptions listed in Table15.6-6.

MATHEMATICAL MODELS USED IN THE ANALYSIS - Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Section 15A.2.

b. The atmospheric dispersion factors used in the analysis were calculated, based on the onsite meteorological measurements program described in Section 2.3 of the Site Addendum, and are provided in Table15A-2.

c. The ~~thyroid inhalation and total-body immersion~~ TEDE doses to a receptor exposed at the exclusion area boundary ~~and~~ , the outer boundary of the low population zone and in the control room were analyzed, using the models described in Sections ~~15A.2.4 and 15A.2.5, respectively~~ 15A.2, 15A.3 and 15A.4.

d. Buildup of activity in the control room and the integrated doses to the control room personnel are analyzed, based on models described in Section 15A.3.

IDENTIFICATION OF LEAKAGE PATHWAYS AND RESULTANT LEAKAGE ACTIVITY - For evaluating the radiological consequences of a postulated LOCA, the resultant activity released to the containment atmosphere is assumed to leak directly to the environment.

No credit is taken for ground deposition or radioactive decay during transit to the exclusion area boundary ~~or~~ , LPZ outer boundary or control room.

The offsite doses from all those pathways at the exclusion area boundary (EAB) and low population zone (LPZ) boundary and the doses to the control room personnel are included within the composite results reported in Table 15.6-8.

15.6.5.4.1.2 Radioactive Releases Due to Leakage from ECCS and Containment Spray Recirculation Lines

Subsequent to the injection phase of ESF system operation, the water in the containment recirculation sumps is recirculated by the residual heat removal, ECCS centrifugal charging and safety injection pumps, and the containment spray pumps. Due to the operation of the ECCS and the containment spray system, most of the radioiodine released from the core would be contained in the containment sump. It is conservatively assumed that a leakage rate of 2 gpm from the ECCS and containment spray recirculation lines exists for the duration of the LOCA. This leakage would occur inside the containment as well as inside the auxiliary building. For this analysis, all the leakage is assumed to occur inside the auxiliary building. Only trace quantities of radioiodine are expected to be airborne within the auxiliary building due to the temperature and pH level of the recirculated water. However, 10 percent of the radioiodine in the leaked water is assumed to become airborne and exhausted from the unit vent to the environment through the auxiliary building emergency exhaust filters (90% efficient). No credit is taken for holdup (i.e. decay) or mixing in the auxiliary building; however, mixing and holdup in the sumps are factored into the release and decay removal constants for this pathway.

Radiological Consequences of ECCS/CS Recirculation Line Leakage - The assumptions used to calculate the amount of radioiodine released to the environment are given in Table 15.6-6. The dose models are presented in Section 15.A. The offsite doses from all dose pathways at the exclusion area boundary (EAB) and low population zone (LPZ) boundary and the doses to the control room personnel are included within the composite results reported in Table 15.6-8.

4 gpm (3 gpm below the water line, 1 gpm above the water line)

15.6.5.4.1.3 Releases Due to Leakage of Radioactive Iodine from the RWST

An assessment was performed to calculate the thyroid doses at the exclusion area boundary (EAB), low population zone (LPZ) outer boundary, and to the control room personnel associated with an assumed ~~3 gpm~~ leakage pathway from the containment recirculation sumps through ECCS isolation valves back to the RWST, which is vented to the atmosphere. This calculation was performed to address the scenario presented in Reference 25.

for the first 24 hours and 8% for the remainder of the event

above the water line

The calculation assumed that 10% of the radioiodine leaked to the RWST becomes airborne, mixes with the RWST volume, and is released to the environment. Credit is taken for decay in the RWST. The assumptions used to calculate the amount of radioiodine released to the environment are given in Table 15.6-6. ~~The dose models are presented in Section 15.A. The doses at the EAB, LPZ outer boundary, and to control room personnel are less than the values reported in Table 15.6-8.~~

air

The vast majority of the radioiodine in the 3 gpm delivered below the water line is retained in the liquid remaining in the RWST. This retention in the liquid is supported by a calculation performed in accordance with NUREG/CR-5950, accounting for gradual changes in pH and iodine concentration in the RWST liquid.

The offsite doses from all dose pathways at the exclusion area boundary (EAB) and low population zone (LPZ) boundary and the doses to the control room personnel are included within the composite results reported in Table 15.6-8.

15.6.5.4.1.4 Releases Prior to Containment Purge Isolation

Operation of the containment mini-purge system is allowed during power operation. Therefore, during the initial stage of the LOCA sequence, it is possible that the containment mini-purge system would not be isolated. Table 9.4-13 discusses NRC guidance regarding modeling of the potential contribution that this pathway would make to post-LOCA radiological consequences.

An assessment was performed to calculate the doses at the exclusion area boundary (EAB), low population zone (LPZ) outer boundary, and to the control room personnel associated with this pathway.

Insert 15.6.5.4.1.4

~~The calculation assumed that the initial radioiodine concentration in the reactor coolant system fluids is the same as used for the pre-accident spike cases analyzed for the SGTR and MSLB accidents. The blowdown rate of RCS fluids into the reactor building is taken from FSAR Table 6.2.1-32. The assumptions used to calculate the amount of radioactivity released to the environment are given in Table 15.6-6.~~ The offsite doses from all dose pathways at the exclusion area boundary (EAB) and low population zone (LPZ) boundary and the doses to the control room personnel are included within the composite results reported in Table 15.6-8.

15.6.5.4.2 Identification of Uncertainties and Conservatisms in the Analysis

The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of a LOCA result principally from assumptions made involving the amount of the gaseous fission products available for release to the environment and the meteorology present at the site during the course of the accident. The most significant of these assumptions are:

a. The ECCS is designed to prevent fuel cladding damage that would allow the release of the fission products contained in the fuel to the reactor coolant. Severe degradation of the ECCS (i.e., to the unlikely extent of simultaneous failure of redundant components) would be necessary in order for the release of fission products to occur of the magnitude assumed in the analysis.

for the sprayed region, it

b. ~~The release of fission products to the containment is assumed tooccur instantaneously.~~

none

b. ~~It~~ is assumed that ~~50 percent~~ of the iodines released to the containment atmosphere is adsorbed onto the internal surfaces of the containment or adheres to internal components; however, it is ~~estimated~~ that theremoval

recognized

of airborne iodines by various physical phenomena such as adsorption, adherence, and settling could reduce the resultant doses ~~by a factor of 3 to 10 (Ref. 20)~~.

c. The activity released to the containment atmosphere is assumed to leak to the environment at the containment leakage rate of 0.2-volume percent/ day for the first 24 hours and 0.1-volume percent/day thereafter. The initial containment leakage rate is based on the peak calculated internal containment pressure anticipated after a LOCA. The pressure within the containment actually decreases with time. Taking into account that the containment leak rate is a function of pressure, the resultant doses could be significantly reduced ~~by a factor of 5 to 10 (Ref. 20)~~.

Insert 15.6.5.4.2

~~e.~~ ~~The meteorological conditions assumed to be present at the site during the course of the accident are based on χ/Q values, which are expected to be exceeded 5 percent of the time. This condition results in the poorest values of atmospheric dispersion calculated for the exclusion area boundary and the LPZ outer boundary. Furthermore, no credit has been taken for the transit time required for activity to travel from the point of release to the exclusion area boundary and LPZ outer boundary. Hence, the radiological consequences evaluated under these conditions are conservative.~~

~~Limited credit has been taken for the transit time required for activity to travel from the point of release to the control room air intake ductwork.~~ Since the safety injection signal will generate a Phase A containment isolation signal, which in turn will generate a control room ventilation isolation signal prior to activity reaching the control room air intake ductwork, there is no requirement to perform response time testing on the control room ventilation isolation functions for LOCAmitigation.

15.6.5.4.3 Conclusions

15.6.5.4.3.1 Filter Loadings

No recirculating or single-pass filters are used for fission product cleanup and control within the containment following a postulated LOCA. The only ESF filtration systems expected to be operating under post-LOCA conditions are the control room HVAC system and the auxiliary building emergency exhaust filtration system.

Activity loadings on the control room charcoal adsorbers are based on the flowrate through the adsorber, the concentration of activity at the adsorber inlet, and the adsorber efficiency. Based on the radioactive iodine release assumptions previously described, the assumption that 25 percent of the core inventory of isotopes I-127 and I-129 is available for release from the containment atmosphere and the assumption that the charcoal adsorber is 100 percent efficient, the calculated filter loadings are in accordance with Regulatory Guide 1.52, which limits the maximum loading to 2.5 mg of iodine per gram of activated charcoal. The 100 percent efficiency assumption is conservative for the purpose of checking filter loading and is not to be confused with the 95% efficiency assumption used for radiological consequences as listed in Table 15.A-1.

15.6.5.4.3.2 Doses to a Receptor at the Exclusion Area Boundary and Low Population Zone Outer Boundary

The potential radiological consequences resulting from the occurrence of the postulated LOCA have been conservatively analyzed, using assumptions and models described in previous sections.

The ~~total-body dose due to immersion and the thyroid dose due to inhalation have~~ TEDE has been analyzed for the ~~0-2~~ worst 2 hour dose at the exclusion area boundary and for the duration of the accident at the LPZ outer boundary. The results, with margin, are listed in Table 15.6-8. The resultant doses are within the guideline values of 10 CFR ~~100~~ 50.67 and Regulatory Guide 1.183.

15.6.5.4.3.3 Doses to Control Room Personnel

Radiation doses to control room personnel following a postulated LOCA are based on the ventilation, cavity dilution, and dose model discussed in Section 15A.3.

Control room personnel are subject to a ~~total-body dose due to immersion and a thyroid dose due to inhalation. These doses have~~ TEDE. This dose has been analyzed, and ~~are~~ is provided in Table 15.6-8. The listed dose~~s~~, with margin, ~~are~~ is within the limits established by GDC-19 and 10 CFR 50.67.

15.6.6 A NUMBER OF BWR TRANSIENTS

This section is not applicable to the Callaway Plant.

15.6.7 REFERENCES

1. Huegel, D. S., et. al., “RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses,” WCAP-14882-P-A, April 1999.

2. Friedland, A. J., and Ray, S., "Revised Thermal Design Procedure", WCAP-11397-P-A, April 1989.

3. SGTR Analysis letters SLNRC 86-01 (1-8-86), SLNRC 86-03 (2-11-86) SLNRC 86-05 (4-1-86), SLNRC 86-08 (9-4-86), ULNRC-1442 (2-3-87), ULNRC-1518 (5-27-87), ULNRC-1849 (10-21-88), ULNRC-2145 (1-29-90), and the NRC SER dated 8-6-90.

4. WCAP-16140, “Callaway Replacement Steam Generator Program NSSS Engineering Report,” June 2004.

30. Huegel, D. S., et. al., “RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Reactor Non-LOCA Safety Analyses”, WCAP-14882-P-A (Proprietary), April 1999.

31. International Commission on Radiological Protection, “Limits for Intakes of Radionuclides by Worker,” ICRP Publication 30, 1979.

32. Regulatory Guide 1.195, “Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors,“ May 2003.

33. K. F. Eckerman and J. C. Ryman, “External Exposure to Radionuclides in Air, Water, and Soil,” Federal Guidance Report 12, EPA-402-R-93-081, Environmental Protection Agency, 1993.

34. EPA Federal Guidance Report No. 11, EPA-520/1-88-020, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988.

35. EPA Federal Guidance Report No. 12, EPA-402-R-93-081, "External Exposure to Radionuclides in Air, Water, and Soil," 1993.

36. NUREG/CR-5950, "Iodine Evolution and pH Control", ORNL/TM-12242, Revision 3, December 1992.

TABLE 15.6-1 TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN REACTOR COOLANT INVENTORY

Insert T15.6-1

Accident	Event	Time (sec)
Inadvertent opening of a pressurizer safety valve	Safety valve opens fully	0.0
	Low pressurizer pressure reactor trip setpoint reached	32.8
	Rods begin to drop	34.8
	Minimum DNBR occurs	35.5
~~Steam generator tube rupture with stuck-open atmospheric steam dump (ASD) valve~~	~~Tube rupture occurs~~	~~0.0~~
	~~Reactor trip signal~~	~~597~~
	~~Safety injection signal~~	~~597~~
	~~Rod motion~~	~~599~~
	~~Feedwater terminated~~	~~603~~
	~~Ruptured steam generator atmospheric/ steam dump valve opens~~	~~604~~
	~~Safety injection begins~~	~~612~~
	~~Auxiliary feedwater injection~~	~~659~~
	~~Operator isolates ruptured steam generator by closing manual block valve~~	~~1804~~
	~~Operator initiates RCS cooldown via intact steam generator atmospheric steam dump valves~~	~~2404~~
	~~Operator completes RCS cooldown~~	~~3383~~
	~~Operator initiates RCS depressurization via pressurizer PORVs~~	~~3563~~
	~~Operator completes RCS depressurization~~	~~3622~~
	~~Operator terminates safety injection~~	~~3922~~
	~~Operator equalizes primary-secondary pressure~~	~~4816~~
	~~RHR cut-in conditions reached~~	~~18000~~

TABLE 15.6-1 (Sheet 2)

Accident	Event	Time (sec)
Steam generator tube rupture with overfill	Tube rupture occurs	0.
	Reactor trip signal and loss of offsite power	0.
	Safety injection signal	0.
	Auxiliary feedwater injection starts	5.
	Safety injection delivered	15.
	Feedwater terminated	17.
	Operator terminates auxiliary feedwater from TDAFW pump to ruptured steam generator	600.
	Ruptured steam generator water relief begins	957.
	Operator terminates auxiliary feedwater from MDAFW pump to ruptured steam generator	1200.
	Operator initiates RCS cooldown via intact steam generator atmosperhic steam dump valves	1800.
	Operator completes RCS cooldown	2495.0
	Operator initiates RCS depressurization via pressurizer PORVs	2675.
	Operator completes RCS depressurization	2741.0
	Operator terminates safety injection	3041.
	Cooldown to RHR cut-in begins	3840.
	Operator equalizes primary-secondary pressure	3941.
	Ruptured SG safety valve begins to relieve steam	5020.
	RHR cut-in conditions reached	23029.
	Ruptured SG reaches 212°F	33804

TABLE 15.6-2 PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCE OF THE CVCS LETDOWN LINE RUPTURE OUTSIDE OF CONTAINMENT

with an assumed iodine spike that increases the rate of iodine release into the reactor coolant by a factor of 500. This increased rate is assumed for 8 hours.

			3636 (includes 2% uncertainty)
	b.	Reactor coolant initial iodine activity	Dose equivalent of 1.0 µCi/gm of I-131 (~~adjusted consistent with Table 15.6-4 item I.C.1~~)
	c.	Reactor coolant initial noble gas activity	~~Based on 1-percent fuel defects. See Table 15A-5~~. Dose equivalent of 225 µCi/gm Xe-133
	d.	Iodine spiking factor	500
II.		Atmospheric Dispersion Factors	See Table 15A-2.
III.		Activity Release	
	a.	Break flow rate, gpm	158.9
	b.	Duration, secs	1810
	c.	Fraction of iodine and alkali metal activity in the spill that is airborne	~~0.10~~ 0.20

TABLE 15.6-3 RADIOLOGICAL CONSEQUENCES OF A CVCS LETDOWN LINE BREAK OUTSIDE OF CONTAINMENT

	Doses (rem) TEDE
Exclusion Area Boundary (0-2 hr)	3.88E-01
~~Thyroid~~	~~5.5E+00~~
~~Whole body~~	~~1.9E-01~~
Low Population Zone Outer Boundary (duration)	1.33E-01
~~Thyroid~~	~~5.5E-01~~
~~Whole body~~	~~1.9E-02~~
Control Room (30 days)	1.93

TABLE 15.6-4 PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE (SGTR)

I. Source Data	
a. Core power level, MWt	3,636 (includes 2% uncertainty)
b. Steam generator tube leakage, gpm	1
c. Reactor coolant iodine activity:	
1. Case 1	The initial reactor coolant iodine activity corresponds to an isotope mixture that bounds Technical Specification allowable conditions for both tight and open fuel defects. The isotopic mix is based on the initial RCS concentrations from Table 15A-5. This table provides conservative values for the iodine isotopic spectrum that bound the RCS concentrations which could be expected with either tight or open fuel defects. Case 1 then includes an accident initiated, spiked release rate that increases by a factor of 335 during the accident sequence.
2. Case 2	The initial reactor coolant iodine activity corresponds to an assumed pre-accident iodine spike which results in concentrations that are a factor of 60 higher than those used in Case 1.
d. Reactor coolant noble gas activity, both cases	~~Based on 1-percent failed fuel as provided in Table 15A-5 (~~225 μCi/gm DOSE EQUIVALENT XE-133~~)~~.
e. Secondary system initial activity	10% of Case 1 primary side activity
f. Reactor coolant mass in total primary system, lbs	5.8E+5
g. Steam generator water mass (each), lbs	9.3E+4
h. Offsite power	Lost
i. Primary-to-secondary leakage duration	~~80 minutes~~ 5.86 hours
II. Atmospheric Dispersion Factors	See Table 15A-2

TABLE 15.6-4 (Sheet 2)

Insert T15.6-4sheet2

III. ~~Activity Release Data~~	
a. ~~Affected steam generator~~	
1. ~~Reactor coolant discharged to steam generator, lbs~~	~~486,000(1)~~
2. ~~Flashed reactor coolant, percent~~	~~16(2)~~
3. ~~Iodine partition factor for flashed fraction of reactor coolant~~	~~1.0~~
4. ~~Steam release to atmosphere, lbs~~	
~~0-2 hrs~~	~~123,200~~
~~2-8 hrs~~	~~0~~
5. ~~Iodine carryover factor for the nonflashed fraction of reactor coolant that mixes with the initial iodine activity in the steam generator~~	~~0.01~~
b. ~~Unaffected steam generators~~	
1. ~~Primary-to-secondary leakage, lbs~~	~~4,032(3)~~
2. ~~Flashed reactor coolant, percent~~	~~Variable~~
3. ~~Total steam release, lbs~~	
~~0-2 hours~~	~~1.53E+6(4)~~
~~2-8 hours~~	~~0~~
4. ~~Iodine carryover factor~~	~~0.01(5)~~
5. ~~RHR Cut-in time, hrs~~	~~5~~

Notes:

TABLE 15.6-4 (Sheet 3)

1. ~~The noble gas release calculation assumed a conservatively high, constant 65 lbm/sec break flow rate for the first hour and 10 lbm/sec thereafter through 8 hours, even though RHR operation is assumed to begin at 5 hours. The iodine release calculation is based on the conservative break flow rate of 65 lbm/sec until cooldown is completed.~~

2. ~~The assumed flashed fraction is 16% until closure of the SG atmospheric steam dump block valve. Following closure of the block valve, a variable flashed fraction is assumed which conservatively bounds the values calculated by the RETRAN code. The intact steam generator flashed fraction is conservatively assumed to be the same as in the ruptured steam generator.~~

3. ~~Based on 1 gpm leakage and conservative density of 62.4 lbm/cu.ft., giving a mass flow rate of 0.14 lbm/sec for 8 hours, even though RHR operation is assumed to begin at 5 hours.~~

4. ~~Assumes that 1.25E06 lbm of steam is relieved for decay heat removal during 5 hour cooldown to RHR operating conditions. To maximize dose effects, this release is included in the first two hours following tube rupture.~~

5. ~~A partition factor of 1.0 is assumed for the flashed fraction.~~

Insert
T15.6-4sheet3

Control Room (30 days) 5.9E-01

TABLE 15.6-5 RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE WITH STUCK-OPEN ATMOSPERHIC STEAM DUMP VA

		TEDE Doses (rem)
1.	Case 1, accident initiated iodine spike	
	Exclusion Area Boundary (~~0-2~~ 0.1-2.1 hr)	1.38
	~~Thyroid~~	~~2.3E01~~
	~~Whole body~~	~~8.0E-01~~
	Low Population Zone Outer Boundary (duration)	5.0E-01
	~~Thyroid~~	~~2.3E00~~
	~~Whole body~~	~~8.5E-02~~
	Control Room (30 days)	5.9E-01
2.	Case 2, pre-accident iodine spike	
	Exclusion Area Boundary (0-2 hr)	2.00
	~~Thyroid~~	~~5.9E01~~
	~~Whole body~~	~~5.4E-01~~
	Low Population Zone Outer Boundary (duration)	7.2E-01
	~~Thyroid~~	~~5.9E00~~
	~~Whole body~~	~~5.8E-02~~
	Control Room (30 days)	2.44

Control Room (30 days) 2.44

Control Room (30 days) 6.9E-01

TABLE 15.6-5A RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE WITH OVERFILL

		Doses (rem)
1.	Case 1, accident initiated iodine spike	
	Exclusion Area Boundary (~~0-2 hr~~ 0.2 - 2.2 hrs)	2.13
	~~Thyroid~~	~~2.3E01~~
	~~Whole body~~	~~7.5E-01~~
	Low Population Zone Outer Boundary (duration)	7.3E-01
	~~Thyroid~~	~~2.4E00~~
	~~Whole body~~	~~7.5E-02~~
	Control Room (30 days)	6.9E-01
2.	Case 2, pre-accident iodine spike	
	Exclusion Area Boundary (0-2 hr)	3.50
	~~Thyroid~~	~~7.1E01~~
	~~Whole body~~	~~6.3E-01~~
	Low Population Zone Outer Boundary (duration)	1.21
	~~Thyroid~~	~~8.0E00~~
	~~Whole body~~	~~6.3E-02~~
	Control Room (30 days)	1.60

TABLE 15.6-6 PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT-ACCIDENT

Insert T15.6-6A

I.	Source Data				
	a.	Core power level, MWt		3636	(includes 2% uncertainty)
	b.	Burnup, full power days		~~1,000~~ 547	
	~~c.~~	~~Percent of core activity initially airborne in the containment~~			
		~~1. Noble gas~~		~~100~~	
		~~2. Iodine~~		~~50*~~	
	d.	Percent of core activity ~~immediately deposited~~ in containment			
		1. Noble gas		0	
		2. Iodine		~~50~~	see Item c.2 above
		3. Other Particulates		100 (remains in liquid)	
	e.	Core inventories		Table 15A-3	
	f.	Iodine distribution, percent	Containment Leakage Release		ECCS and RWST Release
		1. Elemental	4.85	~~91~~	97
		2. Organic	0.15	~~4~~	3
		3. Particulate	95	~~5~~	0

Insert T15.6-6B

II.	Atmospheric Dispersion Factors			See Table 15A-2	
III.	Activity Release Data				
	a.	Containment leak rate, volume percent/day			
		1. 0-24 hours		0.20	
		2. 1-30 days		0.10	
	b.	Percent of containment leakage that is unfiltered		100	
	c.	Credit for containment sprays			
		1. Spray ~~iodine~~ removal constants (per hour)			
		a. Elemental iodine		~~10.0~~ 20.0	
		b. Organic iodine		0.0	
		c. Particulates		~~0.45~~	6.46 prior to DF limit 0.646 thereafter

TABLE 15.6-6 (Sheet 2)

Key

2. ~~Maximum iodine~~ decontamination factors for the containment atmosphere

	a.	Elemental	~~20.7~~ 200
	b.	Organic	0
	c.	Particulate	50
3.	Sprayed volume, percent		85
4.	Unsprayed volume, percent		15
5.	Sprayed-unsprayed mixing rate, CFM		~~85,000~~ 13,500
6.	Containment volume, ft^3		~~2.5E+6~~ 2.7E+6

e. ECCS recirculation leakage

1.	Leak rate (0 - 30 days), gpm	2.0
2.	initial air volume at beginning of recirculation [Volume], ft^3	57,224
3.	Fraction iodine airborne	0.1
4.	Emergency exhaust ESF filter efficiency, %	90.0

f. RWST leakage

1.	Leak rate (0 - 30 days), gpm	~~3.0~~ 4 gpm total 3 gpm below water line 1 gpm above water line
2.	RWST ~~volume~~ initial air volume at beginning of recirculation, gal.	~~400,000~~ 256,626
3.	Fraction iodine airborne	0.1 (of 1 gpm above water line) for first 24 hours 0.05 (of 1 gpm above water line) for remainder of event

IV. Control room parameters — Tables 15A-1 and 15A-2

Insert T15.6-6C

* ~~Half instantaneously plates out leaving 25% immediately available for leakage from the containment.~~

Note: The release rate from the RWST to the environment is based on the volume displacement from the incoming leakage.

Delete

TABLE 15.6-7 DESIGN COMPARISON TO THE REGULATORY POSITIONS OF REGULATORY GUIDE 1.4 "ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS OF COOLANT ACCIDENT FOR PRESSURIZED WATER REACTORS," REVISION 2, JUNE 1974

Regulatory Guide 1.4 Position	Design
1. The assumptions related to the release of radioactive material from the fuel and containment are as follows:	
a. Twenty-five percent of the equilibrium radioactive iodine inventory developed from maximum full power operation of the core should be assumed to be immediately available for leakage from the primary reactor containment. Ninety-one percent of this 25 percent is to be assumed to be in the form of elemental iodine, 5 percent of this 25 percent in the form of particulate iodine, and 4 percent of this 25 percent in the form of organic iodides.	1a. Complies.
b. One hundred percent of equilibrium radioactive noble gas inventory developed from maximum full power operation of the core should be assumed to be immediately available for leakage from the reactor containment.	1b. Complies.
c. The effects of radiological decay during holdup in the containment or other buildings should be taken into account.	1c. Complies. Credit for radioactive decay is taken until the activity is assumed to be released.
d. The reduction in the amount of radioactive material available for leakage to the environment by containment sprays, recirculating filter systems, or other engineered safety features may be taken into account, but the amount of reduction in concentration of radioactive materials should be evaluated on an individual case basis.	1d. Complies. See Table 15.6–6 for reduction taken.

TABLE 15.6-7 (Sheet 2)

Regulatory Guide 1.4 Position	Design
e. The primary reactor containment should be assumed to leak at the leak Delete incorporated or to be incorporated as a technical specification requirement at peak accident pressure for the first 24 hours, and at 50 percent of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing.	1e. Complies.
2. Acceptable assumptions for atmospheric diffusion and dose conversion are:	
a. The 0-8 hour ground level release concentrations may be reduced by a factor ranging from one to a maximum of three (see Figure 1) for additional dispersion produced by the turbulent wake of the reactor building in calculating potential exposures. The volumetric building wake correction, as defined in section 3-3.5.2 of Meteorology and Atomic Energy 1968, should be used only in the 0-8 hour period; it is used with a shape factor of 1/2 and the minimum cross-sectional area of the reactor building only.	2a. Complies. Atmospheric dispersion factors were calculated based on the onsite meteorological measurement programs described in Section 2.3 of the Site Addendum.
b. No correction should be made for depletion of the effluent plume of radioactive iodine due to deposition on the ground, or for the radiological decay of iodine in transit.	2b. Same as 2a above.

TABLE 15.6-7 (Sheet 3)

Regulatory Guide 1.4 Position		Design
c. For the first 8 hours, the breathing rate of persons offsite should be assumed to b Delete 10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.75×10^{-4} cubic meters per second. After that until the end of the accident, the breathing rate should be assumed to be 1.75×10^{-4} cubic meters per second. After that until the end of the accident, the rate should be assumed to be 2.32×10^{-4} cubic meters per second. (These values were developed from the average daily breathing rate [2×10^{7} cm^3/day] assumed in the report of ICRP, Committee II-1959.)	2c.	Complies. See Table 15A-1.
d. The iodine dose conversion factors are given in ICRP Publication 2, Report of Committee II, "Permissible Dose for Internal Radiation," 1959.	2d.	The dose conversion factors provided in Regulatory Guide 1.109 are used. See Table 15A-4.
e. External whole body doses should be calculated using "Infinite Cloud" assumptions, i.e., the dimensions of the cloud are assumed to be large compared to the distance that the gamma rays and beta particles travel. "Such a cloud would be considered an infinite cloud for a receptor at the center because any additional [gamma and] beta emitting material beyond the cloud dimensions would not alter the flux of [gamma rays and] beta particles to the receptor" (Meteorology and Atomic Energy, Section 7.4.1.1-editorial additions made so that gamma and beta emitting material could be considered). Under these conditions the rate of energy absorption per unit volume is equal to the rate of energy released per unit volume. For an infinite uniform cloud containing χ curies of beta radioactivity per cubic meter the beta dose in air at the cloud center is:	2e.	The dose factors given in Regulatory Guide 1.109, for noble gases; for iodine whole body dose factors with 5 cm body tissue attenuation; and for beta-skin dose factors with credit for attenuation in the dead skin layer, are used. See Table 15A-4.

$$ {}_{\beta}D'_{\infty} = 0.457\, E_{\beta}\, \chi $$

TABLE 15.6-7 (Sheet 4)

Regulatory Guide 1.4 Position	Design
The surface body dose rate from beta emitters in the infinite cloud can be approximated as Delete half this amount (i.e., ${}_{\beta}D'_{\infty} = 0.23\,\bar{E}_{\beta}\,\chi$).	
For gamma emitting material the dose rate in air at the cloud center is: $${}_{\gamma}D'_{\infty} = 0.507\,\bar{E}_{\gamma}\,\chi$$	
From a semi-infinite cloud, the gamma dose rate in air is: $${}_{\gamma}D'_{\infty} = 0.25\,\bar{E}_{\gamma}\,\chi$$	
Where:	
${}_{\beta}D'_{\infty}$ = beta dose rate from an infinite cloud (rad/sec)	
${}_{\gamma}D'_{\infty}$ = gamma dose rate from an infinite cloud (rad/sec)	
$\overline{E_{\beta}}$ = average beta energy per disintegration (Mev/dis)	
$\overline{E_{\gamma}}$ = average gamma energy per disintegration (Mev/dis)	
χ = concentration of beta or gamma emitting isotope in the cloud (curie/m^3)	
f. The following specific assumptions are acceptable with respect to the radioactive cloud dose calculations:	2f. See response to 2e.

TABLE 15.6-7 (Sheet 5)

Delete

Regulatory Guide 1.4 Position		Design
(1) The dose at an [illegible] om the reactor should be calculated based on the maximum concentration in the plume at that distance taking into account specific meteorological, topographical, and other characteristics which may affect the maximum plume concentration. These site related characteristics must be evaluated on an individual case basis. In the case of beta radiation, the receptor is assumed to be exposed to an infinite cloud at the maximum ground level concentration at that distance from the reactor. In the case of gamma radiation, the receptor is assumed to be exposed to only one-half the cloud owing to the presence of the ground. The maximum cloud concentration always should be assumed to be at ground level.		
(2) The appropriate average beta and gamma energies emitted per disintegration, as given in the Table of Isotopes, Sixth Edition, by C. M. Lederer, J. M. Hollander, I. Perlman; University of California, Berkeley; Lawrence Radiation Laboratory; should be used.	2f.2	See response to 2e.
g. The atmospheric diffusion model should be as follows:		

TABLE 15.6-7 (Sheet 6)

Delete

Regulatory Guide 1.4 Position		Design
(1) The basic equation for atmospheric diffusion from a ground level point source is: $\chi/Q = \frac{1}{\pi u \sigma_y \sigma_z}$ Where: χ = the short term average centerline value of the ground level concentration (curie/meter^3); Q = amount of material released (curie/sec); u = windspeed (meter/sec); σ_y = the horizontal standard deviation of the plume (meters) [See Figure V-1, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F.A. Gifford, Jr.]; σ_z = the vertical standard deviation of the plume (meters) [See Figure V-2, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.]	2g.1	Short-term accident atmospheric dispersion factors were calculated based on onsite meteorological measurement programs described in Section 2.3 of the Site Addendum. These factors are for ground level releases and are based on Regulatory Guide 1.XXX methodology and represent the worst of the 5-percent site meteorology and the 0.5-percent worst sector meteorology.
(2) For time periods of greater than 8 hours the plume should be assumed to meander and spread uniformly over a 22.5° sector. The resultant equation is: $\chi/Q = \frac{2.32}{\sigma_z u x}$	2g.2	See response to 2g.1 above.

TABLE 15.6-7 (Sheet 7)

Regulatory Guide 1.4 Position		Design
Where: x = distance from point of release to the receptor; other variables are given in g(1).		
(3) The atmospheric diffusion model[2] for ground level releases is based on the information in the following table.	2g.3	See response to 2g.1 above.

Delete

Time Following Accident	Atmospheric Conditions
0-8 hours	Pasquill Type F, windspeed 1 meter/sec, uniform direction
8-24 hours	Pasquill Type F, windspeed 1 meter/sec, variable direction within a 22.5° sector
1-4 days	(a) 40% Pasquill Type D, windspeed 3 meter/sec (b) 60% Pasquill Type F, windspeed 2 meter/sec (c) Wind direction variable within a 22.5° sector
4-30 days	(a) 33.3% Pasquill Type C, windspeed 3 meter/sec (b) 33.3% Pasquill Type D, windspeed 3 meter/sec (c) 33.3% Pasquill Type F, windspeed 2 meter/sec (d) Wind direction 33.3% frequency in a 22.5° sector

(4) Figures 2A and 2B give the ground level release atmospheric diffusion factors based on the parameters given in g(3).	2g.4	See response to 2g.1 above.

TABLE 15.6-8 RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT-ACCIDENT

		TEDE Total Reported Doses (rem)	TEDE Regulatory Limits (rem)
I.	Exclusion Area Boundary (0.5 - 2.5 hr ~~0-2 hr~~)	5.63	25
	~~Thyroid~~	~~128.4~~	~~300~~
	~~Whole body~~	~~4.75~~	~~25~~
II.	Low Population Zone Outer Boundary (0-30 day)	6.24	25
	~~Thyroid~~	~~132.8~~	~~300~~
	~~Whole body~~	~~1.28~~	~~25~~
III.	Control Room (0-30 day)	4.18*	5
	~~Thyroid~~	~~25.55~~	~~30~~
	~~Whole body~~	~~0.453~~	~~5~~
	~~Beta skin~~	~~7.49~~	~~30~~

*Control Room Dose includes a 0.81 rem TEDE contribution for operator transit dose to/from the Control Room over the duration of the event.

b. All noble gas activity has been removed from the reactor coolant system and transferred to the gas decay tank that is assumed to fail.

c. The maximum content of the waste gas decay tank was conservatively assumed to calculate the isotopic activities for the accumulated radioactivity in the gaseous waste processing system after 40 years' operation and immediately following plant shutdown and degasification of the reactor coolant system.

d. The failure is assumed to occur immediately upon completion of the waste gas transfer, releasing the entire contents of the tank to the radwaste building.

e. The dose is calculated as if the release were from the radwaste building at ground level during the 2-hour period immediately following the accident. No credit for radioactive decay is taken.

15.7.1.5.1.3 Mathematical Models Used in the Analysis

The mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A.

b. The atmospheric dispersion factors used in the analysis were calculated based on the onsite meteorological measurement programs described in Section 2.3 of the Site Addendum.

15A.5.2.3 and 15A.5.2.4,

c. The thyroid inhalation and total-body immersion doses to a receptor at the exclusion area boundary or outer boundary of the low-population zone were analyzed, using the models described in Appendix 15A, Sections ~~15A.2.4 and 15A.2.5,~~ respectively.

15.7.1.5.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

For the purposes of evaluating the radiological consequences due to the postulated waste gas decay tank rupture, the resultant activity is conservatively assumed to be released directly to the environment during the 2-hour period immediately following the occurrence of the accident. This is a considerably higher release rate than that based on the actual building exhaust ventilation rate. Therefore, the results of the analysis are based on the most conservative pathway available.

15.7.1.5.2 Identification of Uncertainties and Conservatisms in the Analysis

The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of a waste gas decay tank rupture result from assumptions were analyzed, using the models described in Appendix 15A, Sections ~~15A.2.4 and 15A.2.5,~~ 15A.5.2.3 and 15A.5.2.4, respectively.

15.7.2.5.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

For the purposes of evaluating the radiological consequences due to the postulated liquid radwaste tank rupture, the resultant activity is conservatively assumed to be rele[illegible] during the 2-hour period immediately following the occurrence of the accident. This is a considerably higher release rate than that based on the actual building exhaust ventilation rate. Therefore, the results of the analysis are based on the most conservative pathway available.

15.7.2.5.2 Identification of Uncertainties and Conservatisms in the Analysis

The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of the liquid radwaste tank rupture result from assumptions made involving the release of the radioactivity from the tanks and the meteorology assumed for the site.

a. It was assumed that the liquid radwaste tank fails when the inventory in the tank is a maximum. This assumption results in the greatest amount of activity available for release to the environment.

b. The contents of the ruptured tank are assumed to be released over a 2-hour period immediately following the accident. If the contents of the tank were assumed to mix uniformly with the volume of air within the radwaste building where the tanks are located, then, using the actual building exhaust ventilation rate, a considerable amount of holdup time would be gained. This reduces the amount of activity released to the environment due to the natural decay. Also, no credit for iodine removal by the radwaste building HVAC charcoal adsorbers is taken.

c. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that meteorological conditions assumed will be present during the course of the accident for any extended period of time.

d. A tank is assumed to have collected liquid waste based on operation at 100-percent power with 1 percent failed fuel for an extended period of time, which is eight times higher than under normal operating conditions.

loaded HI-TRAC VW transfer casks to the ISFSI pad is performed within the bounds of the 10 CFR 72.212 Evaluation Report and the HI-STORM UMAX FSAR.

15.7.4.2 Sequence of Events and Systems Operations

The first step in fuel handling is the safe shutdown of the reactor. After a radiation survey of the containment, the disassembly of the reactor vessel is started. After disassembly is complete, the first fuel handling is started. It is estimated that the earliest time to first fuel transfer after shutdown is 72 hours.

The fuel handling accident is assumed to occur after a fuel assembly has been transferred through the fuel storage pool transfer gate but before it has been placed in its designated location in the fuel storage racks.

15.7.4.3 Core and System Performance

The fuel handling accident in the fuel building does not impact the integrity of the core or its system performance.

15.7.4.4 Barrier Performance

A barrier between the released activity and the environment is the reactor building and the fuel building. Since these buildings are designed seismic Category I, it is safe to assume that during the course of a fuel handling accident their integrity is maintained. This means that the pathway for release of radioactivity for a postulated accident in the fuel building is initially via the auxiliary/fuel building normal exhaust system. After it is isolated on a high radiation signal, the release pathway is via the ESF emergency filtration system. For a postulated accident in the reactor building, the release consists of the total amount of radioactivity which could potentially be released. The fuel storage pool and the refueling pool provide minimum decontamination factors ~~of 100~~ for iodine as discussed in Section 15.7.4.5.1.2.

15.7.4.5 Radiological Consequences

15.7.4.5.1 Method of Analysis

15.7.4.5.1.1 Physical Model

The possibility of a fuel-handling accident is remote because of the many administrative controls and physical limitations imposed on the fuel-handling operations (refer to Section 9.1.4). All refueling operations are conducted in accordance with prescribed procedures.

When transferring irradiated fuel from the core to the fuel storage pool for storage, the reactor cavity and refueling pool are filled with borated water at a boron concentration equal to that in the fuel storage pool, which ensures subcritical conditions in the core even if all rod cluster control (RCC) assemblies were withdrawn. After the reactor head

15.7.4.5.1.2 Assumptions and Conditions

The major assumptions and parameters assumed in the analysis are itemized in Tables 15.7-7 and 15A-1.

In the evaluation of the fuel-handling accident, all the fission product release assumptions of Regulatory Guide ~~1.25~~ 1.183 have been followed. Table 15.7-2 provides a comparison of the design to the requirements of Regulatory Guide ~~1.25~~ 1.183. The following assumptions, related to the release of fission product gases from the damaged fuel assembly, were used in the analyses:

a. The dropped fuel assembly is assumed to be the assembly containing the peak fission product inventory. All the fuel rods contained in the dropped assembly are assumed to be damaged. In addition, for the analyses for the accident in the reactor building the dropped assembly is assumed to damage 20 percent of the rods of an additional assembly.

b. The assembly fission product inventories are based on a radial peaking factor of 1.65.

c. The accident occurs 72 hours after shutdown, which is the earliest time fuel-handling operations can begin. Radioactive decay of the fission product inventories was taken into account during this time period.

d. Only that fraction of the fission products which migrates from the fuel matrix to the gap and plenum regions during normal operation was assumed to be available for immediate release to the water following clad damage.

e. The gap activity released to the fuel pool from the damaged fuel rods Insert 15.7.4.5.1.2.e ~~consists of 10 percent of the total noble gases other than Kr-85, 30 percent of the Kr-85, and 10 percent of the total radioactive iodine contained in the fuel rods at the time of the accident.~~

f. The pool decontamination factor is 1.0 for noble gases.

g. Insert 15.7.4.5.1.2.g ~~The effective pool decontamination factor is 100 for iodine.~~

h. Insert 15.7.4.5.1.2.h ~~The iodine above the fuel pool is assumed to be composed of 75 percent inorganic and 25 percent organic species.~~

i. The activity which escapes from the pool is assumed to be available for release to the environment in a time period of 2 hours.

j. No credit for decay or depletion during transit to the site boundary and outer boundary of the low-population zone is assumed.

from the FHB

k. No credit is taken for mixing or holdup in the fuel building atmosphere. The filter efficiency for the ESF emergency filtration system ~~is~~ assumed to be ~~90~~ 0.0 percent for all species of iodine.

120

l. The fuel building is switched from the auxiliary/fuel building normal exhaust system to the ESF emergency exhaust system within ~~90~~ 120 seconds from the time the activity reaches the exhaust duct. The activity released before completion of the switchover is assumed to be discharged directly to the environment with no credit for filtration or dilution. ~~Even if fuel building ventilation isolation does not occur automatically, the calculated doses will be less than those reported in Table 15.7-8 for the bounding case, inside the reactor building.~~ Response time testing is required per Technical Specification 3.3.8 for the fuel building ventilation isolation function.

m. For the inside the reactor building case, no credit has been taken for the mixing or holdup of the radioactivity in the reactor building atmosphere. ~~It is assumed that no containment coolers or hydrogen mixing fans are operating.~~

n. All gap activity assumed available for release is assumed to be released over two hours.

15.7.4.5.1.3 Mathematical Models Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A, Section 15A.2.

and control room

b. The atmospheric dispersion factors are calculated, based on the onsite meteorological measurements programs described in Section 2.3 of the Site Addendum, and are provided in Table 15A-2.

,

c. The ~~thyroid inhalation and total-body immersion~~ doses to a receptor located at the exclusion area boundary, ~~and~~ outer boundary of the low population zone and control room are described in Appendix 15A, ~~Sections 15A.2.4 and 15A.2.5, respectively~~ Sections 15A.2, 15A.3 and 15A.4.

15.7.4.5.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

For evaluating the radiological consequences due to the postulated fuel-handling accident, the resultant activity is conservatively assumed to be released to the environment during the 0-2-hour period immediately following the occurrence of the accident. This is a considerably higher release rate than that based on the actual ventilation rate. Therefore, the results of the analysis are based on the most conservative pathway available.

Sections 15A.2, 15A.3 and 15A.4

15.7.4.5.2 Identification of Uncertainties and Conservatisms in Analysis

The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of a fuel-handling accident result from assumptions made involving the amount of fission product gases available for release to the environment and the meteorology present at the site during the course of the accident. The most significant of these assumptions are:

a. It is assumed in the analysis that all the fuel rods in the dropped assembly are damaged. This is a highly conservative assumption since, transferring fuel under strict fuel handling procedures, only under the worst possible circumstances could the dropping of a spent fuel assembly result in damage to all the fuel rods contained in the assembly.

b. The fission product gap inventory in a fuel assembly is dependent on the power rating of the assembly and the temperature of the fuel. It has been conservatively assumed that the core has been operating at 100 percent for the entire burnup period. The gap activities are listed in Table 15A-3.

c. Iodine removal from the released fission product gas takes place as the gas rises to the pool surface through the body of liquid in the spent fuel pool. The extent of iodine removal is defined by RG 1.183 guidance. ~~determined by mass transfer from the gas phase to the surrounding liquid and is controlled by the bubble diameter and contact time of the bubble in the solution. The values used in the analysis result in a release of activity approximately a factor of 5 greater than anticipated.~~ The release of activity from the pool to the containment atmosphere is time-dependent and consequently there would be sufficient time for this activity to mix homogeneously in a significant percent of the containment volume.

d. The ESF emergency filtration system charcoal filters are known to operate with at least a 99-percent efficiency. This means a further reduction in the iodine concentrations and thus a reduction in the thyroid doses at the exclusion area boundary and the outer boundary of the low-population zone.

e. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that meteorological conditions assumed will be present during the course of the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meteorological conditions assumed, are conservative.

15.7.4.5.2.1 Filter Loadings

The ESF filtration systems which function to limit the consequences of a fuel-handling accident in the fuel building are the ESF emergency filtration system and the control room filtration system.

The activity loadings on the control room charcoal adsorbers as a function of time have been evaluated for the loss-of-coolant accident, Section 15.6.5. Since these filters are capable of accommodating the design basis LOCA fission product iodine loadings, more than adequate design margin is available with respect to postulated fuel-handling accident releases.

The activity loadings on the ESF filtration system charcoal adsorbers have been evaluated in accordance with Regulatory Guide 1.52, which limits the maximum loading to 2.5 mg of iodine per gram of activated charcoal.

15.7.4.5.2.2 Doses to Receptor at the Exclusion Area Boundary, ~~and~~ Low-Population Zone Outer Boundary and Control Room

The potential radiological consequences resulting from the occurrence of a postulated fuel-handling accident occurring in the fuel building and in the reactor building have been conservatively analyzed, using assumptions and models described in previous sections. The TEDE doses ~~total-body dose due to immersion from direct radiation and the thyroid dose due to inhalation~~ have been analyzed for the worst 2 hour ~~0-2-hour~~ dose at the exclusion area boundary, ~~and~~ for the duration of the accident ~~(0 to 2 hours)~~ at the low-population zone outer boundary and in the control room for 30 days. The results are listed in Table 15.7-8. The resultant doses are ~~well~~ within the guideline values of 10 CFR ~~100~~ 50.67 and Regulatory Guide 1.183.

f

Insert 15.7.5

Delete Table

TABLE 15.7-2 DESIGN COMPARISON TO THE REGULATORY POSITIONS OF REGULATORY GUIDE 1.25 "ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT IN THE FUEL HANDLING AND STORAGE FACILITY FOR BOILING AND PRESSURIZED WATER REACTORS" REVISION 0, DATED MARCH 23, 1972

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
1. The assumptions[1] related to the release of radioactive material from the fuel and fuel storage facility as a result of a fuel handling accident are:		
a. The accident occurs at a time after shutdown identified in the technical specifications as the earliest time fuel handling operations may begin. Radioactive decay of the fission product inventory during the interval between shutdown and commencement of fuel handling operations is taken into consideration.	Complies, except the time after shutdown is identified in Section 16.9.5. Accident occurs 72 hours after shutdown.	Complies, except the time after shutdown is identified in Section 16.9.5. Accident occurs 72 hours after shutdown
b. The maximum fuel rod pressurization[2] is 1200 psig.	Calculations performed as directed by footnote 2 indicate that the assumed pool water decontamination factor is valid for internal pressures up to 1500 psig.	Calculations performed as directed by footnote 2 indicate that the assumed pool water decontamination factor is valid for internal pressures up to 1500 psig.
c. The minimum water depth[2] between the top of the damaged fuel rods and the fuel pool surface is 23 feet.	Complies. Water depth is greater than 23 feet. The release point is assumed to be at the top of the fuel pool storage racks.	Complies. Water depth is greater than 23 feet. The release point is assumed to be at the top of the reactor vessel flange.
d. All of the gap activity in the damaged rods is released and consists of 10% of the total noble gases other than Kr-85, 30% of the Kr-85, and 10% of the total radioactive iodine in the rods at the time of the accident. For the purpose of sizing filters for the fuel handling accident addressed in this guide, 30% of the I-127 and I-129 inventory is assumed to be released from the damaged rods.	Complies.	Complies.
e. The values assumed for individual fission product inventories are calculated assuming full power operation at the end of core life immediately preceding shutdown and such calculation should include an appropriate radial peaking factor. The minimum acceptable radial peaking factors are 1.5 for BWR's and 1.65 for PWR's.	Complies. A peaking factor of 1.65 is used.	Complies. A peaking factor is 1.65 is used.
f. The iodine gap inventory is composed of inorganic species (99.75%) and organic species (.25%).	Complies.	Complies.

Delete table

TABLE 15.7-2 (Sheet 2)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
g. The pool decontamination factors for the inorganic and organic species are 133 and 1, respectively, giving an overall effective decontamination factor of 100 (i.e., 99% of the total iodine released from the damaged rods is retained by the pool water). This difference in decontamination factors for inorganic and organic iodine species results in the iodine above the fuel pool being composed of 75% inorganic and 25% organic species.	Complies.	Complies.
h. The retention of noble gases in the pool is negligible (i.e., decontamination factor of 1).	Complies. A decontamination factor of 1 is used.	Complies. A decontamination factor of 1 is used.
i. The radioactive material that escapes from the pool to the building is released from the building[3] over a 2-hour time period.	Complies. A 0-2 hour release from the pool to the building to the environment is assumed.	Complies. A 0-2 hour release from the pool to the building to the environment is assumed.
j. If it can be shown that the building atmosphere is exhausted through adsorbers designed to remove iodine, the removal efficiency is 90% for inorganic species and 70% for organic species.[4]	Not applicable; complies with Regulatory Guide 1.52 as described in Table 9.4-2.	No credit is taken for the normal purge filters.
k. The effluent from the filter system passes directly to the emergency exhaust system without mixing[5] in the surrounding building atmosphere and is then released (as an elevated plume for those facilities with stacks[6]).	Complies.	Complies.
2. The assumptions for atmospheric diffusion are: a. Ground Level Releases (1) The basic equation for atmospheric diffusion from a ground level point source is: $\chi/Q = \frac{1}{\pi u \sigma_y \sigma_z}$ Where: χ = the short term average centerline value of the ground level concentration (curies/m^3)	Short-term atmospheric dispersion factors corresponding to ground level release and accident conditions were based on meteorological measurement programs described in Section 2.3 of the Site Addendum. The dispersion factors are in compliance with the methodology described in Regulatory Guide 1.XXX (see Site Addendum Section 2.3.4.2.1) and represent the worst of the 5 percent overall site meteorology and the 0.5 percent worst sector meteorology.	

TABLE 15.7-2 (Sheet 3)

Delete table

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
Q = amount of material released (curies/sec)		
u = windspeed (meters/sec)		
σ_y = the horizontal standard deviation of the plume (meters) [See Figure V-1, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.]		
σ_z = the vertical standard deviation of the plume (meters) [See Figure V-2, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.]		
(2) For ground level releases, atmospheric diffusion factors[7] used in evaluating the radiological consequences of the accident addressed in this guide are based on the following assumptions:		
(a) windspeed of 1 meter/sec;		
(b) uniform wind direction;		
(c) Pasquill diffusion category F.		
(3) Figure 1 is a plot of atmospheric diffusion factors (χ/Q) versus distance derived by use of the equation for a ground level release given in regulatory position 2.a.(1) and under the meteorological conditions given in regulatory position 2.a.(2).		

TABLE 15.7-2 (Sheet 4)

Delete table

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
(4) Atmospheric diffusion factors for ground level releases may be reduced by a factor ranging from one to a maximum of three (see Figure 2) for additional dispersion produced by the turbulent wake of the reactor building. The volumetric building wake correction as defined in Subdivision 3-3.5.2 of Meteorology and Atomic Energy-1968, is used with a shape factor of 1/2 and the minimum cross-sectional area of the reactor building only.		
b. Elevated Releases		
(1) The basic equation for atmospheric diffusion from an elevated release is: $$\chi/Q = \frac{e^{-h^2/2\sigma_z^2}}{\pi u \sigma_y \sigma_z}$$ Where: χ = the short term average centerline value of the ground level concentration (curies/m^3) Q = amount of material released (curies/sec) u = windspeed (meters/sec) σ_y = the horizontal standard deviation of the plume (meters) [See Figure V-1, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.] σ_z = the vertical standard deviation of the plume (meters) [See Figure V-2, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.]	Not applicable. Ground level releases were assumed.	Not applicable. Ground level releases were assumed.

TABLE 15.7-2 (Sheet 5)

Delete table

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
h = effective height of release (meters)		
(2) For elevated release; atmospheric diffusion factors[7] used in evaluating the radiological consequences of the accident addressed in this guide are based on the following assumptions:		
(a) windspeed of 1 meter/sec;		
(b) uniform wind direction;		
(c) envelope of Pasquill diffusion categories for various release heights;		
(d) a fumigation condition exists at the time of the accident.[8]		
(3) Figure 3 is a plot of atmospheric diffusion factors versus distance for an elevated release assuming no fumigation, and Figure 4 is for an elevated release with fumigation.		
(4) Elevated releases are considered to be at a height equal to no more than the actual stack height. Certain site conditions may exist, such as surrounding elevated topography or nearby structures, which will have the effect of reducing the effective stack height. The degree of stack height reduction will be evaluated on an individual case basis.		
3. The following assumptions and equations may be used to obtain conservative approximations of thyroid dose from the inhalation of radioiodine and external whole body dose from radioactive clouds:		
a. The assumptions relative to inhalation thyroid dose approximations are:	Complies. See Appendix 15A, Section 15A.2.4.	Complies. See Appendix 15A, Section 15A.2.4.
(1) The receptor is located at a point on or beyond the site boundary where the maximum ground level concentration is expected to occur.		

TABLE 15.7-2 (Sheet 6)

Delete table

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
(2) No correction is made for depletion of the effluent plume of radioiodine due to deposition on the ground, or for the radiological decay of radioiodine in transit.		
(3) Inhalation thyroid doses may be approximated by use of the following equation:		

$$D = \frac{F_g \, IFPBR(\chi/Q)}{(DF_p)(DF_f)}$$

Where:

D = thyroid dose (rads)

F_g = fraction of fuel rod iodine inventory in fuel rod void space (0.1)

I = core iodine inventory at time of accident (curies)

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

P = fuel peaking factor

B = Breathing rate = 3.47 x 10^{-4} cubic meters per second (i.e., 10 cubic meters per 8 hour work day as recommended by the ICRP)

DF_p = effective iodine decontamination factor for pool water

DF_f = effective iodine decontamination factor for filters (if present)

χ/Q = atmospheric diffusion factor at receptor location (sec/mμ)

TABLE 15.7-2 (Sheet 7)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
R = adult thyroid dose conversion factor for the iodine isotope of interest (rads per curie). Dose conversion factors for Iodine 131-135 are listed in Table I.[9] These values were derived from "standard man" parameters recommended in ICRP Publication 2.[10]		
TABLE 1 Adult Inhalation Thyroid Dose Conversion Factors Iodine Isotope — Conversion Factor (R) (Rads/curie inhaled) 131 — 1.48×10^6 132 — 5.35×10^4 133 — 4.0×10^5 134 — 2.5×10^4 135 — 1.24×10^5	Table 1; the thyroid dose conversion factors given in ICRP-30 are used. See Table 15A-4.	Table 1; the thyroid dose conversion factors given in ICRP-30 are used. See Table 15A-4.
b. The assumptions relative to external whole body dose approximations are:	Complies. See Appendix 15A, Section 15A.2.5.	Complies. See Appendix 15A, Section 15A.2.5.
(1) The receptor is located at a point on or beyond the site boundary where the maximum ground level concentration is expected to occur.		
(2) External whole body doses are calculated using "Infinite Cloud" assumptions, i.e., the dimensions of the cloud are assumed to be large compared to the distance that the gamma rays and beta particles travel. The dose at any distance from the reactor is calculated based on the maximum ground level concentration at that distance. For an infinite uniform cloud containing χ curies of beta radioactivity per cubic meter, the beta dose rate in air at the cloud center is:[11]	(2) See Table 15A-4 for whole body dose conversion factors from Federal Guidance Report 12.	

Delete table

TABLE 15.7-2 (Sheet 8)

Delete table

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
$_{\beta}D'_{\infty} = 0.457\ \bar{E}_{\beta}\chi$ Where: $_{\beta}D'_{\infty}$ = beta dose rate from an infinite cloud (rad/sec) $\bar{E}_{\beta}$ = average beta energy per disintegration (Mev/dis) χ = concentration of beta or gamma emitting isotope in the cloud (curie/m^3) Because of the limited range of beta particles in tissue, the surface body dose rate from beta emitters in the infinite cloud can be approximated as being one-half this amount or: $_{\beta}D'_{\infty} = 0.23\ E_{\beta}\chi$ For gamma emitting material the dose rate in tissue at the cloud center is: $_{\gamma}D'_{\infty} = 0.507\ \bar{E}_{\gamma}\chi$ Where: $_{\gamma}D'_{\infty}$ = gamma dose rate from an infinite cloud (rad/sec)		

TABLE 15.7-2 (Sheet 9)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
$\bar{E}_\gamma$ = average gamma energy per disintegration (MEV/dis) However, because of the presence of the ground, the receptor is assumed to be exposed to only one-half of the cloud (semi-infinite) and the equation becomes: $_\gamma D' = 0.25 \bar{E}_\gamma \chi$ Thus, the total beta or gamma dose to an individual located at the center of the cloud path may be approximated as: $_\beta D_\infty = 0.23 \bar{E}_\beta \psi$ or $_\gamma D = 0.25 E_\gamma \psi$ Where ψ is the concentration time integral for the cloud (curie sec/m^3) (3) The beta and gamma energies emitted per disintegration, as given in Table of Isotopes,[12] are averaged and used according to the methods described in ICRP Publication 2.	Delete table	

Notes:

1. The assumptions given are valid only for oxide fuels of the types currently in use and in cases where the following conditions are not exceeded:

   a. Peak linear power density of 20.5 kW/ft for the highest power assembly discharged.

   b. Maximum center-line operating fuel temperature less than 4500°F for this assembly.

TABLE 15.7-2 (Sheet 10)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
c. Average burnup for the peak assembly of 25,000 MWD/ton or less (this corresponds to a peak local burnup of about 45,000 MWD/ton).	Gap fractions of 10% remain valid for fuel assemblies up to 33,000 MWD/MTU. Beyond this burnup, a 12% gap fraction will be used.	Gap fractions of 10% remain valid for fuel assemblies up to 33,000 MWD/MTU. Beyond this burnup, a 12% gap fraction will be used.

Delete table

2. For release pressures greater than 1200 psig and water depths less than 23 feet, the iodine decontamination factors will be less than those assumed in this guide and must be calculated on an individual case basis using assumptions comparable to conservatism to those of this guide.

3. The effectiveness of features provided to reduce the amount of radioactive material available for release to the environment will be evaluated on an individual case basis.

4. These efficiencies are based upon a 2-inch charcoal bed depth with 1/4 second residence time. Efficiencies may be different for other systems and must be calculated on an individual case basis.

5. Credit for mixing will be allowed in some cases; the amount of credit will be evaluated on an individual case basis.

6. Credit for an elevated release will be given only if the point of release is (a) more than two and one-half times the height of any structure close enough to affect the dispersion of the plume or (b) located far enough from any structure which could affect the dispersion of the plume. For those plants without stacks the atmospheric diffusion factors assuming ground level release given in regulatory position 2.b. should be used.

7. These diffusion factors should be used until adequate site meteorological data are obtained. In some cases, available information on such site conditions as meteorology, topography and geographical location may dictate the use of more restrictive parameters to ensure a conservative estimate of potential offsite exposures.

8. For sites located more than 2 miles from large bodies of water such as oceans or one of the Great Lakes, a fumigation condition is assumed to exist at the time of the accident and continue for 1/2 hour. For sites located less than 2 miles from large bodies of water a fumigation condition is assumed to exist at the time of accident and continue for the duration of the release (2 hours).

TABLE 15.7-2 (Sheet 11)

Delete table

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)

9. Dose conversion factors taken from "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, J. J. DiNunno, R. E. Baker, F. D. Anderson, and R. L. Waterfield (1962).

10. Recommendations of the International Commission on Radiological Protection, "Report of Committee II on Permissible Dose for Internal Radiation (1959)," ICRP Publication 2, (New York: Permagon Press, 1960).

11. Meteorology and Atomic Energy-1968, Chapter 7.

12. C. M. Lederer, J. M. Hollander, and I. Perlman, Table of Isotopes, Sixth Edition (New York: John Wiley and Sons, Inc. 1967).

TABLE 15.7-7 PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A FUEL-HANDLING ACCIDENT

		In Fuel Building	In Reactor Building
I.	Source Data		
	a. Core power level, MWt	3,636	3,636
	b. Radial peaking factor	1.65	1.65
	c. Decay time, hours	72	72
	d. Number of fuel assemblies affected	1.0	1.2
	e. Fraction of fission product gases contained in the gap region of the fuel assembly	Per R.G. ~~1.25~~ 1.183	Per R.G. ~~1.25~~ 1.183
II.	Atmospheric Dispersion Factors	See Table 15A-2	See Table 15A-2
III.	Activity Release Data		
	a. Percent of affected fuel assemblies gap activity released to pool	100	100
	b. Pool decontamination factors		
	1. Iodine (effective)	~~100~~ 200	~~100~~ 200
	2. Noble gas	1	1
	c. Filter efficiency, percent	0 until isolation ~~90 thereafter~~ 0 assumed after Isolation	0
	d. Building mixing volumes assumed, percent of total volume	0	0
	e. HVAC exhaust rate, cfm	~~20,000 until isolation~~ ~~9,000 thereafter~~ Activity completely released over 2 hours	Activity completely released over 2 hours.
	f. ~~Building isolation~~ Control room isolation time,	~~90~~ 120 sec	~~2 hours~~ 120 sec
	g. Activity release period, hrs	2	2

TABLE 15.7-8 RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT

	Doses ~~(rem)~~ (rem TEDE)
In Fuel Building	
Exclusion Area Boundary (0-2 hr)	7.8E-01
~~Thyroid~~	~~6.40~~
~~Whole-body~~	~~0.235~~
Low Population Zone Outer Boundary (duration)	2.7E-01
~~Th~~ Control Room	~~0.640~~ 6.5E-01
~~Whole-body~~	~~0.0235~~
In Reactor Building	
Exclusion Area Boundary (0-2 hr)	9.7E-01
~~Thyroid~~	~~61.7~~
~~Whole-body~~	~~0.359~~
Low Population Zone Outer Boundary (duration)	3.4E-01
~~Thyroid~~	~~6.17~~
~~Whole-body~~	~~0.0359~~
Control Room	1.58

. The higher value between the 95th percentile overall site concentration factors for a 90° direction window and the maximum sector concentration factors are used for the off-site doses (EAB and LPZ), and the 95th percentile concentration factors for a 90° direction window are used for the on-site doses (CR and TSC). These dispersion factors are given in Table 15A-2 (see Section 2.3.4 and the Site

EVALUATION MODELS AND PARAMETERS

15A.1 GENERAL ACCIDENT PARAMETERS

1.145 (1983)

This section contains the parameters used in analyzing the radiological consequences of postulated accidents. Table 15A-1 contains the general parameters used in all the accident analyses. For par and vent stack to particular accidents, refer to that accident parameter section. The site specific, ground-level release, short-term dispersion factors ~~(for accidents, ground-level releases are assumed) are based on Regulatory Guide 1.XXX (1978)~~ (Ref. 1) methodology ~~and the 0.5 percent worst sector meteorology and these are given in Table 15A-2 (see Section 2.3.4 and the Site~~ Addendum for additional details on meteorology). The core and gap inventories are given in Table 15A-3. The ~~thyroid (via inhalation pathway), beta skin, and total body (via submersion pathway)~~ TEDE dose factors based on References ~~2 and 3~~ 8 and 9 are given in Table 15A-4.

15A.2 OFFSITE RADIOLOGICAL CONSEQUENCES CALCULATIONAL MODELS

This section presents the models and equations used for calculating the integrated activity released to the environment, the accident flow paths, and the equations for dose calculations. ~~Two major release models are considered: (1) a single holdup system with no internal cleanup and (2) a holdup system wherein a two-region spray model is used for internal cleanup.~~

via the A-E pathway

equipment (pathway C-D-E)

15A.2.1 ACCIDENT RELEASE PATHWAYS

The release pathways for the major accidents are given in Figure 15A-1. The accidents and their pathways are as follows:

LOCA: Immediately following a postulated loss-of-coolant accident (LOCA), the release of radioactivity from the containment is to the environment with the containment spray and ESF systems in full operation. The release in this case is calculated using equation (1) which takes into account a two-region spray model within the containment. The release of radioactivity to the environment due to assumed ESF system leakages in the auxiliary building will be via ESF ~~filters and is calculated using equation (5)~~, using a factor of 0.01 to account for the combined effect of the airborne fraction of radioiodine and the ESF filter efficiency. The total removal ~~constant, λ_1,~~ for this release pathway includes decay ~~(λ_{1d})~~ and release ~~(λ_{1l})~~ ~~removal constants~~ associated with holdup and mixing in the sumps (no holdup or mixing assumed in the auxiliary building)~~; however, no internal removal constant (λ_r) is assumed~~. The release of radioactivity to the environment due to assumed leakage from the RWST ~~is~~ calculated in equation (1)

via the R-E pathway

Section 15A.2.2

from ~~Section 15A.2.2.b.~~ The release of radioactivity to the environment due to the assumed operation of the containment mini-purge system for the first few seconds after a LOCA is calculated ~~using equation (5) from Section 15A.2.2.a,~~ with no credit for filtration, plateout, or valve stroke time.

E

WGDTR: The activity release to the environment due to waste gas decay tank rupture with no holdup. The release pathway is A'-~~D~~. The [...]efore assumed to be the initial source activity

and includes consideration of pH and iodine evolution in conformance with NUREG/CR-5950.

B-E

FHA: The release to the environment due to a fuel handling accident (FHA) in the fuel building is via vents. The release pathway is ~~B-C-D~~. Since the release is calculated without any credit for holdup in the fuel building, the total release will be the product of the initial activity and the filter nonremoval efficiency fraction (for noble gases, the nonremoval efficiency fraction is 1). The release of radioactivity to the environment due to FHA inside the containment is direct and unfiltered, via the ~~A-D~~ pathway without any credit for holdup (see Figure 15A-1). The release is calculated assuming the total gap inventor[...] hour period, reduced only by the pool deconta[...]

include a partition factor of 0.01 for non-noble gases, and 1 for noble gases.

A-E

RE

~~CAE~~: Radioactivity release to the environment due to the control rod ejection (~~CAE~~) accident is direct and unfiltered. The releases from the primary system are calculated using equation 1 which considers holdup in the single-region primary system (the spray removal is not assumed); the secondary (steam) releases via the relief valves ~~are calculated without any holdup~~. The pathways for these releases are ~~A-D~~ and A'-E.

RE

A-E

MSLB, SGTR: Radioactivity releases to the environment due to main steam line break (MSLB) or steam generator tube rupture (SGTR) accidents are direct and unfiltered ~~with no holdup~~ via the A'-E pathway. The activity release calculations for these accidents are complex, involving spiking effects, time-dependent flashing fractions, and scrubbing of flashed activities; the release calculations are described in those sections that address these accidents.

Insert 15A.2.2

As used in the radiological consequence evaluations, partition factor refers to the fraction of the total release that is airborne.

~~15A.2.2.a SINGLE - REGION RELEASE MODEL~~

~~It is assumed that any activity released to the holdup system instantaneously diffuses to uniformly occupy the system volume.~~

~~The following equations are used to calculate the integrated activity released from postulated accidents.~~

~~$A_1(0)$ = initial source activity at time t_o, Ci~~

$A_1(t)$ = source activity at time t seconds, Ci

$$A_1(t) = A_1(0)e^{-\lambda_1 t} \quad (1)$$

Delete

where λ_1 = total removal constant from primary holdup system, sec^{-1}

$$\lambda_1 = \lambda_d + \lambda_{1\ell} + \lambda_r \quad (2)$$

where λ_d = decay removal constant, sec^{-1}

$\lambda_{1\ell}$ = primary holdup leak or release rate, sec^{-1}

λ_r = internal removal constant (i.e., sprays, plateout, etc.), sec^{-1}

Thus, the direct release rate to the atmosphere from the primary holdup system

$$R_u(t) = \lambda_{1\ell}A_1(t)$$

$R_u(t)$ = unfiltered release rate (Ci/sec) (3)

The integrated activity release is the integral of the above equation.

$$IAR(t) = \int_o^t R_u(t) = \int_o^t \lambda_{1\ell}A_1(o)e^{-\lambda_1 t} \quad (4)$$

This yields:

$$IAR(t) = (\lambda_{1\ell}A_1(o)/\lambda_1)(1 - e^{-\lambda_1 t}) \quad (5)$$

15A.2.2.b TWO - REGION RELEASE MODEL FOR DOSES DUE TO LEAKAGE FROM THE ECCS SUMPS TO THE RWST

It is assumed that the activity released to the holdup system (in this case, the containment recirculation sumps) instantaneously diffuses to uniformly occupy the sump volume. Removal mechanisms from the sumps include decay and release (i.e., leakage) to the RWST. Expanding upon the equations developed in Section 15A.2.2.a above, the release rate from the RWST to the environment is given by

$$R_2(t) = 0.1\,\lambda_{2\ell}A_2(t) \quad (1a)$$

where $R_2(t)$ = the unfiltered release rate from the RWST vent, Ci/sec

0.1 = assumed percent of ~~radioiodine~~ released to the RWST that becomes airborne

Delete

$\lambda_{2\ell}$ = release rate constant for leakage from the RWST to the environment, based on an assumed 3 gpm leak rate from the sumps that is uniformly mixed in the RWST volume, hr^{-1}

$A_2(t)$ = RWST activity, Ci

The activity time rate of change for the RWST is given by

$$\frac{dA_2(t)}{dt} = \lambda_{1\ell}A_1(t) - \lambda_2 A_2(t) \qquad (2a)$$

where $\lambda_{1\ell}$ = release rate constant for leakage from the uniformly mixed sumps to the RWST, based on an assumed 3 gpm leak rate from the sumps to the RWST, hr^{-1}

λ_2 = $\lambda_d + \lambda_{2\ell}$

$A_1(t)$ = containment sump activity, Ci

Using equation (1) from Section 15A.2.2.a above,

$$\frac{dA_2(t)}{dt} + \lambda_2 A_2(t) = \lambda_{1\ell}A_1(0)e^{-\lambda_1 t} \qquad (3a)$$

where $A_1(0)$ = initial sump activity, assumed to be 50% of the initial iodine core inventory

λ_1 = $\lambda_d + \lambda_{1\ell}$

λ_d = decay removal constant for the sumps, hr^{-1}

The solution of this equation is given by

$$A_2(t) = \frac{A_1(0)\lambda_{1\ell}}{(\lambda_2 - \lambda_1)}[e^{-\lambda_1 t} - e^{-\lambda_2 t}] + A_2(0)e^{-\lambda_2 t} \qquad (4a)$$

Therefore, the integrated release from the RWST is given by

$$IAR_2(t) = \int_o^t R_2(t)dt$$

$$IAR_2(t) = 0.1\lambda_{2\ell}\int_o^t A_2(t)dt$$

Delete

$$IAR_2(t) = \frac{0.1A_1(0)\lambda_{1\ell}\lambda_{2\ell}}{\lambda_2\lambda_1(\lambda_2-\lambda_1)}[\lambda_2(1 - e^{-\lambda_1 t}) + \lambda_1(e^{-\lambda_2 t} - 1)] + \left[\frac{0.1A_2(0)\lambda_{2\ell}}{\lambda_2}\right](1 - e^{-\lambda_2 t}) \quad (5a)$$

15A.2.3 TWO - REGION SPRAY MODEL IN CONTAINMENT (LOCA)

A two-region spray model is used to calculate the integrated activity released to the environment. The model consists of a sprayed and unsprayed region in containment and a constant mixing rate between them.

As it is assumed that there are no sources after initial release of the fission products, the remaining processes are removal and transfer so that the multivolume containment is described by a system of coupled first-order differential equations of the form

$$\frac{dA_i}{dt} = -\sum_{j=1}^{K_i} \lambda_{ij}A_i - \sum_{\ell=1}^{n-1} Q_{i\ell}\frac{A_i}{V_i} + \sum_{\ell=1}^{n-1} Q_{i\ell}\frac{A_i}{V_i} \quad (6)$$

where

A_i	=	fission product activity in volume i, Ci
n	=	number of volumes considered in the model
$Q_{i\ell}$	=	transfer rate from volume i to volume ℓ, cc/sec
V_i	=	volume of the ith compartment, cc
λ_{ij}	=	removal rate of the jth removal process in volume i, sec^{-1}
K_i	=	total number of removal processes in volume i

This system of equations is readily solved if the coefficients are known.

For a two-region model, the above system reduces to

$$\frac{dA_1}{dt} = -\sum_{j=1}^{K_1} -\lambda_{1j}A_1 - Q_{12}\frac{A_1}{V_1} + Q_{21}\frac{A_2}{V_2} \qquad (6a)$$

$\boxed{\text{Delete}}$

$$\frac{dA_2}{dt} = -\sum_{j=1}^{K_2} -\lambda_{2j}A_1 - Q_{21}\frac{A_2}{V_2} + Q_{12}\frac{A_1}{V_1} \qquad (6b)$$

To solve the two preceding equations, use the method of Laplace transforms.

use $L\frac{df(t)}{dt} = SF(s) - f(0)$

where f(t) is any regular function of t,

L is the symbol of L-transform

S is the Laplace variable

and F(s) is the Laplace transform of f(t).

The solutions to the equations are:

$$A_1(t) = \frac{A_1(0)}{S_1 - S_2}(S_1 e^{S_2 t} - S_2 e^{S_1 t}) + \frac{A_2(0)\frac{Q_{21}}{V_2} - A_1(0)\left(\sum_{j=1}^{K_1}\lambda_{1j} + \frac{Q_{12}}{V_1}\right)}{S_1 - S_2}(e^{S_1 t} - e^{S_2 t}) \qquad (7)$$

and

$$A_2(t) = \frac{A_2(0)}{S_1 - S_2}(S_1 e^{S_2 t} - S_2 e^{S_1 t}) + \frac{A_1(0)\frac{Q_{12}}{V_1} - A_2(0)\left(\sum_{j=1}^{K_2}\lambda_{2i} + \frac{Q_{21}}{V_2}\right)}{S_1 - S_2}(e^{S_1 t} - e^{S_2 t}) \qquad (8)$$

where, $S_1, S_2 = -1/2\left[\sum_{j=1}^{K_1}\lambda_{1j} + \sum_{j=1}^{K_2}\lambda_{2j} + \frac{Q_{12}}{V_1} + \frac{Q_{21}}{V_2}\right]$

Delete

$$\pm 1/2\left[\left(\sum_{j=1}^{K_1}\lambda_{1j}+\sum_{j=1}^{K_2}\lambda_{2j}+\frac{Q_{12}}{V_1}+\frac{Q_{21}}{V_2}\right)^2 - 4\left(\sum_{j=1}^{K_1}\lambda_{1j}\sum_{j=1}^{K_2}\lambda_{2j}+\frac{Q_{12}}{V_1}\sum_{j=1}^{K_2}\lambda_{2j}+\frac{Q_{12}}{V_1}\sum_{j=1}^{K_1}\lambda_{1j}\right)\right]^{1/2} \quad (8)$$

At time $t_1 > t_0$, according to equations (7) and (8):

$$A_1(t_1) = \frac{A_2(t_0)\frac{Q_{21}}{V_2} - A_1(t_0)\left(\sum_{j=1}^{K_1}\lambda_{1j} + \frac{Q_{12}}{V_1}\right)}{S_1 - S_2}\left(e^{S_1(t_1-t_0)} - e^{S_2(t_1-t_0)}\right)$$

$$+\frac{A_1(t_0)}{S_1 - S_2}\left(S_1 e^{S_2(t_1-t_0)} - S_2 e^{S_1(t_1-t_0)}\right) \quad (10a)$$

$$A_2(t_1) = \frac{A_1(t_0)\frac{Q_{12}}{V_1} - A_2(t_0)\left(\sum_{j=1}^{K_2}\lambda_{2j} + \frac{Q_{21}}{V_2}\right)}{S_1 - S_2}\left(e^{S_1(t_1-t_0)} - e^{S_2(t_1-t_0)}\right)$$

$$+\frac{A_2(t_0)}{S_1 - S_2}\left(S_1 e^{S_2(t_1-t_0)} - S_2 e^{S_1(t_1-t_0)}\right) \quad (10b)$$

Where S_1, S_2 are given in equation (9).

$A(t_1)$ = Activity in the primary containment at any time, t_1

= $A_1(t_1) + A_2(t_1)$

= $$\frac{A_1(t_0)\sum_{j=1}^{K_1}\lambda_{1j} + A_2(t_0)\sum_{j=1}^{K_2}\lambda_{2j}}{S_1 - S_2}\left(e^{S_2(t_1-t_0)} - e^{S_1(t_1-t_0)}\right)$$

$$\frac{A_1(t_0)+A_2(t_0)}{S_1 - S_2}\left(S_1 e^{S_2(t_1-t_0)} - S_2 e^{S_1(t_1-t_0)}\right) \qquad (11)$$

To calculate the integrated activity released to the atmosphere we must first calculate the release rate of activity. This is found from

$$R(t) = \lambda_{1\ell} A(t) \qquad (11a)$$

Delete

The integrated activity released from time t_0 - t_1 is then

$$IAR = \int_{t_0}^{t} R(t)dt$$

This solves as

$$IAR = \lambda_{1\ell}\left[m_2\left(\frac{e^{S_2(t_1-t_0)}-1}{S_2} - \frac{e^{S_1(t_1-t_0)}-1}{S_1}\right) + m_3\left(\frac{S_2}{S_1}\left(e^{S_1(t_1-t_0)}-1\right)\right)\right] \qquad (12)$$

where (13a)

$$m_2 = \frac{A_1(t_0)\sum_{j=1}^{K_1}\lambda_{1j} + A_2(t_0)\sum_{j=--1}^{K_2}\lambda_{2i}}{S_1 - S_2}$$

and (13b)

$$m_3 = \frac{A_1(t_0) + A_2(t_0)}{S_1 - S_2}$$

15A.2.4 OFFSITE THYROID DOSE CALCULATION MODEL

Offsite thyroid doses are calculated using the equation:

$$D_{TH} = \sum_i DCF_{Thi} \quad \sum (IAR)_{ij\,j} \quad (BR)_j \quad (\chi/Q)_j \qquad (14)$$

where

$(IAR)_{ij}$ = integrated activity of isotope i released* during the time interval j in Ci and $(BR)_j$ = breathing rate during the time interval j in meter^3/second

$(\chi/Q)_j$ = offsite atmospheric dispersion factor during time interval j in second/meter^3

$(DCF)_{Thi}$ = thyroid dose conversion factor via inhalation for isotope i in rem/Ci

Delete

D_{Th} = thyroid dose via inhalation in rems

* No credit is taken for cloud depletion by ground deposition and radioactive decay during transport to the exclusion area boundary or the outer boundary of the low-population zone.

15A.2.5 OFFSITE TOTAL-BODY DOSE CALCULATIONAL MODEL

Assuming a semi-infinite cloud of gamma emitters, offsite total-body doses are calculated using the equation:

$$D_{TB} = \sum_i DCF_{\gamma i} \sum_j (IAR)_{ij} (\chi / Q)_j$$

where

$(IAR)_{ij}$ = integrated activity of isotope i released* during the jth time interval in Ci

and $(\chi/Q)_j$ = offsite atmospheric dispersion factor during time interval j in second/meter^3

$(DCF)_{\lambda i}$ = total-body gamma dose conversion factor for the ith isotope in rem-meter^3/Ci-sec

D_{TB} = total-body dose in rems

15A.3 CONTROL ROOM RADIOLOGICAL CONSEQUENCES CALCULATIONAL MODELS

Only radiation doses to a control room operator due to postulated LOCA are presented in this chapter since a study of the radiological consequences in the control room due to various postulated accidents indicate that the LOCA is the limiting case.

Unfiltered air may also leak into the equipment room and from the equipment room into the control building and control room filtration inlet plenum.¶
Experience gained from the development and performance of inleakage measurement testing using the atmospheric tracer depletion test methodology led to the identification of three interacting zones for modeling CREVS operation and the analysis of control room dose. The model applies the alternate source term established per Amendment ___ of the Callaway Operating License. Accordingly, the dose analysis model includes a three-zone model for which the atmospheric tracer depletion test method explicitly determines inleakage values for the control room envelope (CRE), control building envelope (CBE) and equipment room envelope (ERE) in accordance with the requirements of the Technical Specifications.¶

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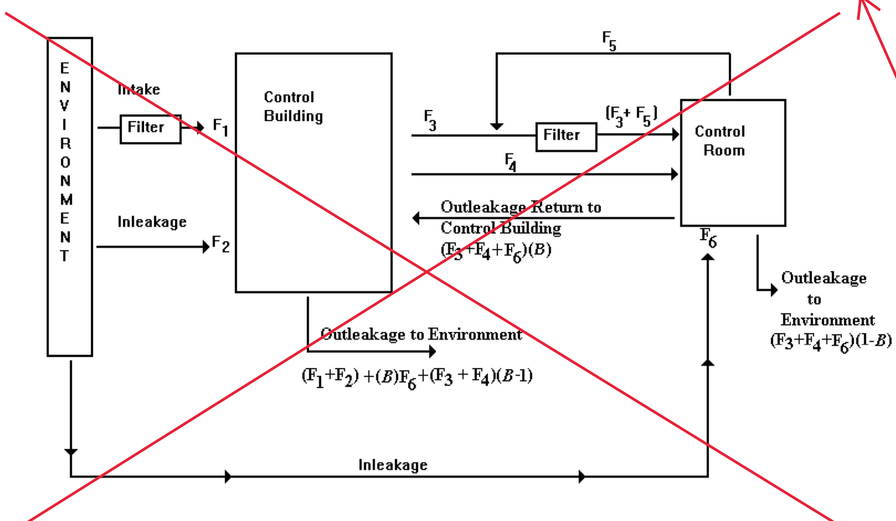
each time interval are found by multiplying the activity release to the environment by the appropriate χ/Q for that time interval. The flow path model is shown ~~below~~ in Figure 15A-3.



Once activity is brought into the control building, mixing within the control building is afforded by the control room pressurization fan. ~~Only one-half of the control building volume is considered as the mixing volume.~~ The control room filtration system fan takes air from the control building and the control room (recirculation) and discharges to the control room through the control room filtration safety grade filters.

The control room ventilation isolation signal (CRVIS) starts both trains of the control room pressurization system and the control room filtration system. For the determination of doses to control room personnel, the worst single failure has been ascertained to be the failure of the filtration fan in one of the two filtration system trains. Operator action is required to isolate the train with the failed filtration fan. At the same time, one train of the control room pressurization system will also be isolated. ~~Prior to isolation, a potential pathway exists allowing air from the control building to enter the control room, bypassing the control room pressurization fan and one control room filtration fan operate for the duration of the accident. No bypass pathways then exist for unfiltered air to enter the control room.~~ This failure allows control building air to bypass the control room filtration filter and enter the control room.

~~Owning~~ Owing to this single failure of the control room filtration fan, the assumed failure of one of the two containment spray (CS) trains, and two of the four hydrogen mixing subsystem fans, inherent in the LOCA analysis parameters given in Table 15.6-6 should not be applied in this analysis. With both trains of CS and four hydrogen mixing fans operating,

In accordance with Section 5.1.3 of Regulatory Guide 1.183, a conservative combination of cases may be used instead of comparing the results of multiple individual cases with alternative single failures.

mixing between the new ... iving much greater iodine ... ses to ~~control room~~ ... given in Table 15.6-6.

The activity in the control building and control room is calculated ~~by solving the following coupled set of first order differential equations~~.

$$\frac{dA_{CB}(t)}{dt} = [(1-\eta)F_1 + F_2]\chi/Q[R(t)] + \beta\lambda_{4\ell}A_{CR}(t) - \lambda_3 A_{CB}(t)$$

$$\frac{dA_{CR}(t)}{dt} = [(1-\eta)\lambda_{3f} + \lambda_{3u}]A_{CB}(t) - \lambda_4 A_{CR}(t) + F_6[R(t)]\chi/Q$$

where $A_{CB}(t)$ = activity in control building at time t, curies

$A_{CR}(t)$ = activity in control room at time t, curies

by the RADTRAD-NAI code which uses Equation 1 In Section 15A.2.2 for all compartments for the duration of the event. The total effective dose equivalent (TEDE) is then determined and reported by RADTRAD-NAI for the full 30 day event.

the

F_1 = filtered intake rate, meter^3/sec

F_2 = unfiltered Control Building intake (inleakage), meter^3, /sec

F_6 = unfiltered Control Room in leakage, meter^3/sec

χ/Q = atmospheric dispersion factor, sec/meter^3

$R(t)$ = activity release rate in Ci/sec as given in Equation 3 of Section 15A.2.2.a, Equation 1a of Section 15A.2.2.b, or Equations 11 and 11a of Section 15A.2.3

λ_3 = $\lambda_d + \lambda_{3\ell} + \lambda_{3f} + \lambda_{3u}$, total removal rate from the control building, sec^{-1}

λ_d = isotopic decay constant, sec^{-1}

$\lambda_{3\ell}$ = outleakage to atmosphere from the control building (=((F$_1$ + F$_2$) - (β - 1)(F$_3$ + F$_4$)+βF$_6$)/V$_{CB}$ with V$_{CB}$ being control building mixing volume in meter^3), sec^{-1}

λ_{3f} = filtered flow from control building into control room ($=F_3/V_{CB}$, F_3 in meter^3/sec), sec^{-1}

λ_{3u} = unfiltered flow from control building into control room ($=F_4/V_{CB}$, F_4 in me Delete ec^{-1}

λ_4 = $\lambda_d + \lambda_r + \lambda_{4\ell}$, total removal rate from the control room, sec^{-1}

λ_r = recirculation removal rate ($=\eta F_5/V_{CR}$ with F_5 being recirculation flow rate in meter^3/sec through filter with efficiency η and V_{CR} being control room volume in meter^3), sec^{-1}

$\lambda_{4\ell}$ = leakage to all destinations from the control room ($= [F_3 + F_4 + F_6]/V_{CR}$), sec^{-1}

Upon solving this coupled set of differential equations, the integrated activity in the control room (IA_{CR}) is determined by the expression

$$IA_{CR}(t) = \int_o^t A_{CR}(t)dt$$

This $IA_{CR(t)}$ is used to calculate the doses to the operator in the control room. This activity is multiplied by an occupancy factor which accounts for the time fraction the operator is in the control room.

15A.3.2 CONTROL ROOM THYROID DOSE CALCULATIONAL MODEL

Control room thyroid doses via inhalation pathway are calculated using the following equation:

$$D_{TH-CR} = \frac{BR}{V_{CR}} \sum_i DCF_{Thi} \sum_j (IA_{CRij}) \times O_j$$

where

$D_{Th\text{-}CR}$ = control room thyroid dose in rem

and

BR	=	breathing rate assumed to be always 3.47 x 10^{-4} $meter^3$/ second
V_{CR}	=	volume of the control room in cubic meters
DCF_{Thi}	=	Delete thyroid do on factor for adult via inhalation in rem/Ci for isotope i
IA_{CRij}	=	integrated activity in control room in Ci-sec for isotope i during time interval j
o_j	=	control room occupancy fraction during time interval j

15A.3.3 CONTROL ROOM BETA-SKIN DOSE CALCULATIONAL MODEL

The beta-skin doses to a control room operator are calculated using the following equation:

$$D_{\beta - CR} = \frac{1}{V_{CR}} \sum_i DCF_{\beta i} \sum_j (IA_{CRij}) \times o_j$$

where $D_{\beta\text{-}CR}$ and $DCF_{\beta i}$ are the beta-skin doses in the control room in rem and the beta-skin dose conversion factor for isotope i in rem-$meter^3$/Ci-sec, respectively. The other symbols are explained in Section 15A.3.4.

15A.3.4 CONTROL ROOM TOTAL-BODY DOSE CALCULATION

Due to the finite structure of the control room, the total-body gamma doses to a control room operator will be substantially less than what they would be due to immersion in an infinite cloud of gamma emitters. The finite cloud gamma doses are calculated using Murphy's method (Ref. 4) which models the control room as a hemisphere. The following equation is used:

$$D_{TB - CR} = \frac{1}{V_{CR}^{(GF)}} \sum_i DCF_{\gamma i} \sum_j (IA_{CRij}) \times o_j$$

where

GF	=	dose reduction due to control room geometry factor
GF	=	$1173/(V_1)^{0.338}$

~~V_1 = volume of the control room in cubic feet~~

~~$D_{TB\text{-}CR}$ = total-body dose in the control room in rem,~~

~~and other quantities have been defined in subsections 15A.2.5 and 15A.3.4.~~

~~15A.3.4.1 Model for Radiological Consequences Due to Radioactive Cloud External to the Control Room~~

~~This dose is calculated based on the semi-infinite cloud model which is modified using the protection factors described in Section 7.5.4 of Reference 5 to account for the control room walls.~~

Insert 15A.4,5

15A.6 REFERENCES

1. USNRC Regulatory Guide ~~1.XXX~~ 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, Reissued February 1983 See Appendix 3A ~~and Section 2.3.4.2.1~~ and Section 2.3.4.4 of the Callaway Site Addendum.

2. USNRC Regulatory Guide 1.109, Rev. 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50 Appendix I," October 1977.

3a. Kocher, D.C., "Nuclear Decay Data for Radionuclides Occurring in Routine Releases from Nuclear Fuel Cycle Facilities," ORNL/NUREG/TM-102, August 1977.

3b. Berger, M.J., "Beta-Ray Dose in Tissue-Equivalent Material Immersed in a Radioactive Cloud," Health Physics, Vol. 26, pp. 1-12, January 1974.

4. Murphy, K.G. and Campe, K.M., "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," Paper presented at the 13th AEC Air Cleaning Conference.

5. "Meteorology and Atomic Energy 1968," D. H. Slade (ed.), USAEC Report, TID 24190, 1968.

Insert 15A.6

TABLE 15A-1 PARAMETERS USED IN ACCIDENT ANALYSIS

I.	General		
	1.	Core power level, Mwt	3636 (102% power)
	2.	Number of fuel assemblies in the core	193
	3.	Maximum radial peaking factor	1.65
	4.	Percentage of failed fuel	1.0
	5.	Steam generator tube leak, lb/hr	500
II.	Sources		
	1.	Core inventories, Ci	Table 15A-3
	2.	~~Gap inventories, Ci~~	~~Table 15A-3~~
	2.	Primary coolant specific activities, μCi/gm	Table 15A-5
	4.	Primary coolant activity, technical specification limit for iodines - I-131 dose equivalent, μCi/gm	1.0
	5.	Secondary coolant activity technical specification limit for iodines - I-131 dose equivalent, μCi/gm	0.1
III.	Activity Release Parameters		
	1.	Free volume of containment, ft^3	~~2.5~~ 2.70 x 10^6
	2.	Containment leak rate	
		i. 0-24 hours, % per day	0.2
		ii. after 24 hrs, % per day	0.1
IV.	Control Room Dose Analysis (for LOCA)		
	1.	Control building	
		i. Mixing volume, cf	148,000
		ii. Filtered intake, cfm	
		Prior to operator action (0-30 minutes)	900
		After operator action (30 minutes - 720 hours)	450
		iii. Unfiltered inleakage, cfm	**
		iv. Filter efficiency (~~all forms of iodine~~ particulates), %	95
	2.	Control room	
		i. Volume, cf	48,500
		ii. Filtered flow from control building, cfm	440

TABLE 15A-1 (Sheet 2)

	iii.	Unfiltered flow from control building, cfm	
		Prior to operator action (0-30 minutes)	440
		After operator action (30 minutes - 720 hours)	0
	iv.	Filtered recirculation, cfm	$\boxed{1030}$
	v.	Filter efficiency (all forms of iodine), %	95
	vi.	Unfiltered in leakage, cfm	**
V.	Miscellaneous		
	1.	Atmospheric dispersion factors, χ/Q sec/m^3	Table 15A-2
	2.	Dose conversion factors	
		i. TEDE ~~total body and beta skin~~, rem-meter^3/Ci-sec (Sv-meter^3/Bq-sec)	Table 15A-4
		ii. ~~thyroid,~~ rem/Ci (Sv/Bq)	Table 15A-4
	3.	Breathing rates, meter^3/sec	
		i. control room at all times	3.55 x 10^{-4}
		ii. offsite	
		0-8 hrs	3.5 x 10^{-4}
		8-24 hrs	1.8 x 10^{-4}
		24-720 hrs	2.3 x 10^{-4}
	4.	Control room occupancy fractions	
		0-24 hrs	1.0
		24-96 hrs	0.6
		96-720 hrs	0.4

** See Figure 15A-2 for inleakage values used in the accident analysis.

This Figure is replaced with Insert F15A-2

TABLE 15A-2 LIMITING SHORT-TERM ATMOSPHERIC DISPERSION FACTORS (χ/QS) FOR ACCIDENT ANALYSIS

Replace with Insert T15A-2

Location Type/ Time Interval (hrs)	χ/Q ec/meter^3)
Site boundary	
0-2	1.5E-4
Low-population zone	
0-8	1.5E-5
8-24	1.0E-5
24-96	4.6E-6
96-720	1.5E-6
Control room (via containment leakage)	
0-8	7.2E-4
8-24	5.3E-4
24-96	1.7E-4
96-720	0
Control room (via unit vent exhaust)	
0-8	1.3E-4
8-24	9.0E-5
24-96	4.1E-5
96-720	0

TABLE 15A-3 FUEL AND ROD GAP INVENTORIES - CORE (CI)

Replace entire table with insert T15A-3

	Core	
Isotope		
I-131		E+6
I-132	1.44E+8	1.44E+7
I-133	2.04E+8	2.04E+7
I-134	2.25E+8	2.25E+7
I-135	1.91E+8	1.91E+7
Kr-83m	1.27E+7	1.27E+6
Kr-85m	2.72E+7	2.72E+6
Kr-85	8.61E+5	2.58E+5
Kr-87	5.24E+7	5.24E+6
Kr-88	7.38E+7	7.38E+6
Kr-89	9.03E+7	9.03E+6
Xe-131m	1.12E+6	1.12E+5
Xe-133m	6.35E+6	6.35E+5
Xe-133	1.99E+8	1.99E+7
Xe-135m	3.96E+7	3.96E+6
Xe-135	4.38E+7	4.38E+6
Xe-137	1.78E+8	1.78E+7
Xe-138	1.70E+8	1.70E+7

~~*Gap activity is assumed to be 10 percent of fuel activity for all isotopes except for Kr-85; for Kr-85 it is assumed to be 30 percent of the fuel activity.~~

TABLE 15A-4 DOSE CONVERSION FACTORS USED IN ACCIDENT ANALYSIS

Nuclide	Total Body Rem-meter^3 / Ci-sec	Beta Skin Rem-meter^3 / Ci-sec	Thyroid * Rem/Ci
I-131	8.72E-2	3.17E-2	1.49E+6
I-132	5.13E-1	1.32E-1	1.43E+4
I-133	1.55E-1	7.35E-2	2.69E+5
I-134	5.32E-1	9.23E-2	3.73E+3
I-135	4.21E-1	1.29E-1	5.60E+4
Kr-83m	2.40E-6	0	NA
Kr-85m	3.71E-2	4.63E-2	NA
Kr-85	5.11E-4	4.25E-2	NA
Kr-87	1.88E-1	3.09E-1	NA
Kr-88	4.67E-1	7.52E-2	NA
Kr-89	5.27E-1	3.20E-1	NA
Xe-131m	2.91E-3	1.51E-2	NA
Xe-133m	7.97E-3	3.15E-2	NA
Xe-133	9.33E-3	9.70E-3	NA
Xe-135m	9.91E-2	2.25E-2	NA
Xe-135	5.75E-2	5.90E-2	NA
Xe-137	4.51E-2	3.87E-1	NA
Xe-138	2.80E-1	1.31E-1	NA

* ~~The radiological consequences for the replacement SG program (1) have been re-analyzed using the following thyroid dose conversion factors from ICRP-30 and whole body dose conversion factors from Federal Guidance Report 12 (except that RG 1.109 Table B-1 is used for Kr-89 and Xe-137). These factors may be applied to other accident sequences as they are re-analyzed (e.g., the fuel handling accident cases addressed in Section 15.7.4):~~

Replace with Insert T15A-4

Nuclide	Total Body **REM-meter^3 / Ci-sec	Thyroid Rem/ci
I-131	6.73E-02	1.07E+06
I-132	4.14E-01	6.29E+03
I-133	1.09E-01	1.81E+05
I-134	4.81E-01	1.07E+03
I-135	2.95E-01	3.14E+04
Kr-83m	5.55E-06	NA
Kr-85m	2.77E-02	NA

TABLE 15A-4 (Sheet 2)

Nuclide	Total Body **$\frac{\text{REM-meter}^3}{\text{Ci-sec}}$	Thyroid Rem/ci
Kr-85	4.40E-04	NA
Kr-87	1.52E-01	NA
Kr-88	3.77E-01	NA
Kr-89	5.27E-01	NA
Xe-131m	1.44E-03	NA
Xe-133m	5.07E-03	NA
Xe-133	5.77E-03	NA
Xe-135m	7.55E-02	NA
Xe-135	4.40E-02	NA
Xe-137	4.51E-02	NA
Xe-138	2.14E-01	NA

Delete

**Federal Guidance Report 12 uses units of $\frac{\text{Sv} - \text{meter}^3}{\text{Bq} - \text{sec}}$.

Conversion factors are: 1 Sv = 100 Rem and 1 Bq = 2.7E-11 Ci. The above WB dose conversion factors are equal to those in Federal Guidance Report 12.

(1) FSAR sections re-analyzed for radiological consequences as part of the replacement steam generator program include:

15.1.5 STEAM SYSTEM PIPING FAILURE
15.2.6 LOSS OF NONEMERGENCY AC POWER TO THE STATION AUXILIARIES
15.3.3 REACTOR COOLANT PUMP SHAFT SEIZURE (LOCKED ROTOR)
15.4.8 SPECTRUM OF ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENTS
15.6.2 BREAK IN INSTUMENT LINE OR OTHER LINES FROM REACTOR COOLANT PRESSURE BOUNDARY THAT PENETRATE CONTAINMENT
15.6.3 STEAM GENERATOR TUBE FAILURE

TABLE 15A-5 INITIAL RADIOACTIVITY FOR ACCIDENTS THAT USE THE PRIMARY-TO-SECONDARY LEAKAGE RELEASE PATHWAY

I.	Reactor Coolant System Inventories	
	Isotope	Concentration (µCi/gm)
a.	Iodines [1]	
	I-130	1.75E-02
	I-131	0.793
	I-132	2.2
	I-133	1.12
	I-134	4.0
	I-135	2.2
b.	Noble Gases [2]	
	Kr-83m	2.02E-01
	Kr-85m	1.00E+00
	Kr-85	7.45E-02
	Kr-87	5.86E-01
	Kr-88	1.88E+00
	~~Kr-89~~	~~5.04E-02~~
	Xe-131m	1.77E-01
	Xe-133m	9.64E-01
	Xe-133	4.81E+01
	Xe-135m	1.31E-01
	Xe-135	2.87E+00
	~~Xe-137~~	~~9.06E-02~~
	Xe-138	4.40E-01

Insert T15A-5
(additional pages)

TABLE 15A-5 (Sheet ~~2~~ 4)

II. Boron Recycle Holdup Tank Inventories [3]

	Isotope	Concentration (Ci)
a.	Iodines	
	I-131	4.07
	I-132	0.044
	I-133	0.740
	I-134	3.81E-3
	I-135	0.119
b.	Noble Gases	
	Kr-83m	0.169
	Kr-85m	1.92
	Kr-85	11.53
	Kr-87	0.330
	Kr-88	2.34
	Kr-89	1.18E-03
	Xe-131m	15.91
	Xe-133m	23.26
	Xe-133	2560
	Xe-135m	0.0353
	Xe-135	11.83
	Xe-138	0.0463

TABLE 15A-5 (Sheet ~~3~~) 5

III. Boron Recycle Holdup Tank Inventories [3]

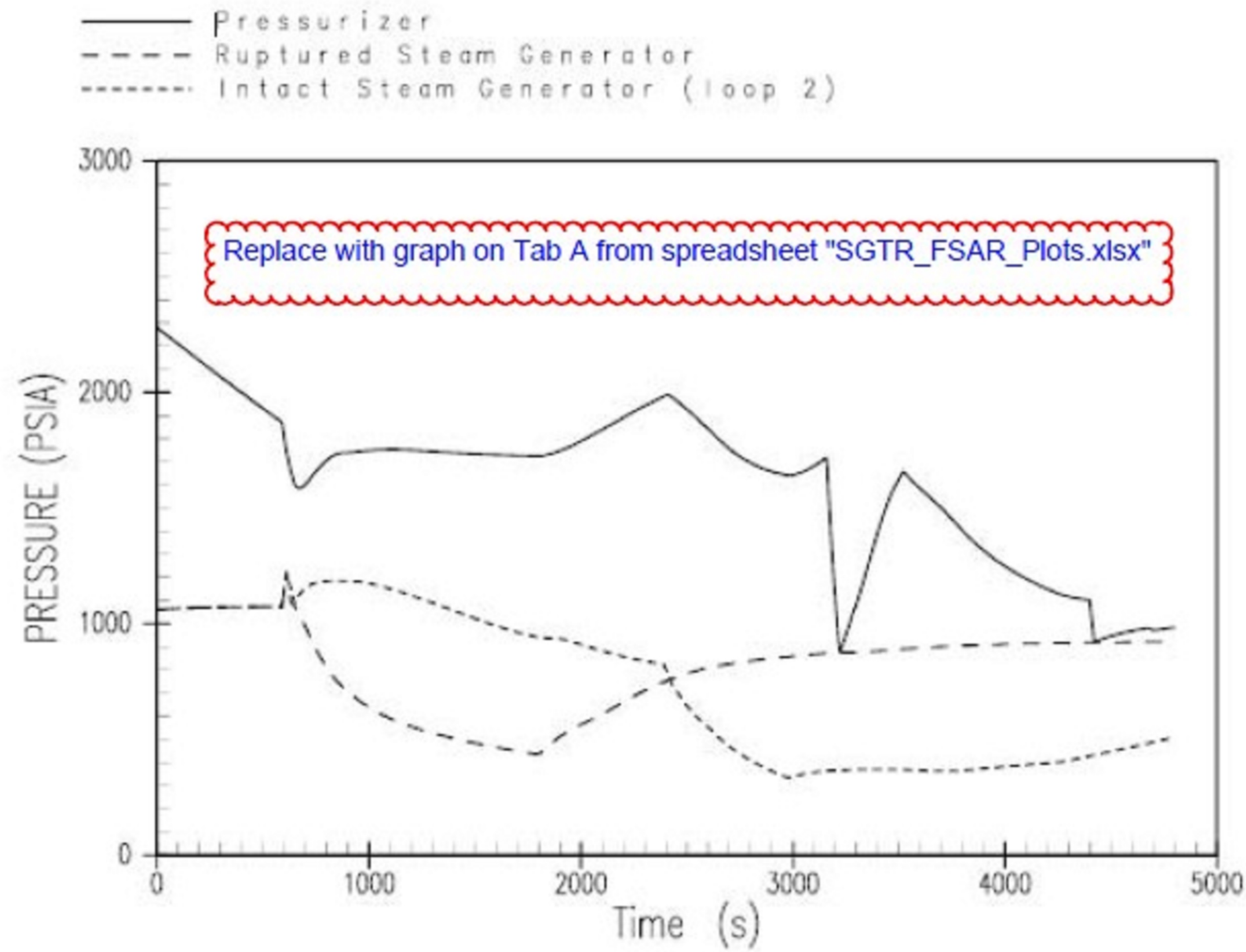
	Isotope	Concentration (Ci)
a.	Iodines	
	I-131	26.4
	I-132	0.533
	I-133	3.40
	I-134	4.60E-4
	I-135	0.377

Notes:

1. The RCS iodine values were obtained by starting with the original Licensing Bases 1% failed fuel projections. Then the shorter-lived iodine isotopic concentrations were increased based on steady-state conditions observed during fuel cycles in which Callaway operated with failed fuel. This isotopic spectrum is intended to bound concentrations that would be encountered with either tight or open fuel defects.
2. The RCS noble gas values were obtained based on the original Licensing Bases 1% failed fuel projections, and then adjusted upwards to account for calorimetric error and capacity factor variations.
3. Radwaste Tank inventories are based on the original Licensing Bases projections and adjusted for capacity factor and plant power uprating.

, Class 3 and Class 6,

Attachment B - SGTR Figures



~~**Note:**~~

~~The thermal-hydraulic model presented in this figure is based on an analysis that credits all three ASDs on the intact steam generat[...]pport the RCS rapid cooldown. This [...]spect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.~~

Delete Note

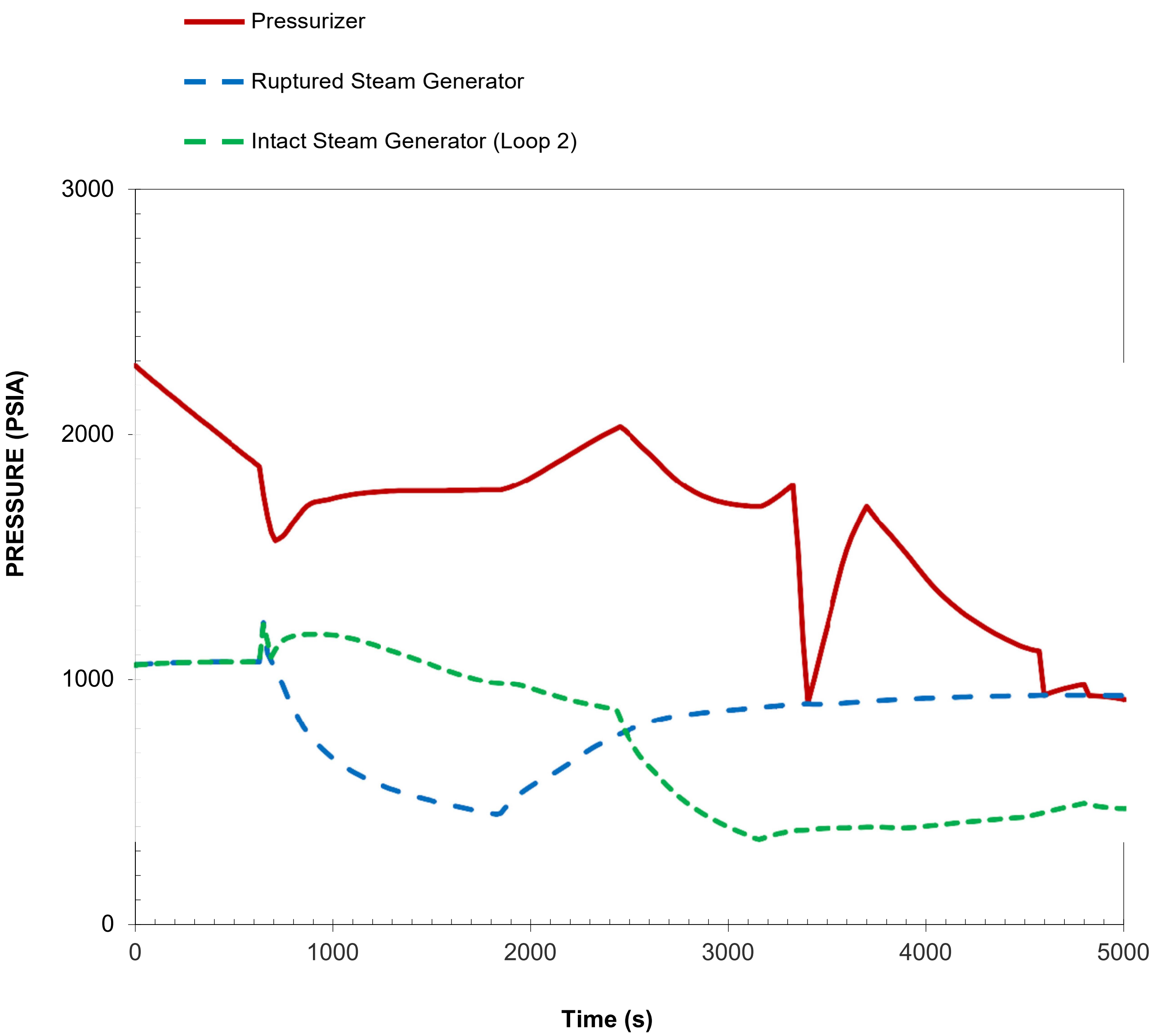
~~Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.~~

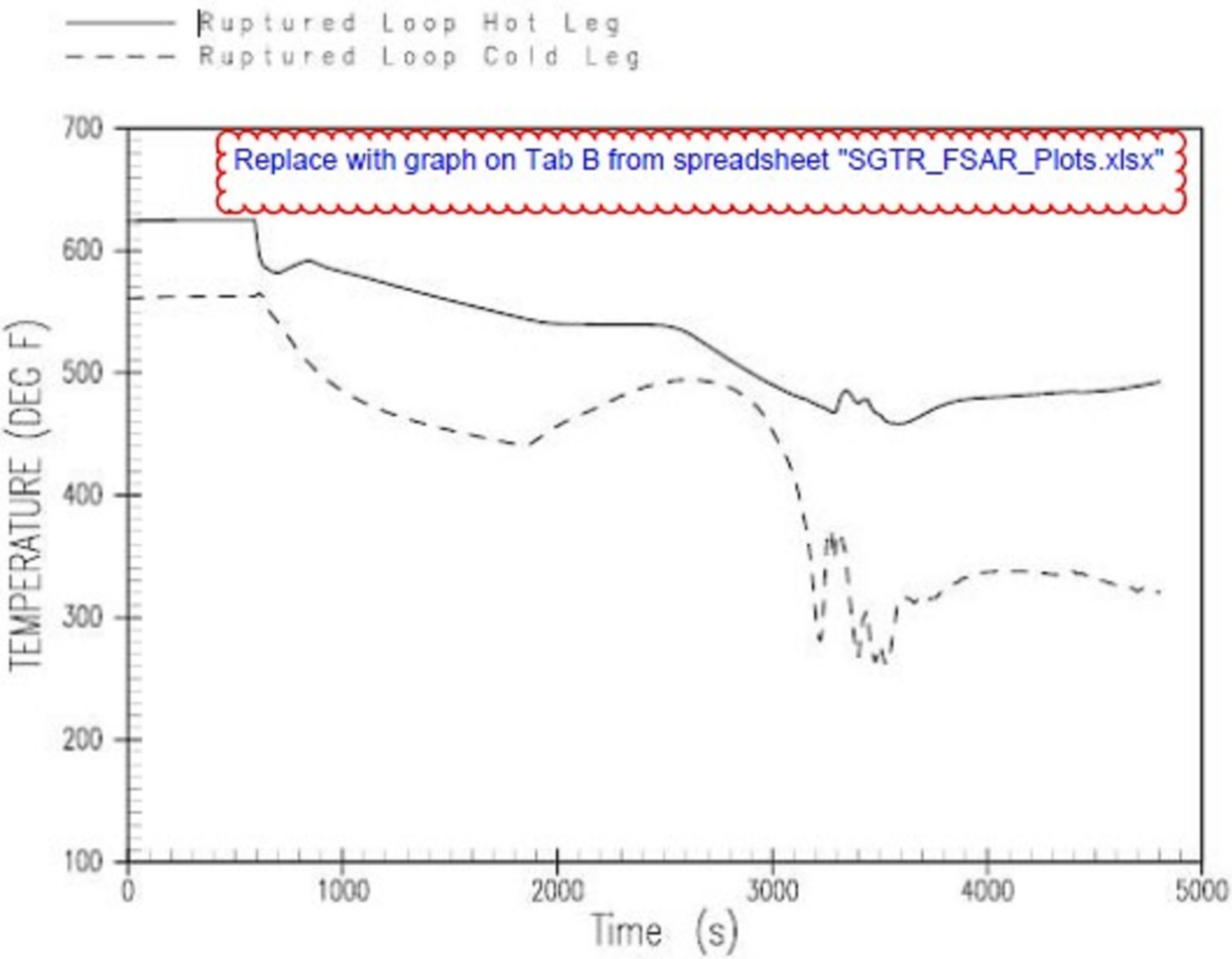
CALLAWAY PLANT

FIGURE 15.6-3A

PRESSURIZER AND STEAM GENERATOR (RUPTURED AND INTACT GENERATORS) PRESSURE TRANSIENTS FOR STEAM GENERATOR TUBE RUPTURE EVENT

REV. 16 6/11


Pressurizer
Ruptured Steam Generator
Intact Steam Generator (Loop 2)
3000
2000
1000
0
PRESSURE (PSIA)
0
1000
2000
3000
4000
5000
Time (s)



Note:

The thermal-hydraulic model presented in this figure is based on an analysis that credits all three ASDs on the intact steam generators to [illegible] Delete Note the RCS rapid cooldown. This [illegible] spect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.

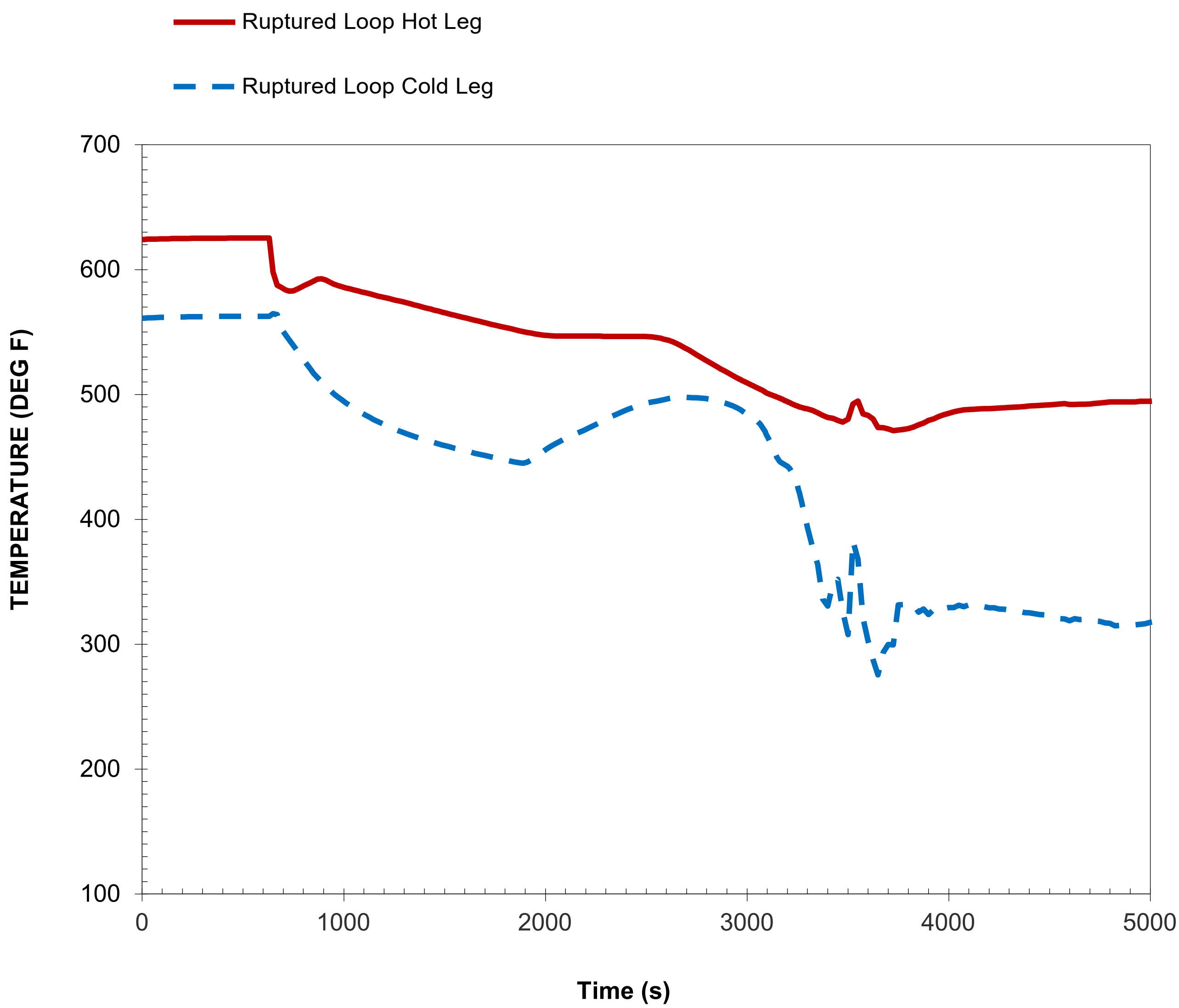
Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.

CALLAWAY PLANT

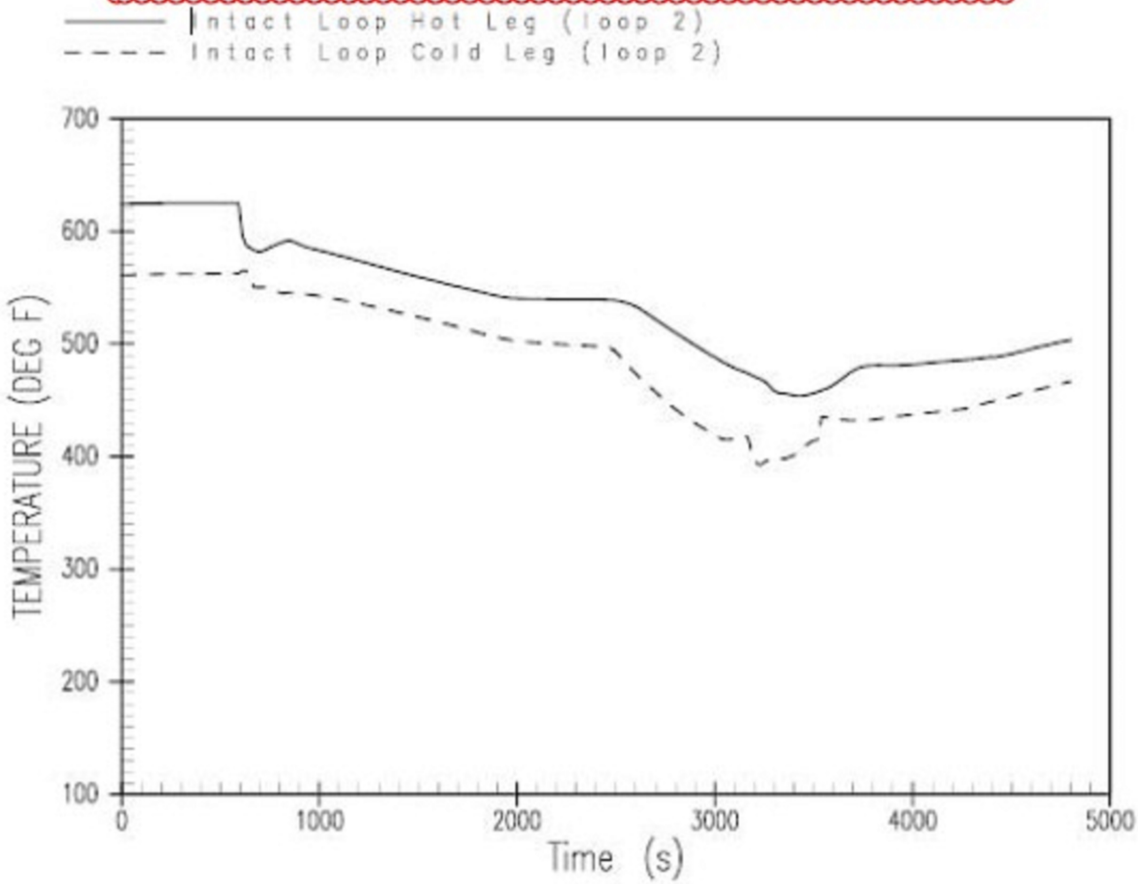
FIGURE 15.6-3B

REACTOR COOLANT SYSTEM TEMPERATURE (RUPTURED LOOP) TRANSIENT FOR STEAM GENERATOR TUBE RUPTURE EVENT

REV. 16 5/11


Ruptured Loop Hot Leg
Ruptured Loop Cold Leg
700
600
500
400
300
200
100
0
1000
2000
3000
4000
5000
TEMPERATURE (DEG F)
Time (s)

Replace with graph on Tab C from spreadsheet "SGTR_FSAR_Plots.xlsx"

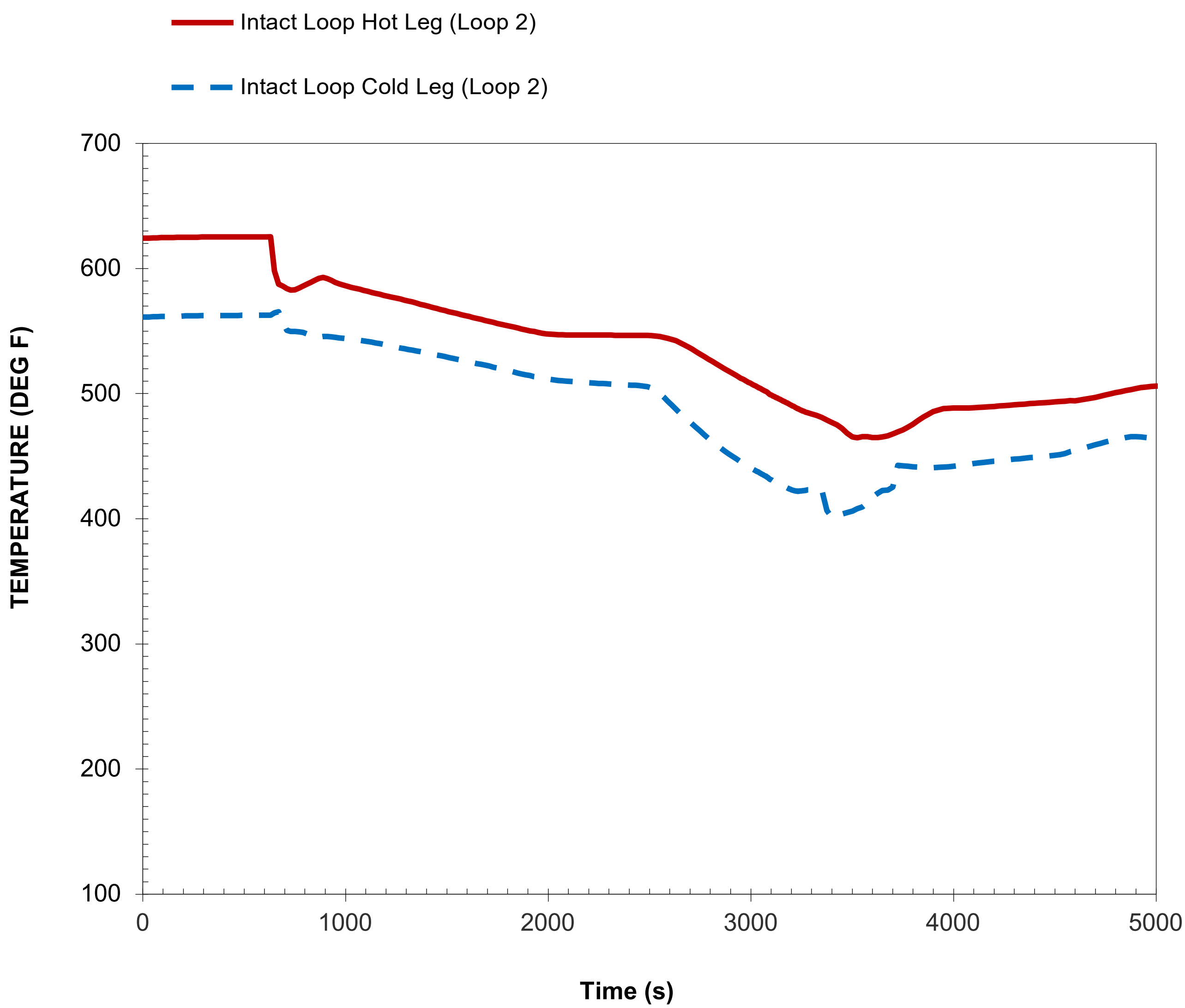


~~Note:~~

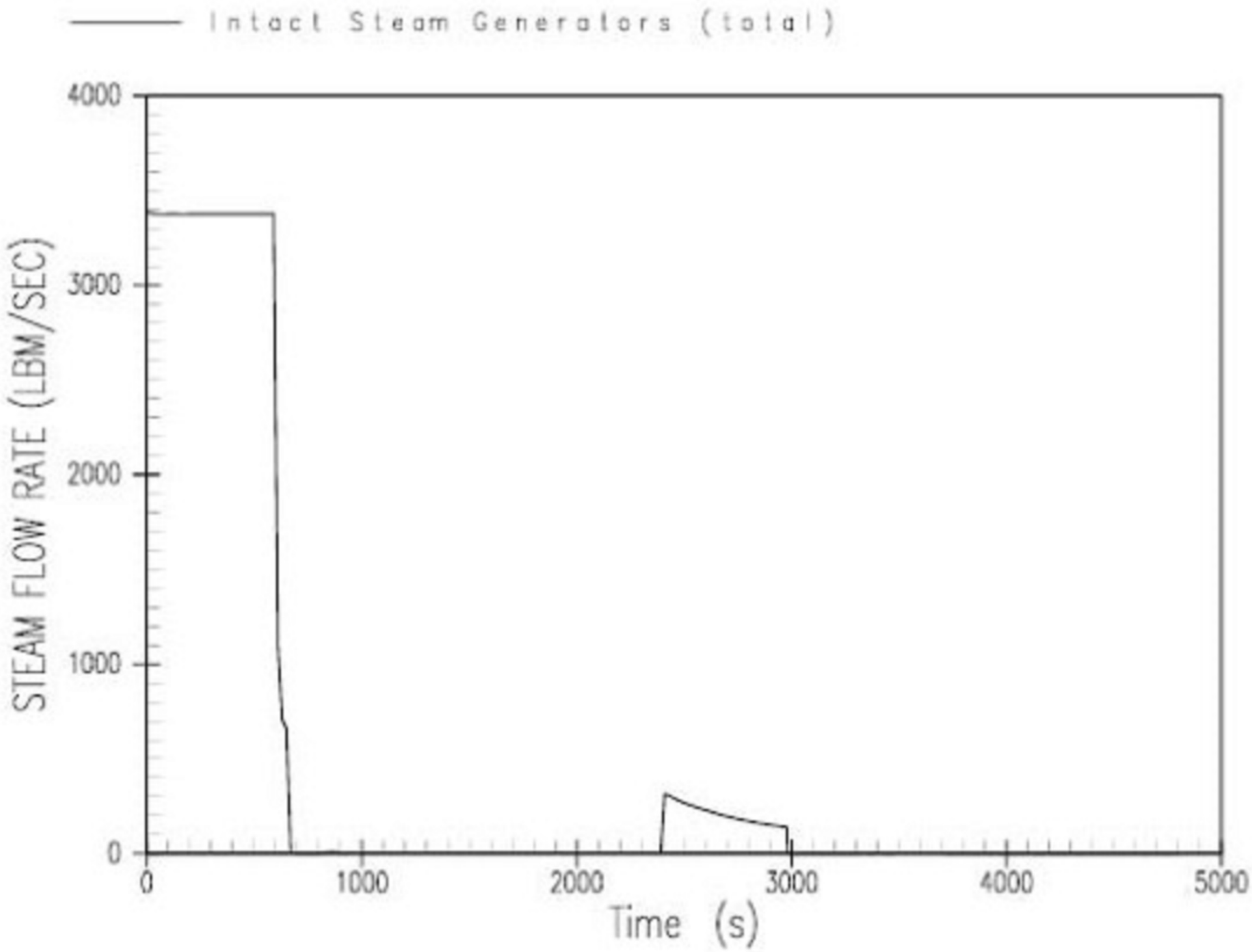
~~The thermal-hydraulic model presented in this figure is based on an analysis~~ Delete Note ~~the intact steam generators to be available to support the RCS rapid cooldown. This is conservative with respect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.~~

~~Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.~~

CALLAWAY PLANT
FIGURE 15.6-3C REACTOR COOLANT SYSTEM TEMPERATURE (INTACT LOOPS) TRANSIENT FOR STEAM GENERATOR TUBE RUPTURE EVENT REV. 16 6/11


Intact Loop Hot Leg (Loop 2)
Intact Loop Cold Leg (Loop 2)
TEMPERATURE (DEG F)
700
600
500
400
300
200
100
0
1000
2000
3000
4000
5000
Time (s)

Replace with graph on Tab D from spreadsheet "SGTR_FSAR_Plots.xlsx"



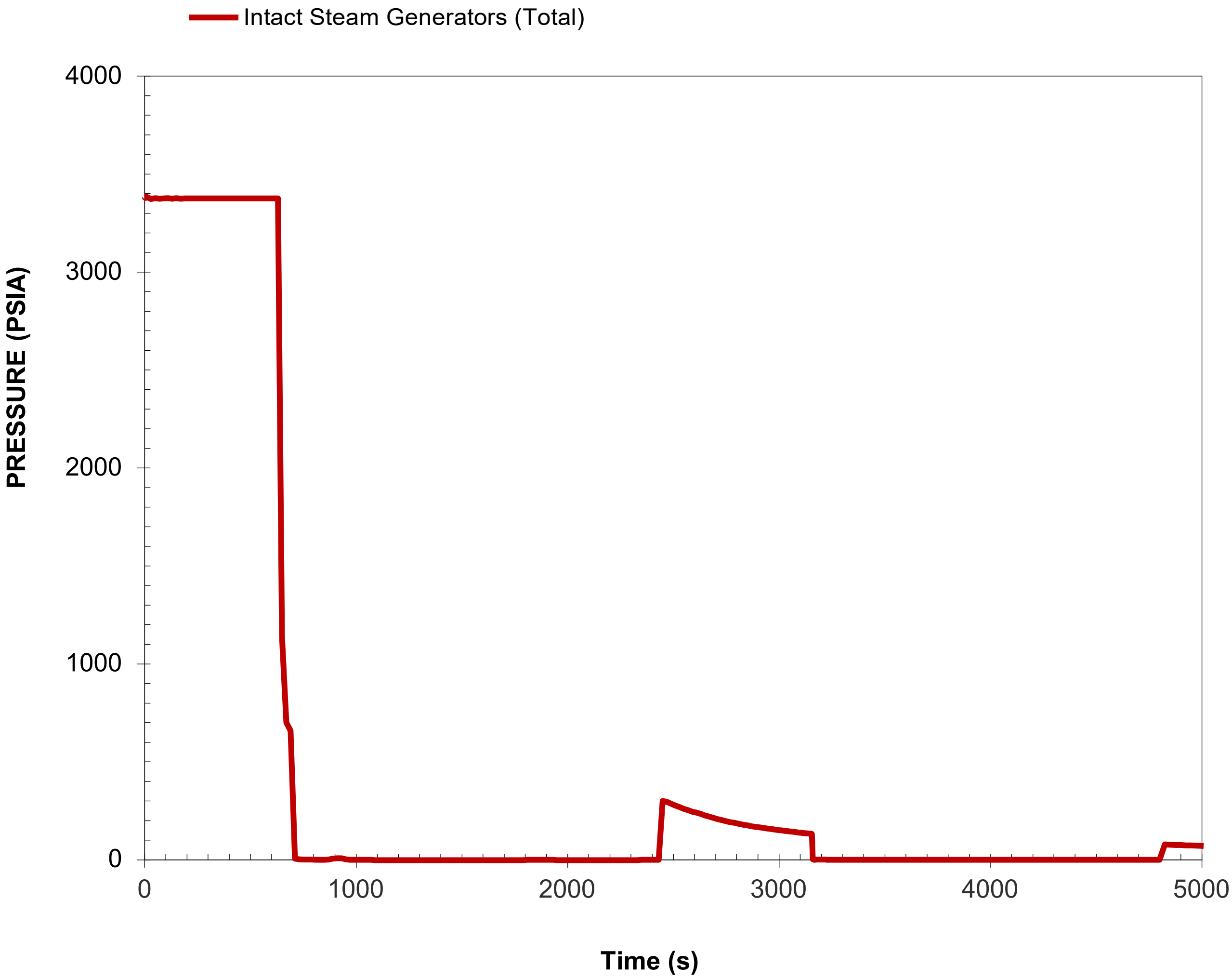
~~Note:~~

~~The thermal-hydraulic model presented in this figure is based on an analysis that c[illegible] the intact steam generators to b[illegible]e RCS rapid cooldown. This is conservative with respect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.~~

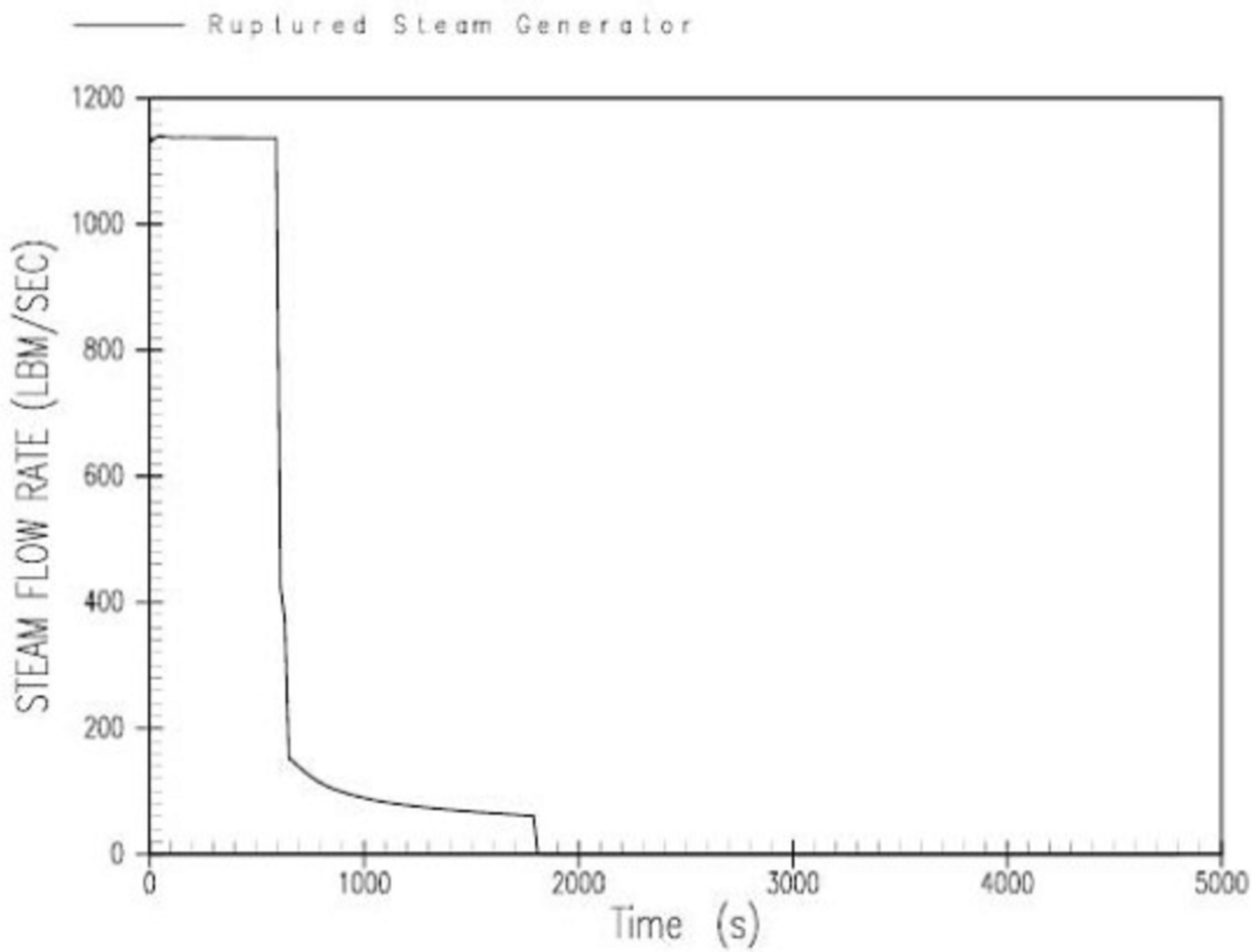
~~Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.~~

Delete Note

CALLAWAY PLANT
FIGURE 15.6-3D STEAM FLOW RATE (INTACT GENERATORS) TRANSIENT FOR STEAM GENERATOR TUBE RUPTURE EVENT REV. 16 6/11


Intact Steam Generators (Total)
4000
3000
2000
1000
0
PRESSURE (PSIA)
0
1000
2000
3000
4000
5000
Time (s)

Replace with graph on Tab E from spreadsheet "SGTR_FSAR_Plots.xlsx"

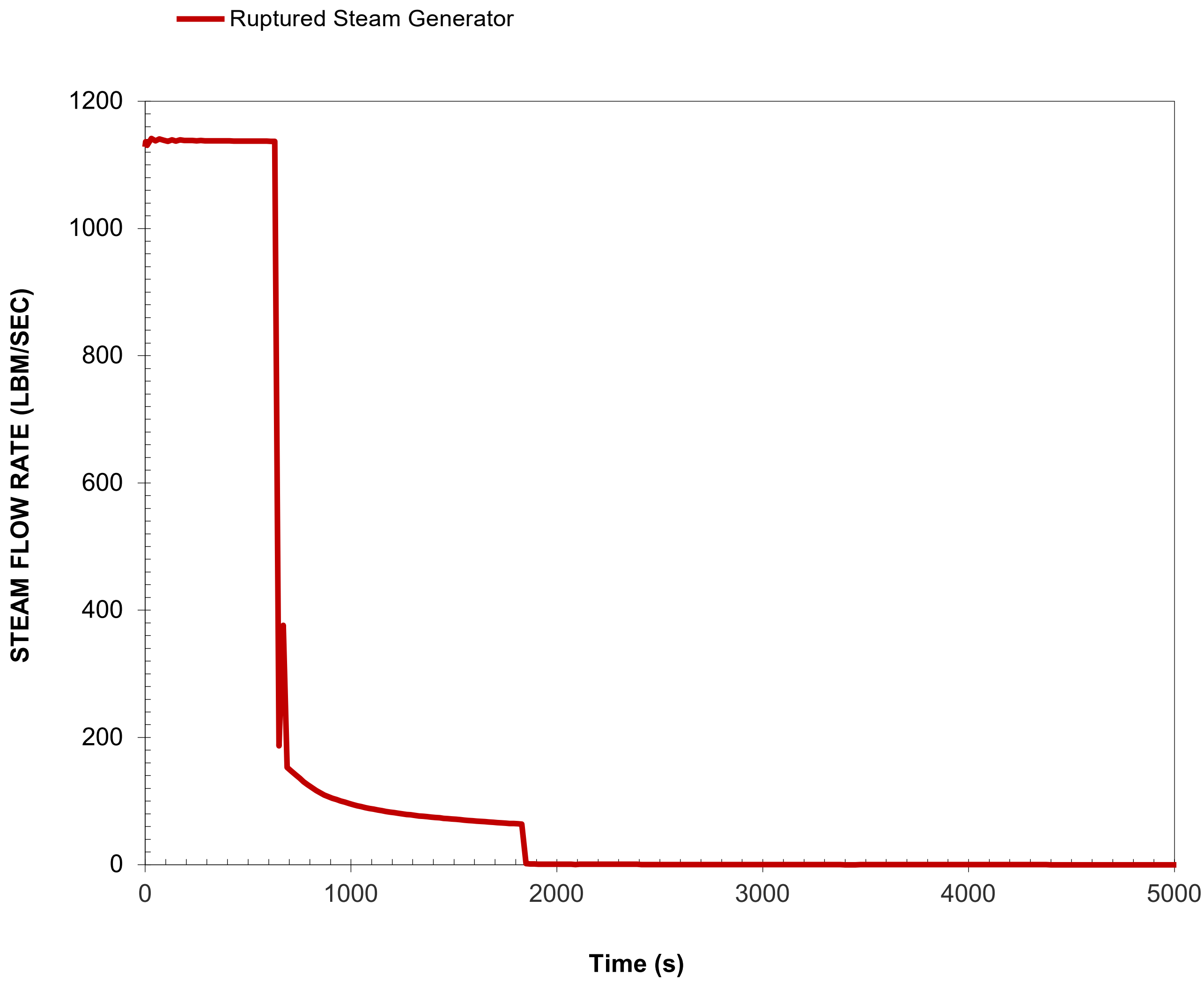


Note:

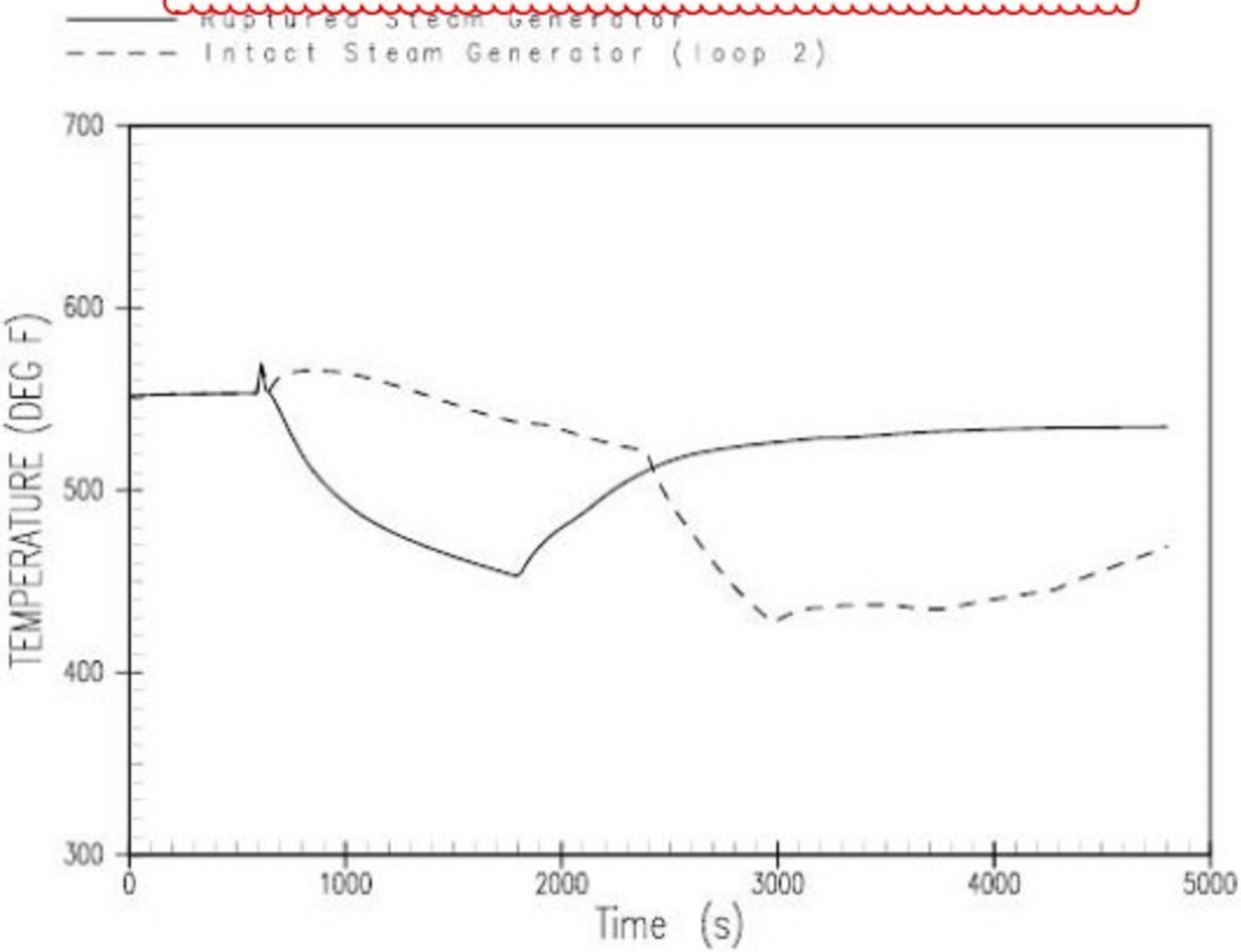
The thermal-hydraulic model presented in this figure is based on an analysis that credits all three ASDs on the intact steam generators t Delete Note the RCS rapid cooldown. Thi pect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.

Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.

CALLAWAY PLANT
FIGURE 15.6-3E STEAM FLOW RATE (RUPTURED GENERATOR) TRANSIENT FOR STEAM GENERATOR TUBE RUPTURE EVENT REV. 16 6/11


Ruptured Steam Generator
STEAM FLOW RATE (LBM/SEC)
1200
1000
800
600
400
200
0
0
1000
2000
3000
4000
5000
Time (s)

Replace with graph on Tab F from spreadsheet "SGTR_FSAR_Plots.xlsx"



~~Note:~~

~~The thermal-hydraulic m~~ Delete Note ~~is based on an analysis that credits all three ASDs on the intact steam generators to be available to support the RCS rapid cooldown. This is conservative with respect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.~~

~~Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.~~

CALLAWAY PLANT

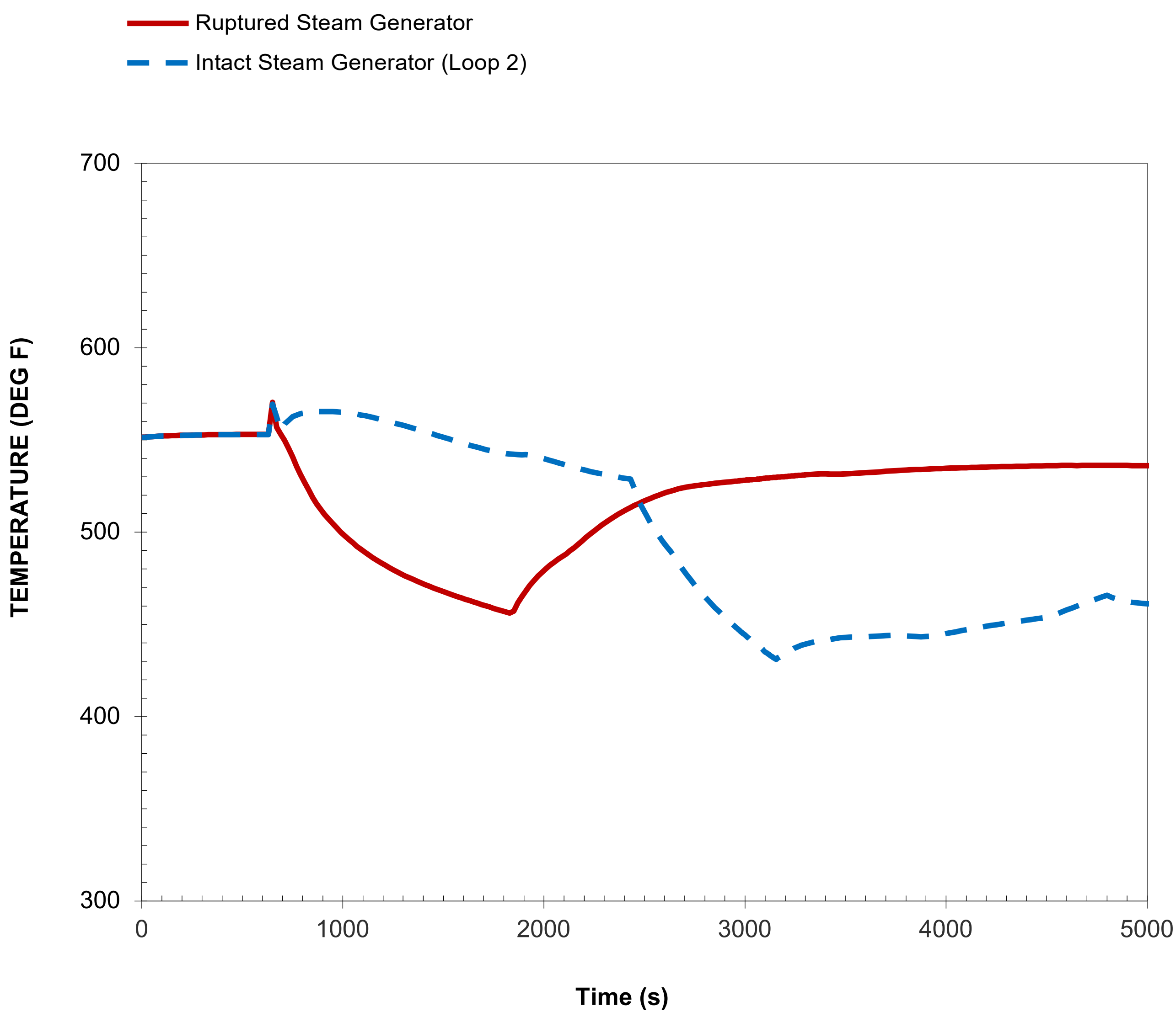
FIGURE 15.6-3F

STEAM GENERATOR TEMPERATURE (RUPTURED AND INTACT GENERATORS) TRANSIENTS FOR STEAM GENERATOR TUBE RUPTURE EVENT

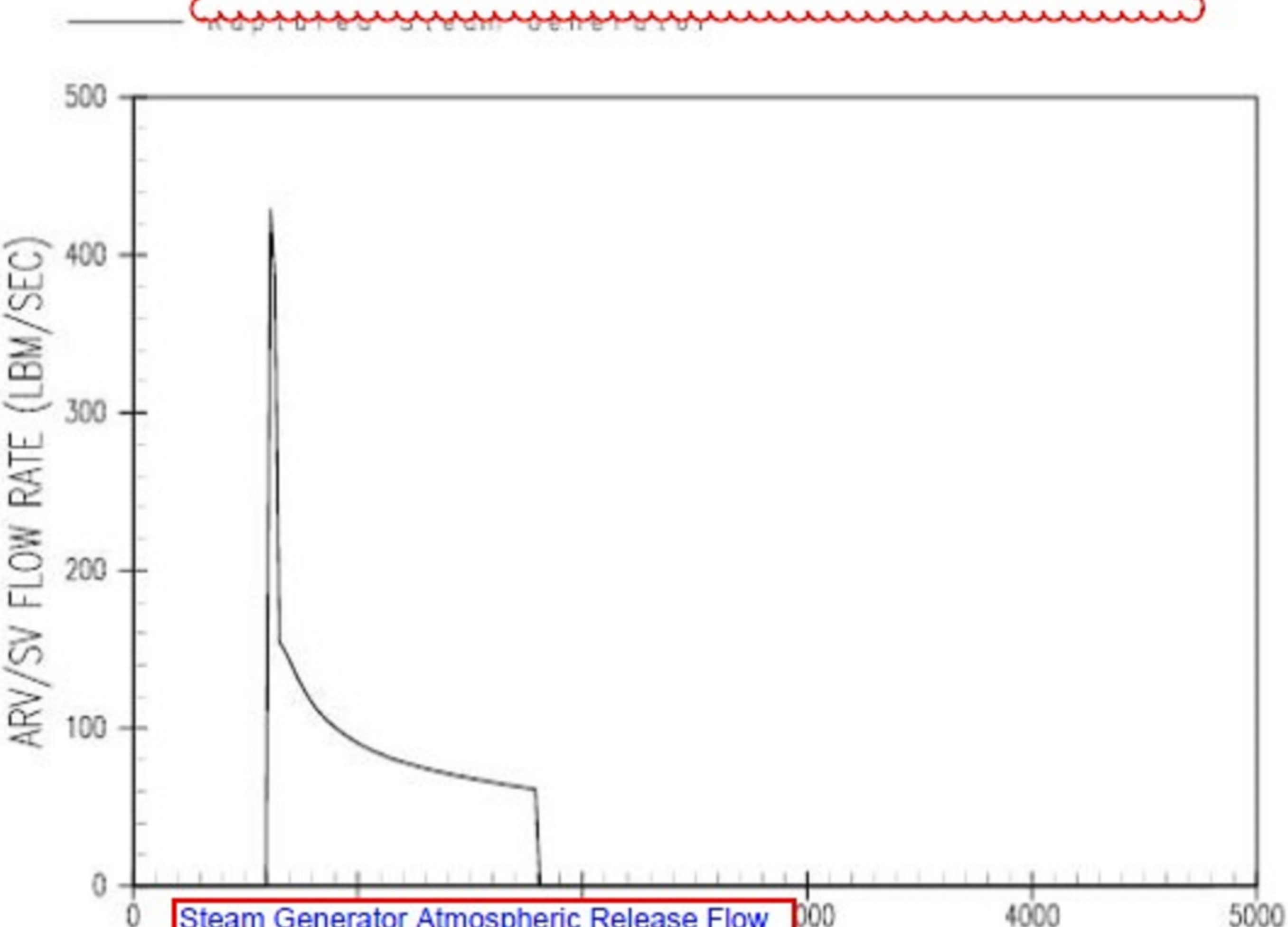
REV. 16 6/11

SGTR_FSAR_Plots.xlsx

Tab F



Replace with graph on Tab G from spreadsheet "SGTR_FSAR_Plots.xlsx"



Steam Generator Atmospheric Release Flow Rate (Ruptured Generator) Transient for Steam Generator Tube Rupture Event

~~Note:~~

~~The thermal-hydraulic model presented in this figure is based on an analy~~ Delete Note ~~on the intact steam gener~~ ~~t the RCS rapid cooldown. This is conservative with respect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.~~

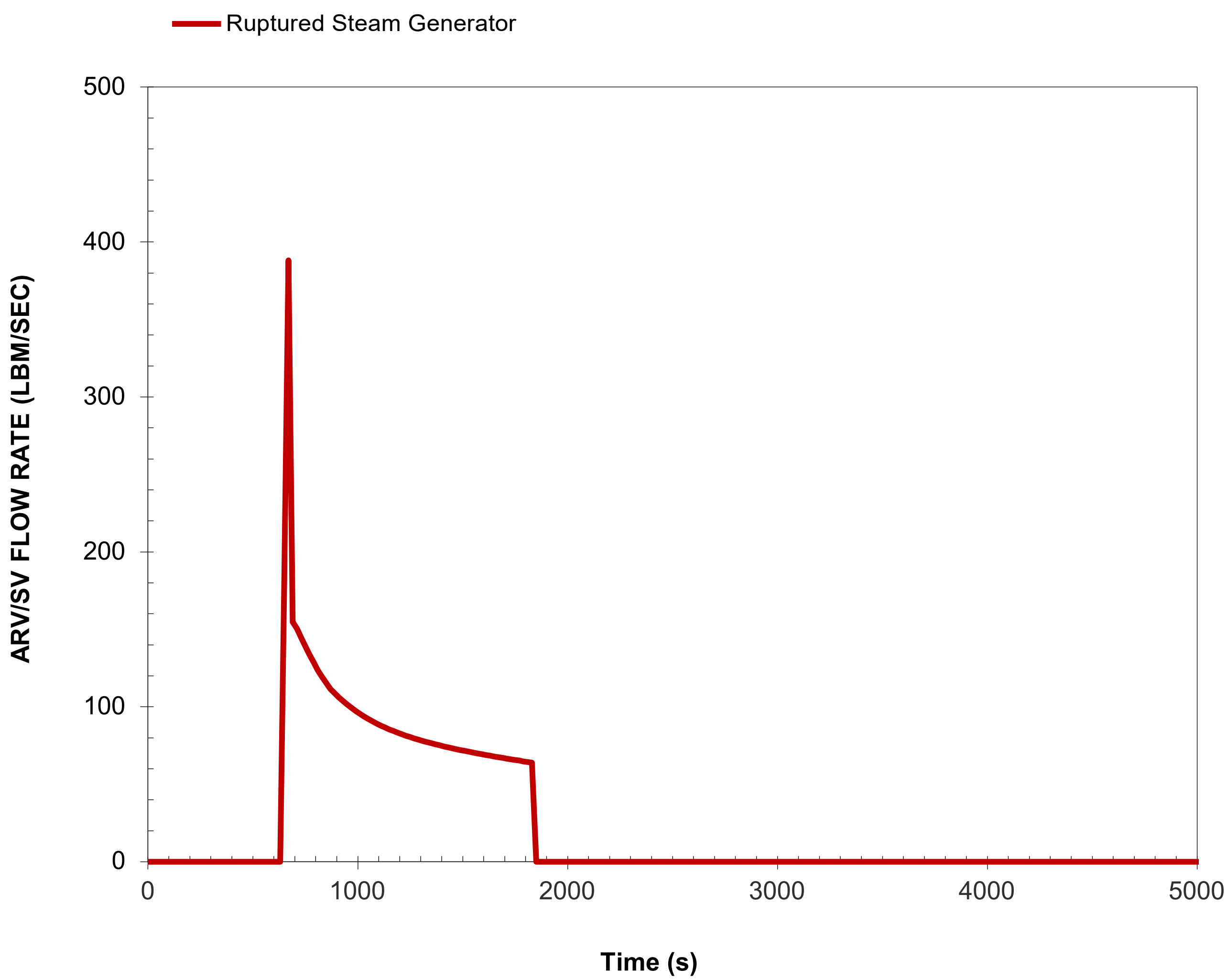
~~Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.~~

CALLAWAY PLANT

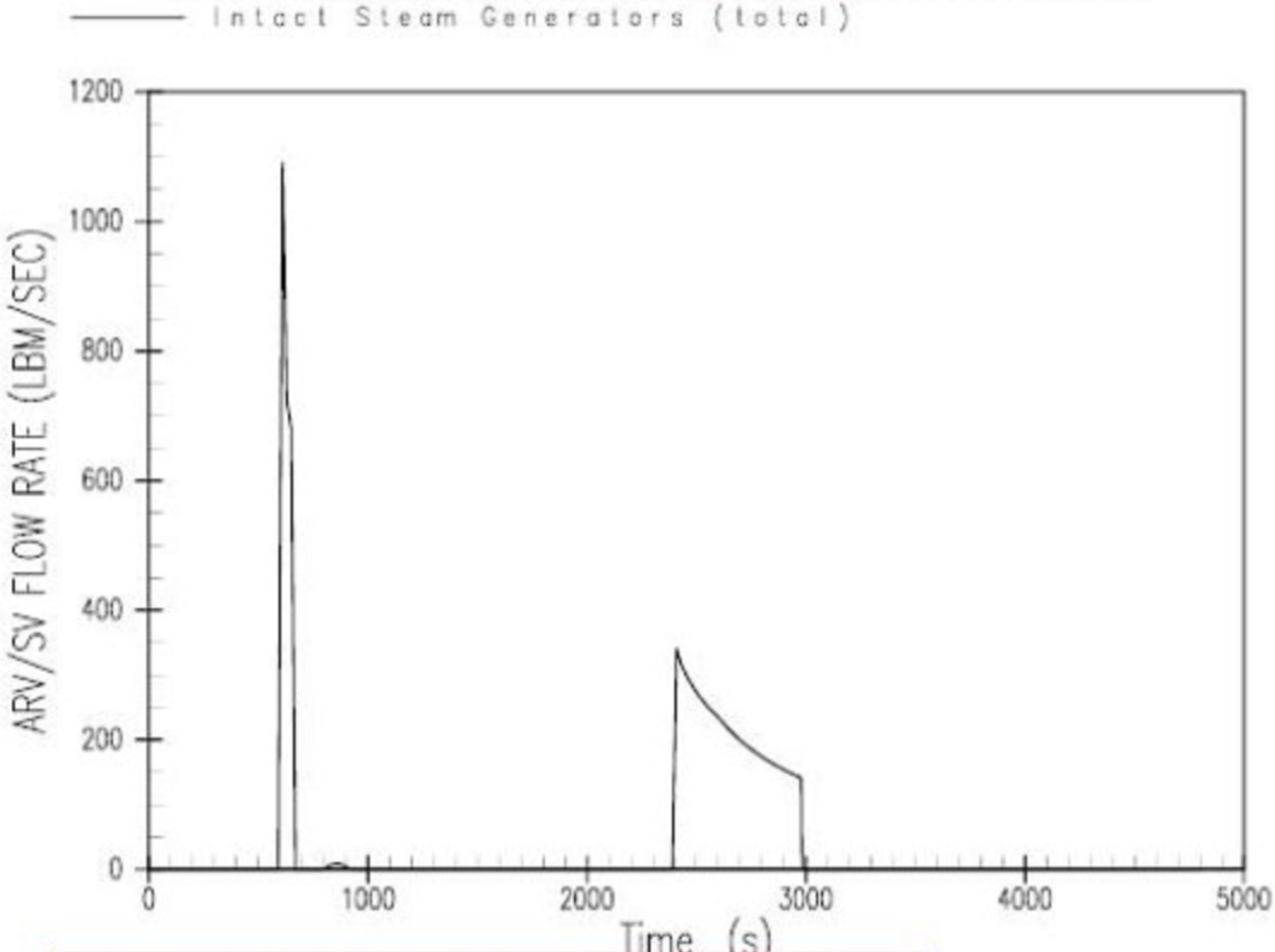
FIGURE 15.6-3G

~~STEAM GENERATOR ATMOSPHERIC RELIEF VALVE FLOW RATE (RUPTURED GENERATOR) TRANSIENT FOR STEAM GENERATOR TUBE RUPTURE EVENT~~

REV. 16 4/11


Ruptured Steam Generator
ARV/SV FLOW RATE (LBM/SEC)
500
400
300
200
100
0
0
1000
2000
3000
4000
5000
Time (s)

Replace with graph on Tab H from spreadsheet "SGTR_FSAR_Plots.xlsx"



Steam Generator Atmospheric Release Flow Rate (Intact Generators) Transient for Steam Generator Tube Rupture Event

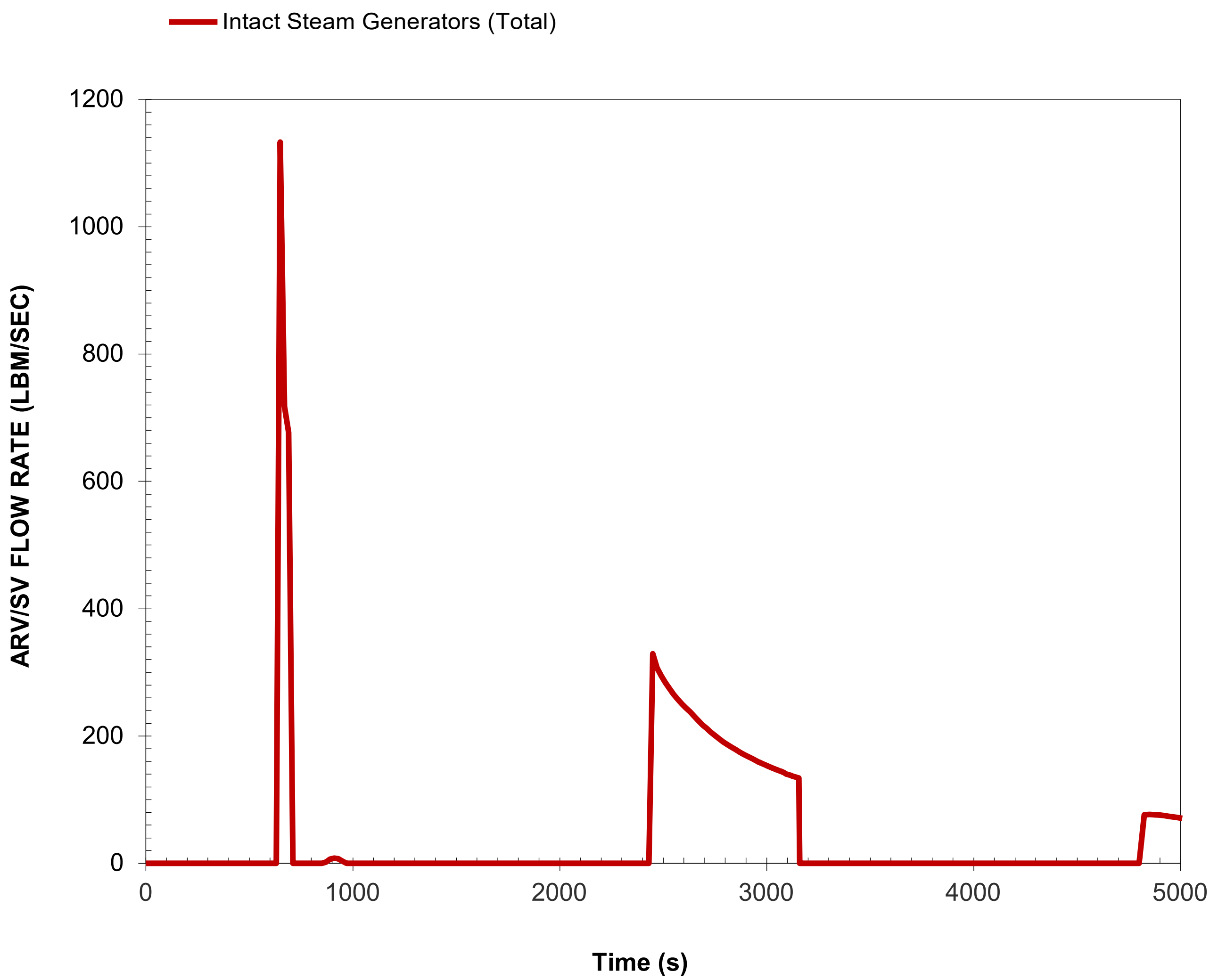
~~Note:~~

~~The thermal-hydraulic ... this figure is based on an analysis ... ASDs on the intact steam generator ... support the RCS rapid cooldown. This is conservative with respect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.~~

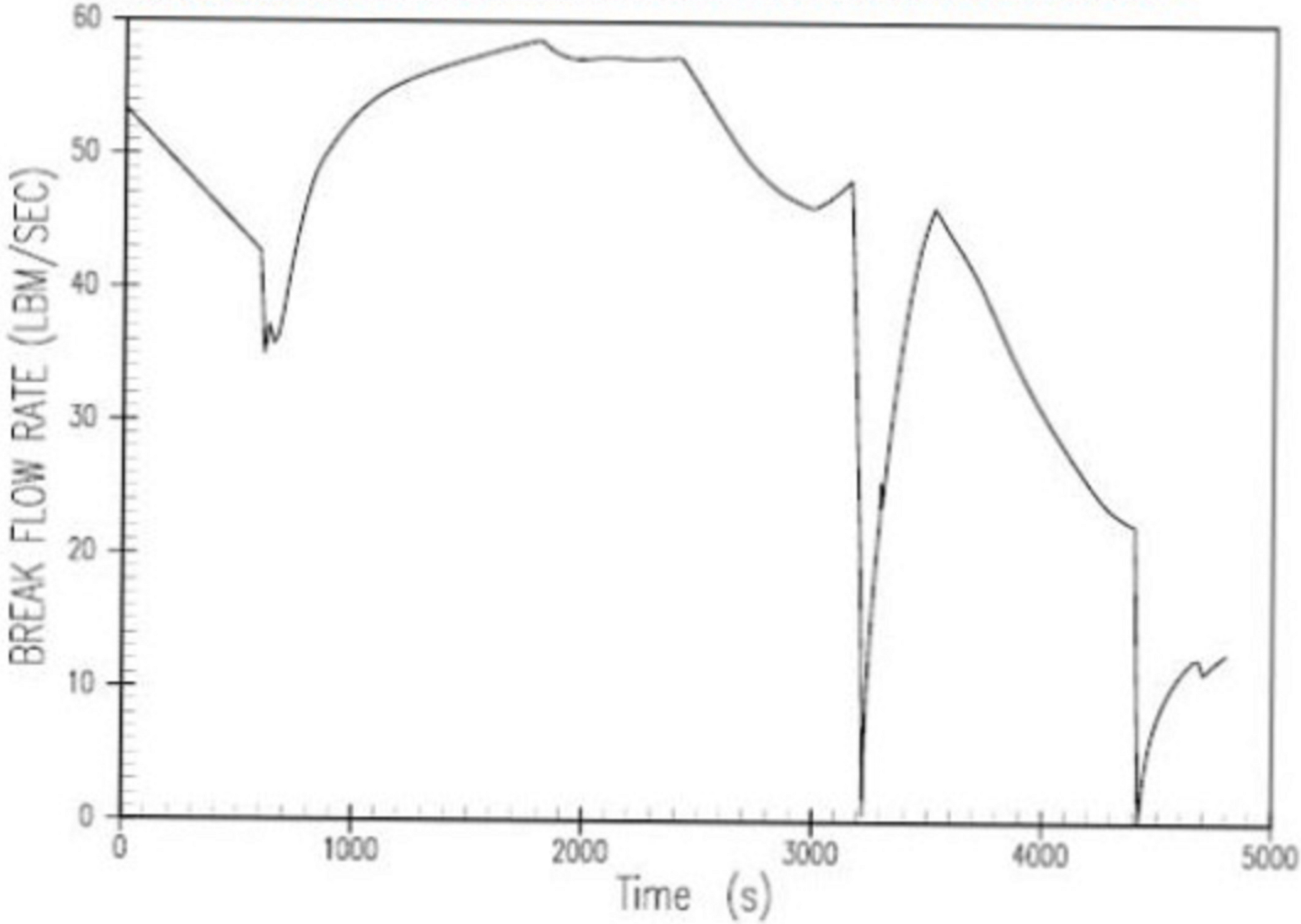
Delete Note

~~Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.~~

CALLAWAY PLANT
FIGURE 15.6-3H ~~STEAM GENERATOR ATMOSPHERIC RELIEF VALVE FLOW RATE (INTACT GENERATOR) TRANSIENT FOR STEAM GENERATOR TUBE RUPTURE EVENT~~ REV. 18 5/11


Intact Steam Generators (Total)
ARV/SV FLOW RATE (LBM/SEC)
1200
1000
800
600
400
200
0
0
1000
2000
3000
4000
5000
Time (s)

Replace with graph on Tab I ("indigo") from spreadsheet "SGTR_FSAR_Plots.xlsx"



Note:

The thermal-hydra[illegible] Delete Note [illegible]gure is based on an analy[illegible], on the intact steam gener[illegible]t the RCS rapid cooldow[illegible]espect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.

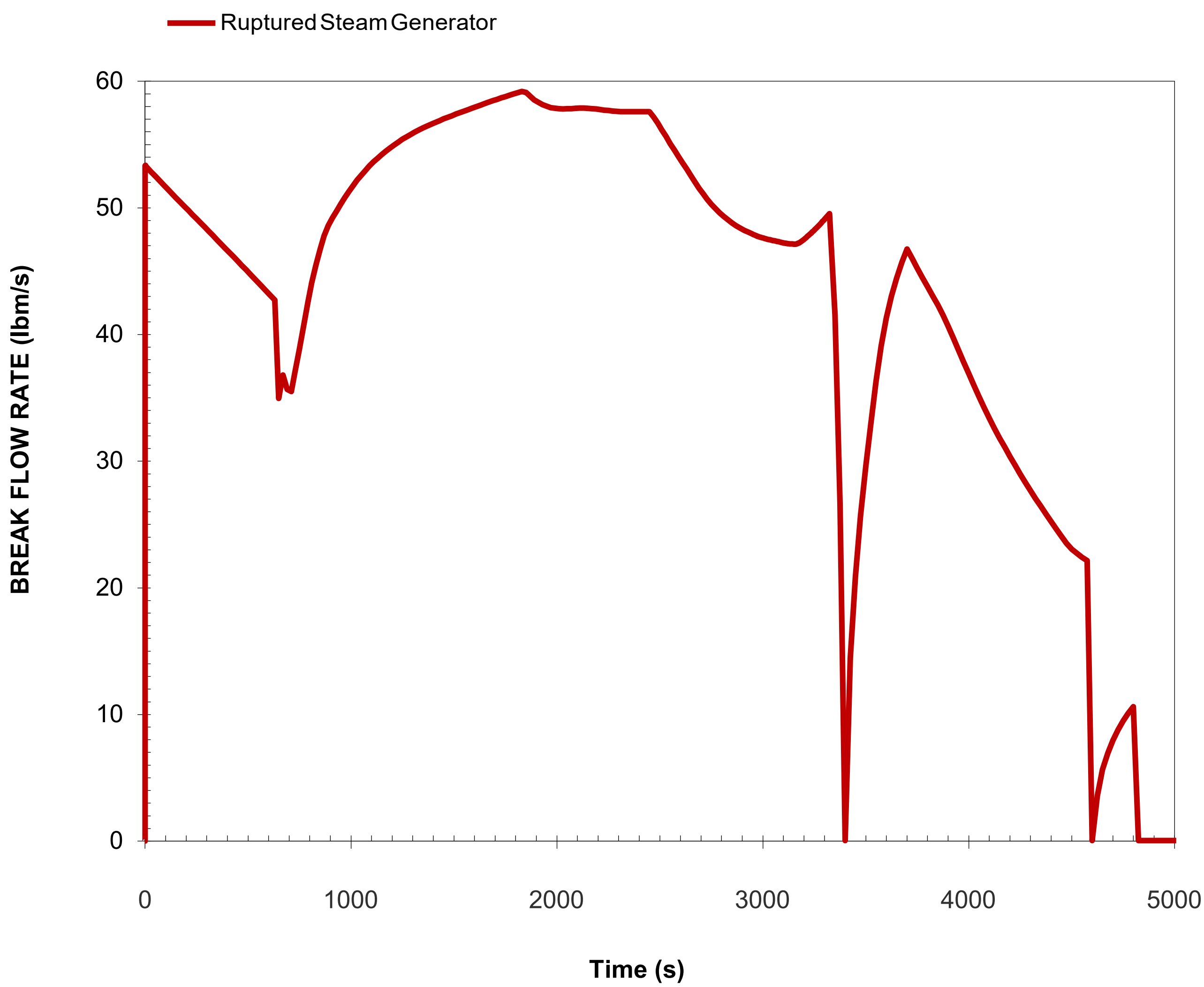
Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.

CALLAWAY PLANT

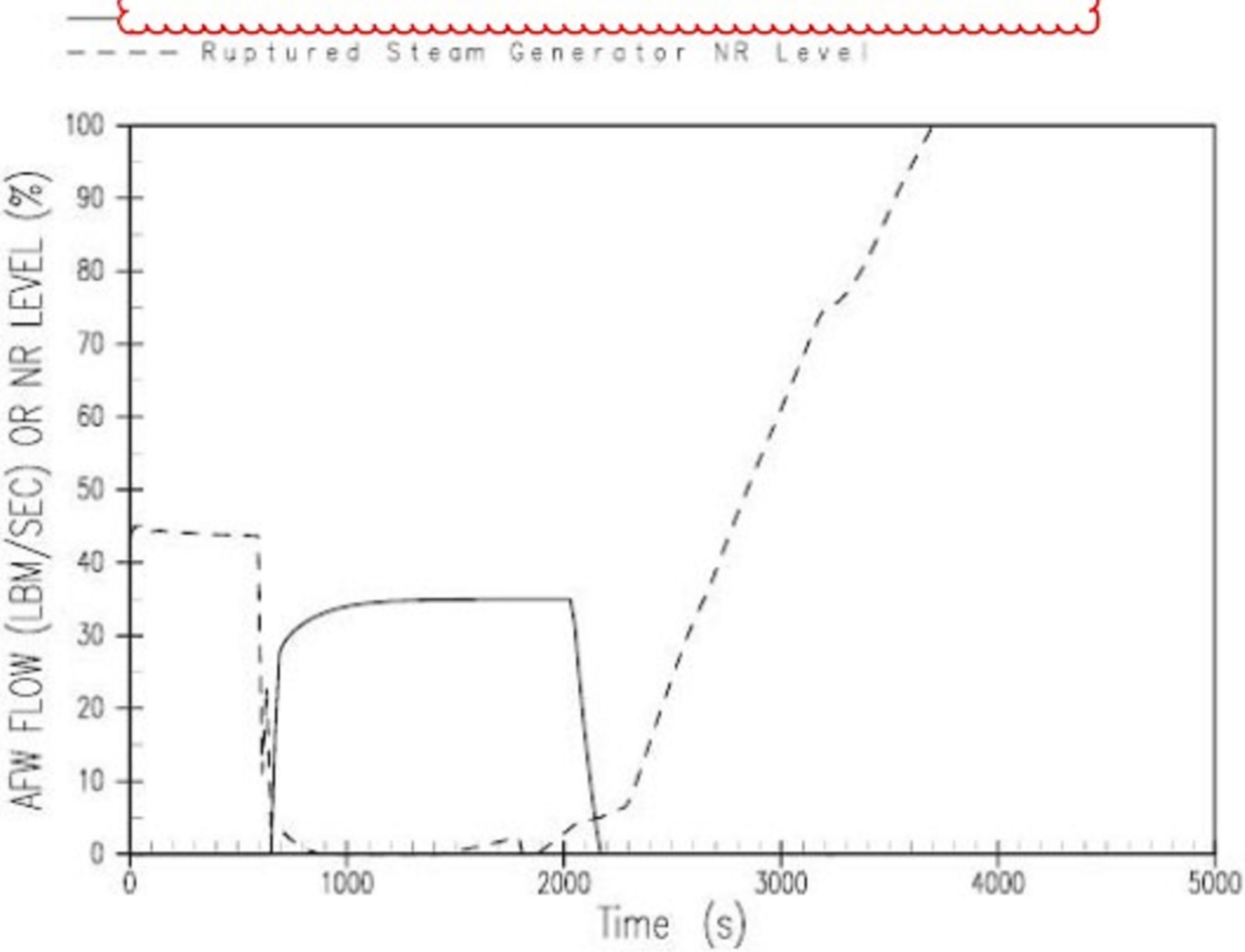
FIGURE 15.6-3I

RUPTURED STEAM GENERATOR BREAK FLOW TRANSIENT FOR STEAM GENERATOR TUBE RUPTURE EVENT

REV. 16 5/11


Ruptured Steam Generator
60
50
40
30
20
10
0
BREAK FLOW RATE (lbm/s)
0
1000
2000
3000
4000
5000
Time (s)

Replace with graph on Tab J from spreadsheet "SGTR_FSAR_Plots.xlsx"



~~Note:~~

~~The thermal-hydra[illegible] figure is based on an analy[illegible] on the intact steam generators to be available to support the RCS rapid cooldown. This is conservative with respect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.~~

Delete Note

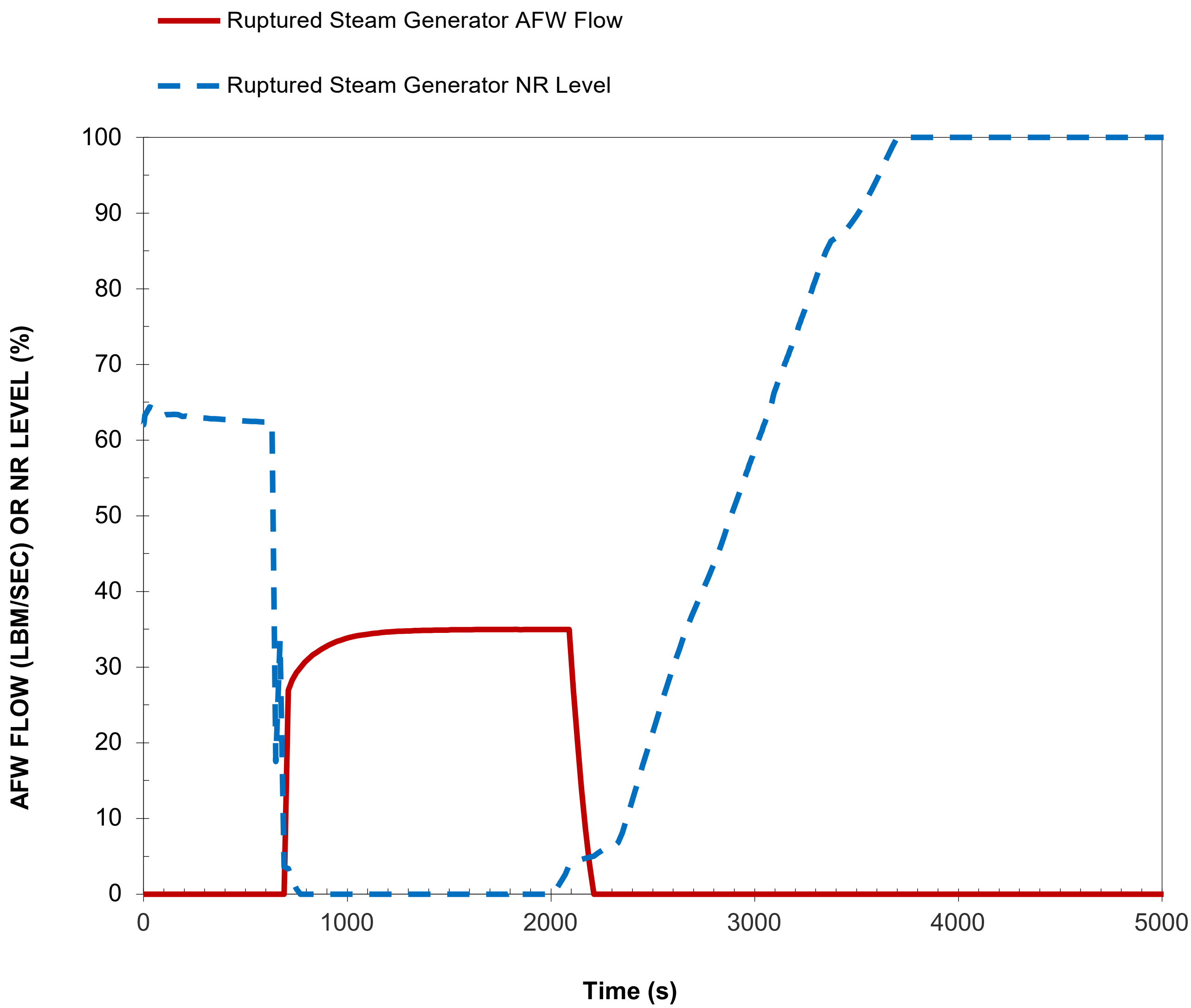
~~Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.~~

CALLAWAY PLANT

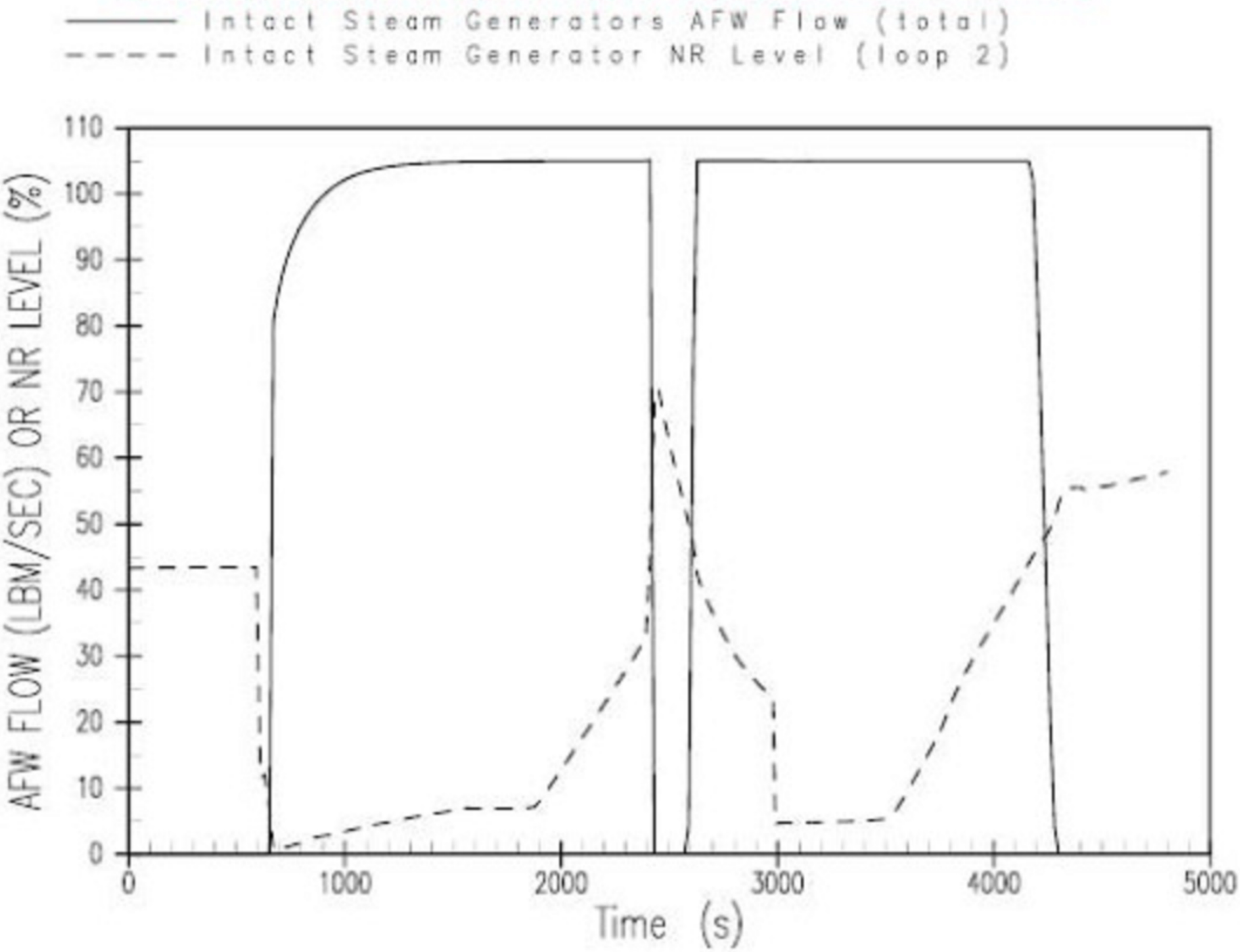
FIGURE 15.6-3J

AUXILIARY FEEDWATER FLOW RATE AND NARROW RANGE LEVEL (RUPTURED GENERATOR) TRANSIENTS FOR STEAM GENERATOR TUBE RUPTURE EVENT

REV. 16 5/11


Ruptured Steam Generator AFW Flow
Ruptured Steam Generator NR Level
AFW FLOW (LBM/SEC) OR NR LEVEL (%)
100
90
80
70
60
50
40
30
20
10
0
0
1000
2000
3000
4000
5000
Time (s)

Replace with graph on Tab K from spreadsheet "SGTR_FSAR_Plots.xlsx"



Note:

The thermal-hydrau[illegible] Delete Note [illegible]igure is based on an analys[illegible]s on the intact steam gener[illegible]ort the RCS rapid cooldown. This is conservative with respect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.

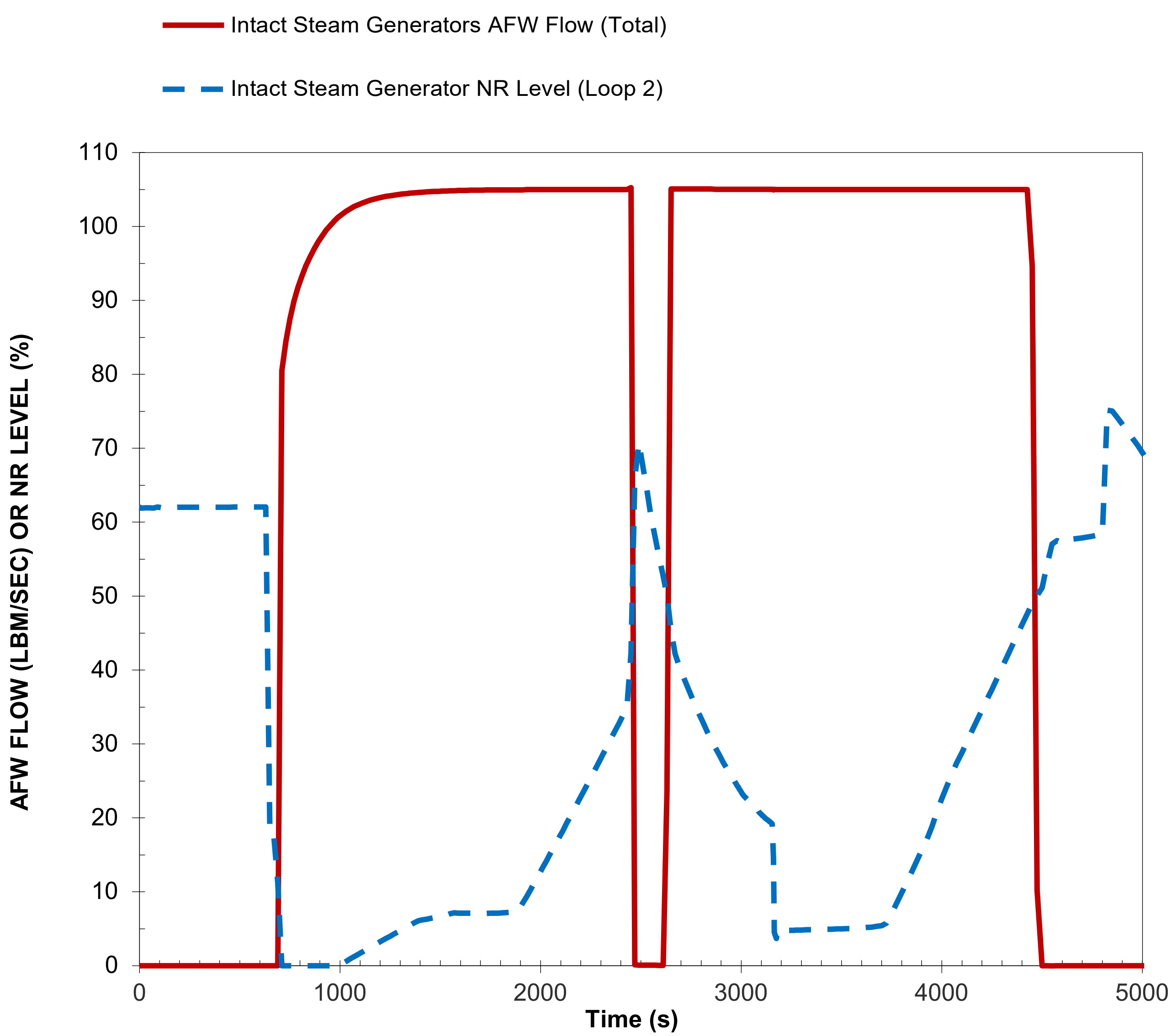
Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.

CALLAWAY PLANT

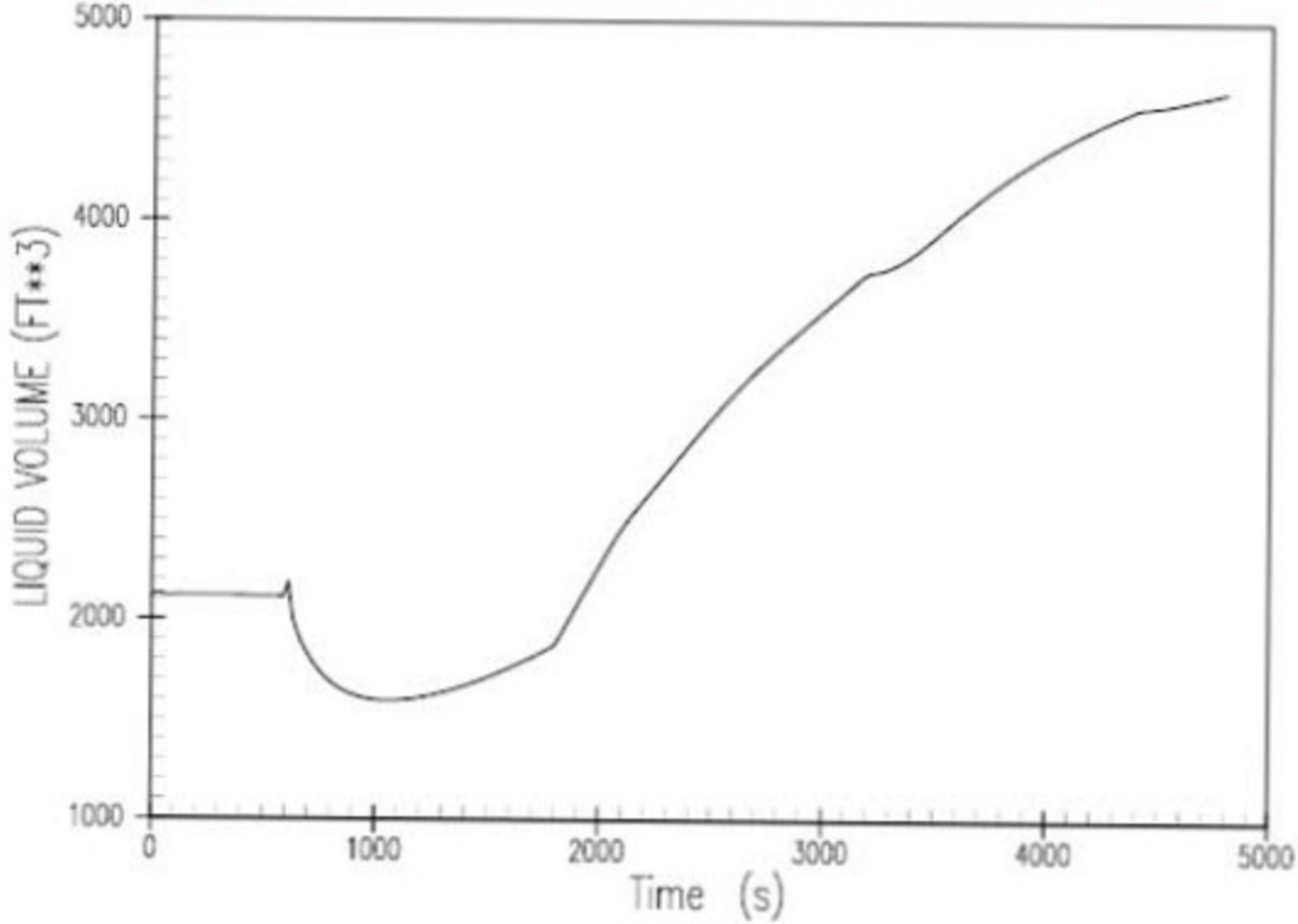
FIGURE 15.6-3K

AUXILIARY FEEDWATER FLOW RATE AND NARROW RANGE LEVEL (INTACT GENERATOR) TRANSIENTS FOR STEAM GENERATOR TUBE RUPTURE EVENT

REV. 16 5/11


Intact Steam Generators AFW Flow (Total)
Intact Steam Generator NR Level (Loop 2)
AFW FLOW (LBM/SEC) OR NR LEVEL (%)
110
100
90
80
70
60
50
40
30
20
10
0
0
1000
2000
3000
4000
5000
Time (s)

Replace with graph on Tab L from spreadsheet "SGTR_FSAR_Plots.xlsx"



~~Note:~~

Delete Note

~~The thermal-hydra[illegible] figure is based on an analy[illegible]s on the intact steam gener[illegible]ort the RCS rapid cooldown. This is conservative with respect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.~~

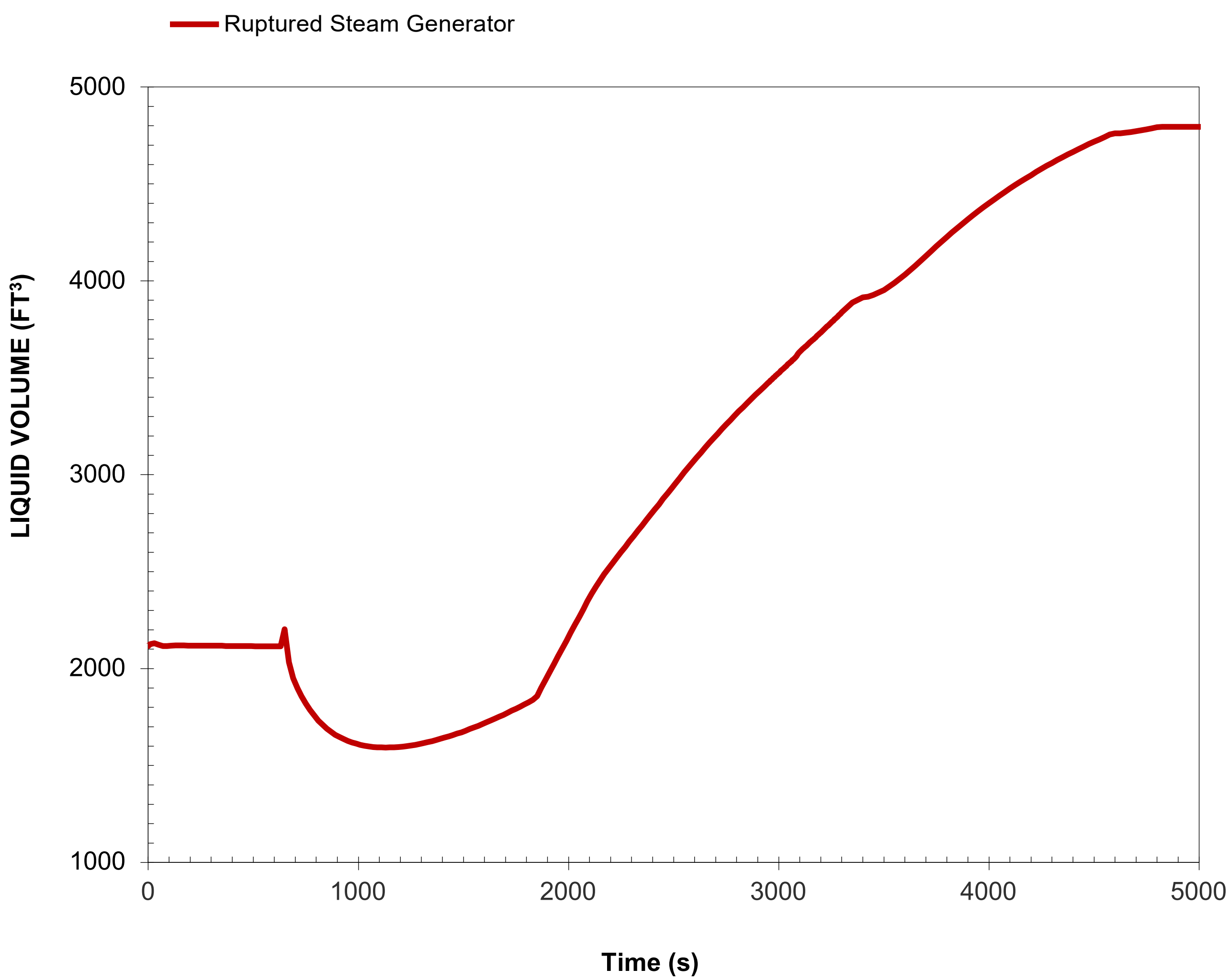
~~Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.~~

CALLAWAY PLANT

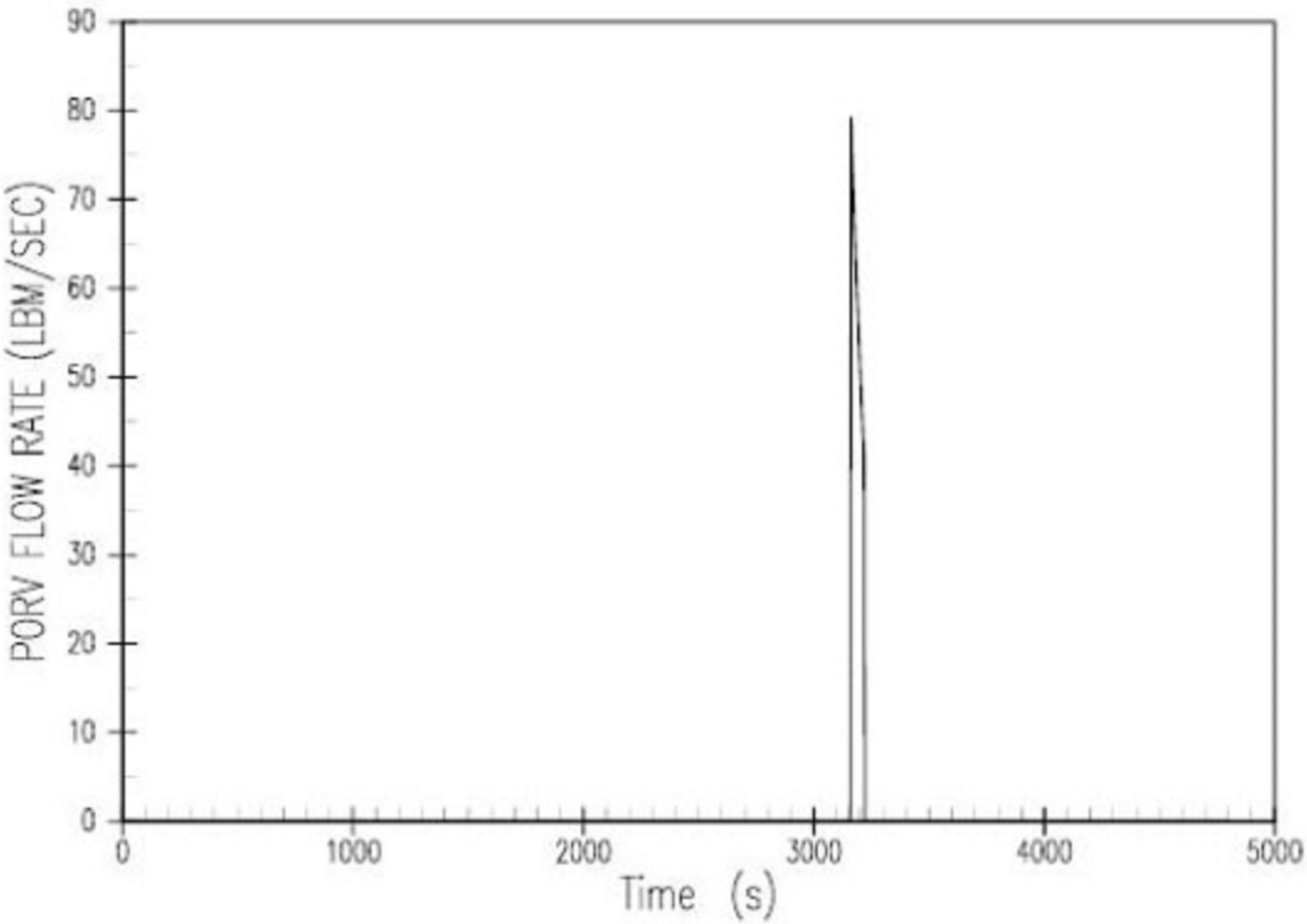
FIGURE 15.6-3L

STEAM GENERATOR LIQUID VOLUME (RUPTURED GENERATOR) TRANSIENT FOR STEAM GENERATOR TUBE RUPTURE EVENT

REV. 16 6/11


Ruptured Steam Generator
5000
4000
3000
2000
1000
LIQUID VOLUME (FT³)
0
1000
2000
3000
4000
5000
Time (s)

Replace with graph on Tab M from spreadsheet "SGTR_FSAR_Plots.xlsx"



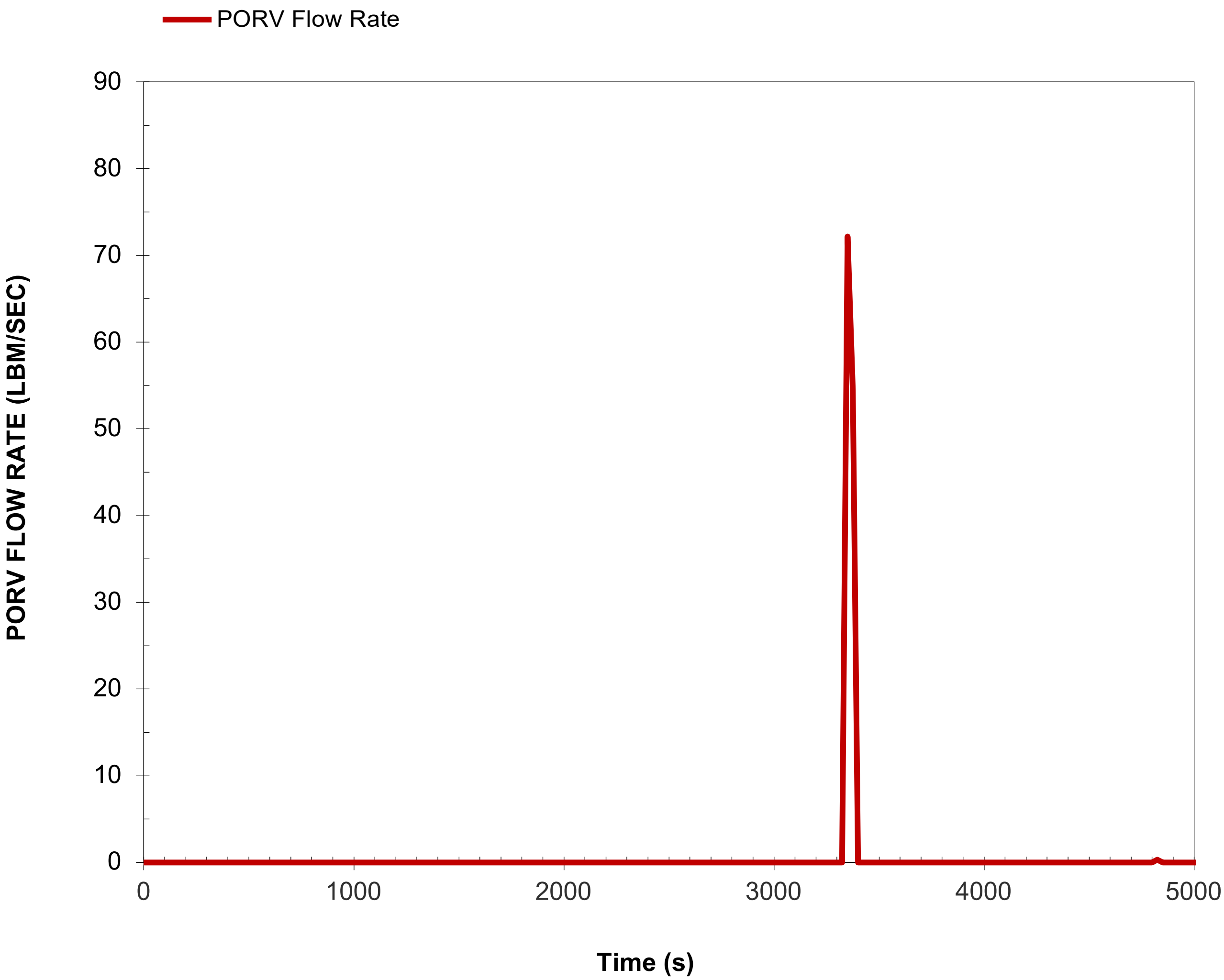
~~Note:~~

~~The thermal-hydra~~ [illegible] ~~gure is based on an analy~~ [illegible] ~~, on the intact steam gener~~ [illegible] ~~rt the RCS rapid cooldow~~ [illegible] ~~respect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.~~

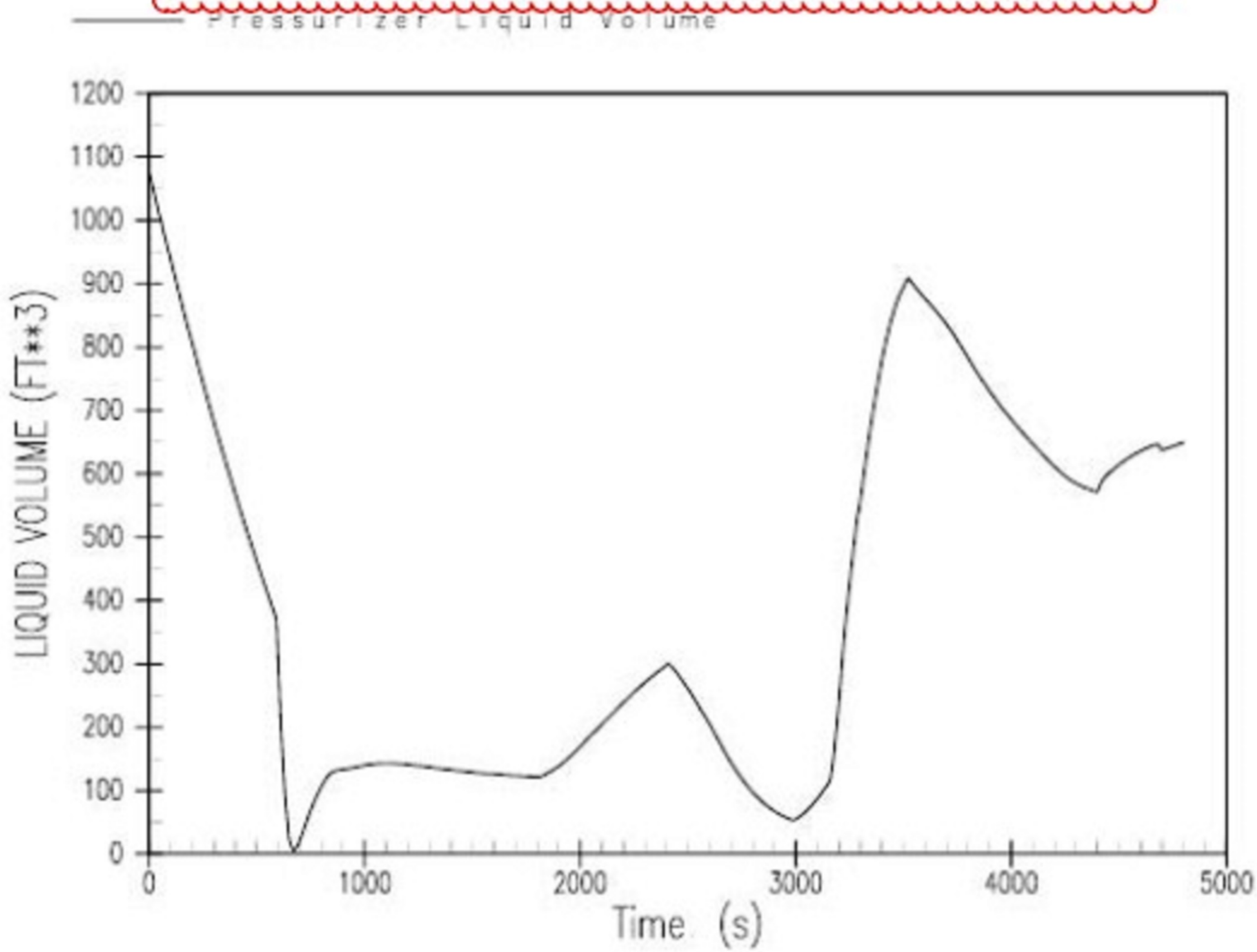
Delete Note

~~Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.~~

CALLAWAY PLANT
FIGURE 15.6-3M PRESSURIZER PORV FLOW RATE TRANSIENT FOR STEAM GENERATOR TUBE RUPTURE EVENT REV. 16 6/11


PORV Flow Rate
PORV FLOW RATE (LBM/SEC)
0
10
20
30
40
50
60
70
80
90
0
1000
2000
3000
4000
5000
Time (s)

Replace with graph on Tab N from spreadsheet "SGTR_FSAR_Plots.xlsx"



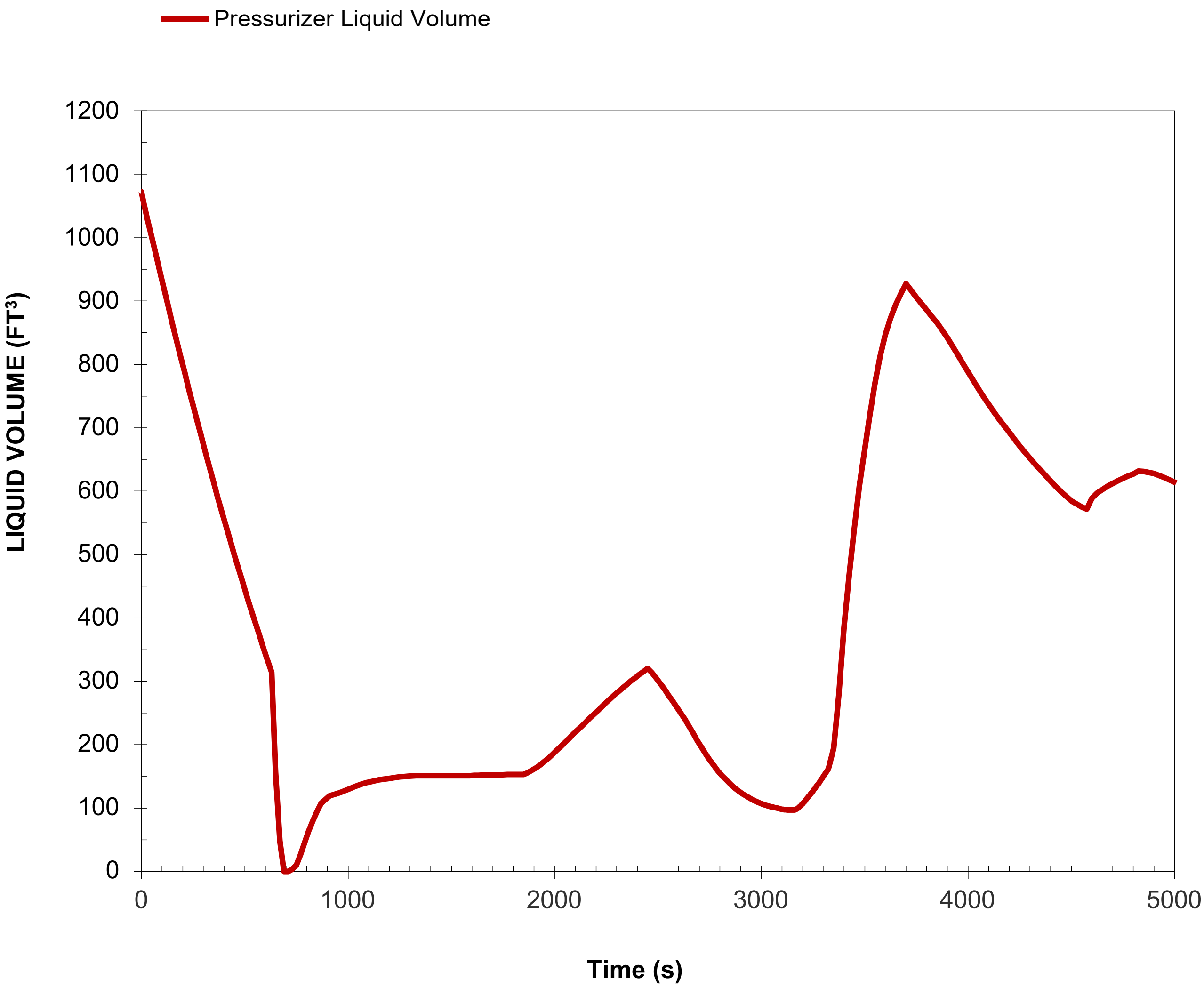
Delete Note

~~**Note:**~~

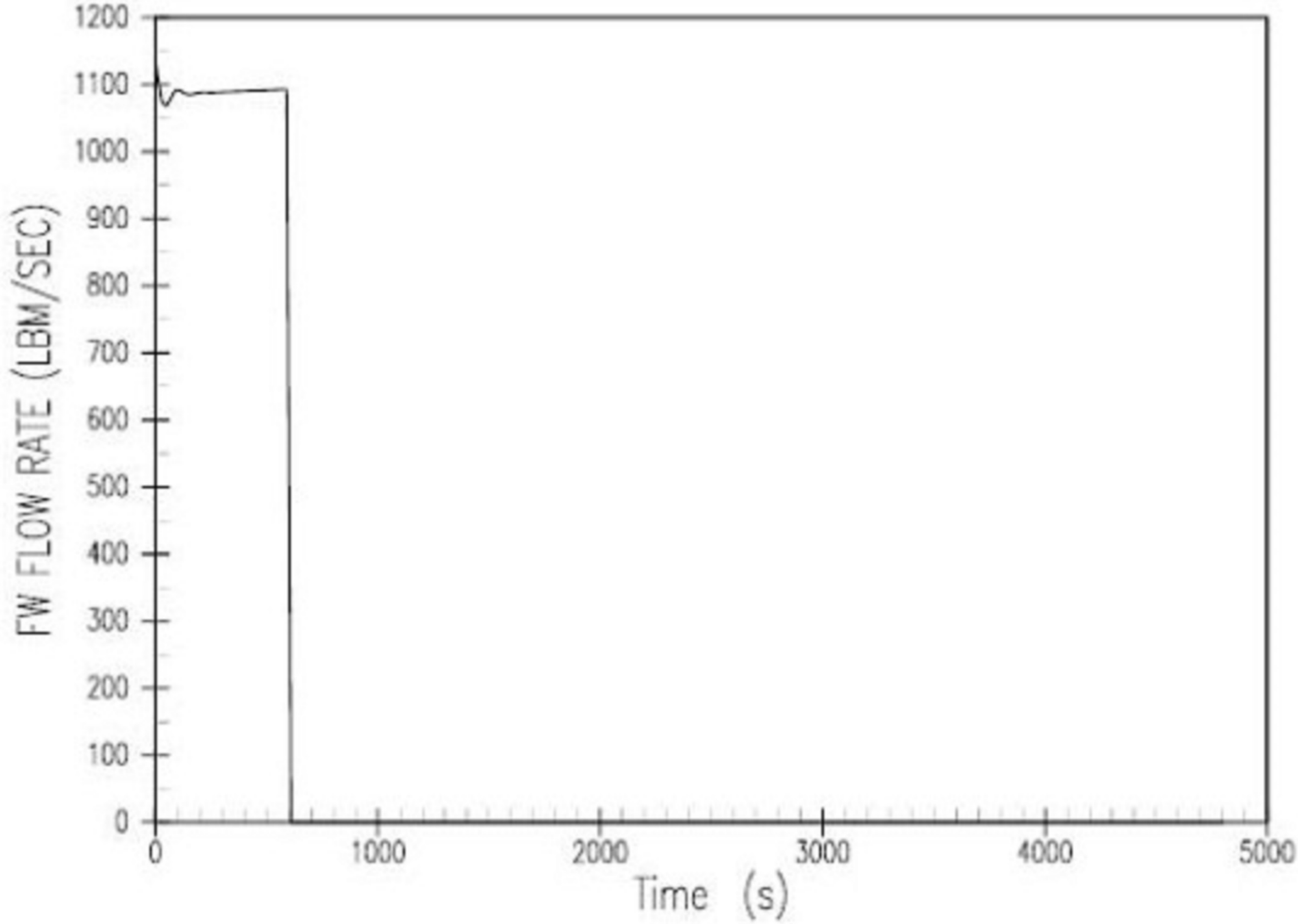
~~The thermal-hydraulic [illegible] this figure is based on an analysis [illegible] ASDs on the intact steam generators to be available to support the RCS rapid cooldown. This is conservative with respect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.~~

~~Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.~~

CALLAWAY PLANT
FIGURE 15.6-3N **PRESSURIZER LIQUID VOLUME TRANSIENT FOR STEAM GENERATOR TUBE RUPTURE EVENT** **REV. 10 5/11**


Pressurizer Liquid Volume
LIQUID VOLUME (FT³)
1200
1100
1000
900
800
700
600
500
400
300
200
100
0
0
1000
2000
3000
4000
5000
Time (s)

Replace with graph on Tab O from spreadsheet "SGTR_FSAR_Plots.xlsx"



~~**Note:**~~

~~The thermal-hydraulic model presented in this figure is based on an analys[illegible] three ASDs on the intact steam gener[illegible]le to support the RCS rapid cooldown. This is conservative with respect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.~~

~~Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.~~

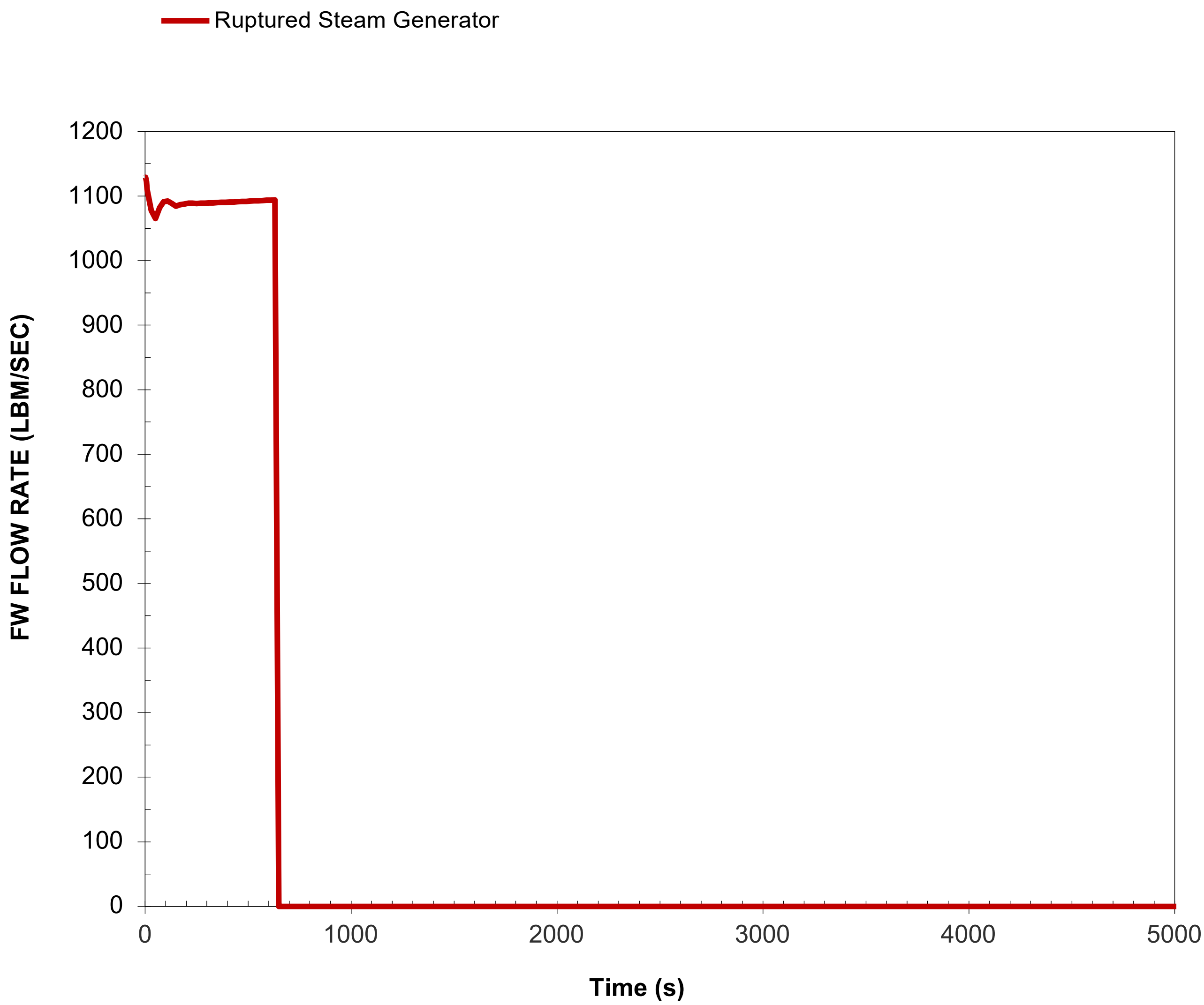
Delete Note

CALLAWAY PLANT

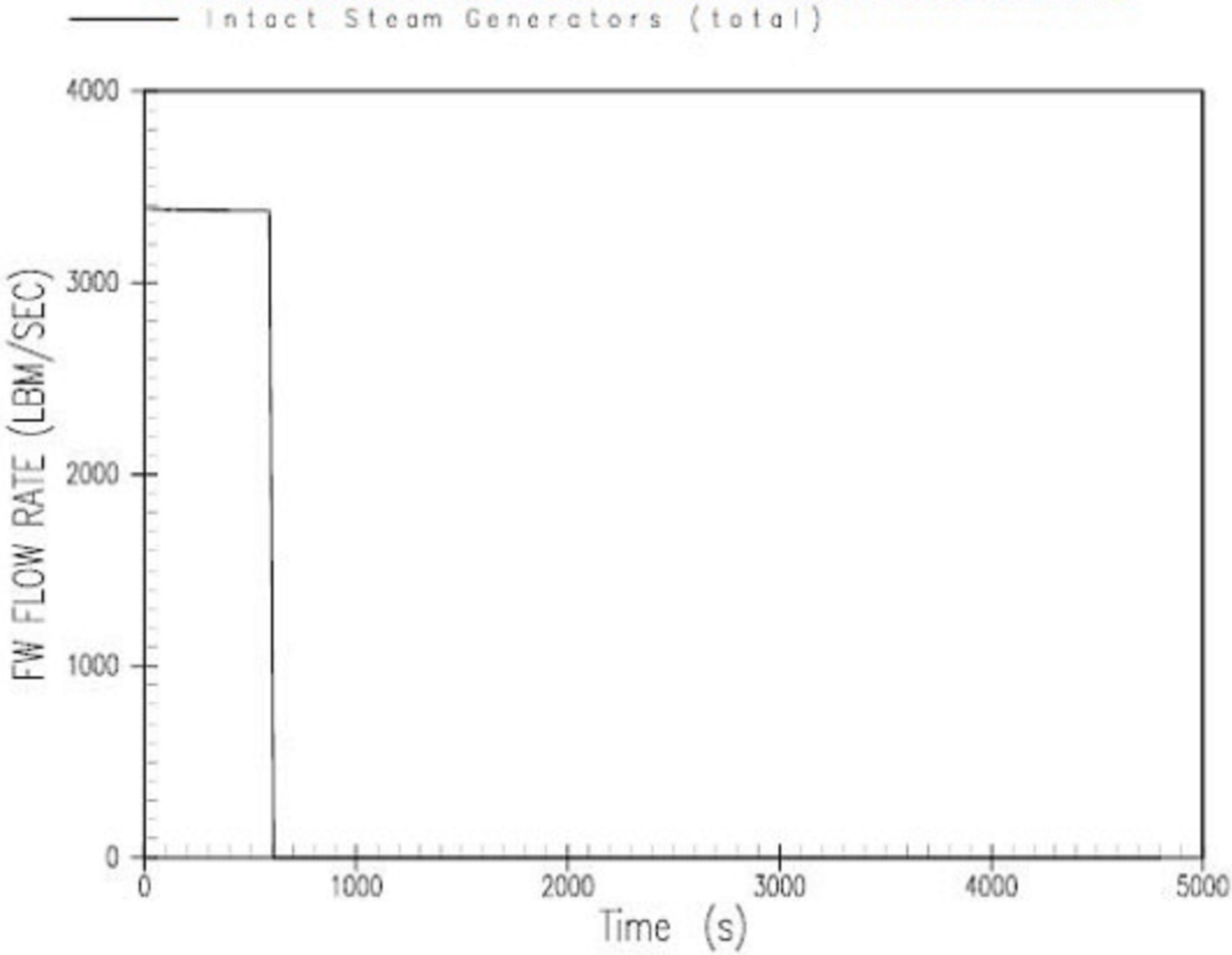
FIGURE 15.6-3O

FEEDWATER FLOW RATE (RUPTURED GENERATOR) TRANSIENT FOR STEAM GENERATOR TUBE RUPTURE EVENT

REV. 16 5/11


Ruptured Steam Generator
FW FLOW RATE (LBM/SEC)
1200
1100
1000
900
800
700
600
500
400
300
200
100
0
0
1000
2000
3000
4000
5000
Time (s)

Replace with graph on Tab P from spreadsheet "SGTR_FSAR_Plots.xlsx"



Note:

The thermal-hydraulic model presented in this figure is based on an analys[illegible] s on the intact steam genera[illegible] rt the RCS rapid cooldown [illegible] respect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.

Delete Note

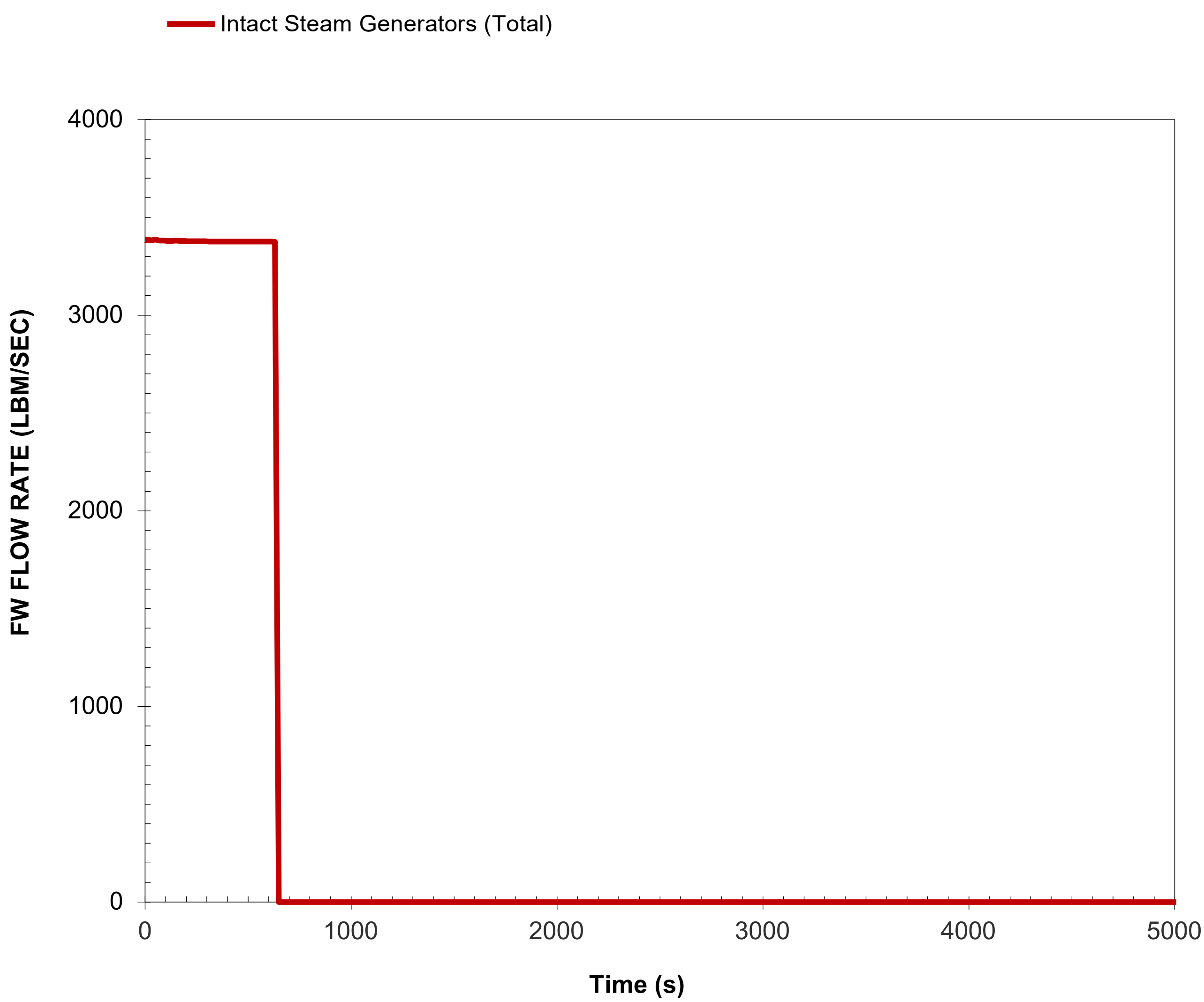
Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.

CALLAWAY PLANT

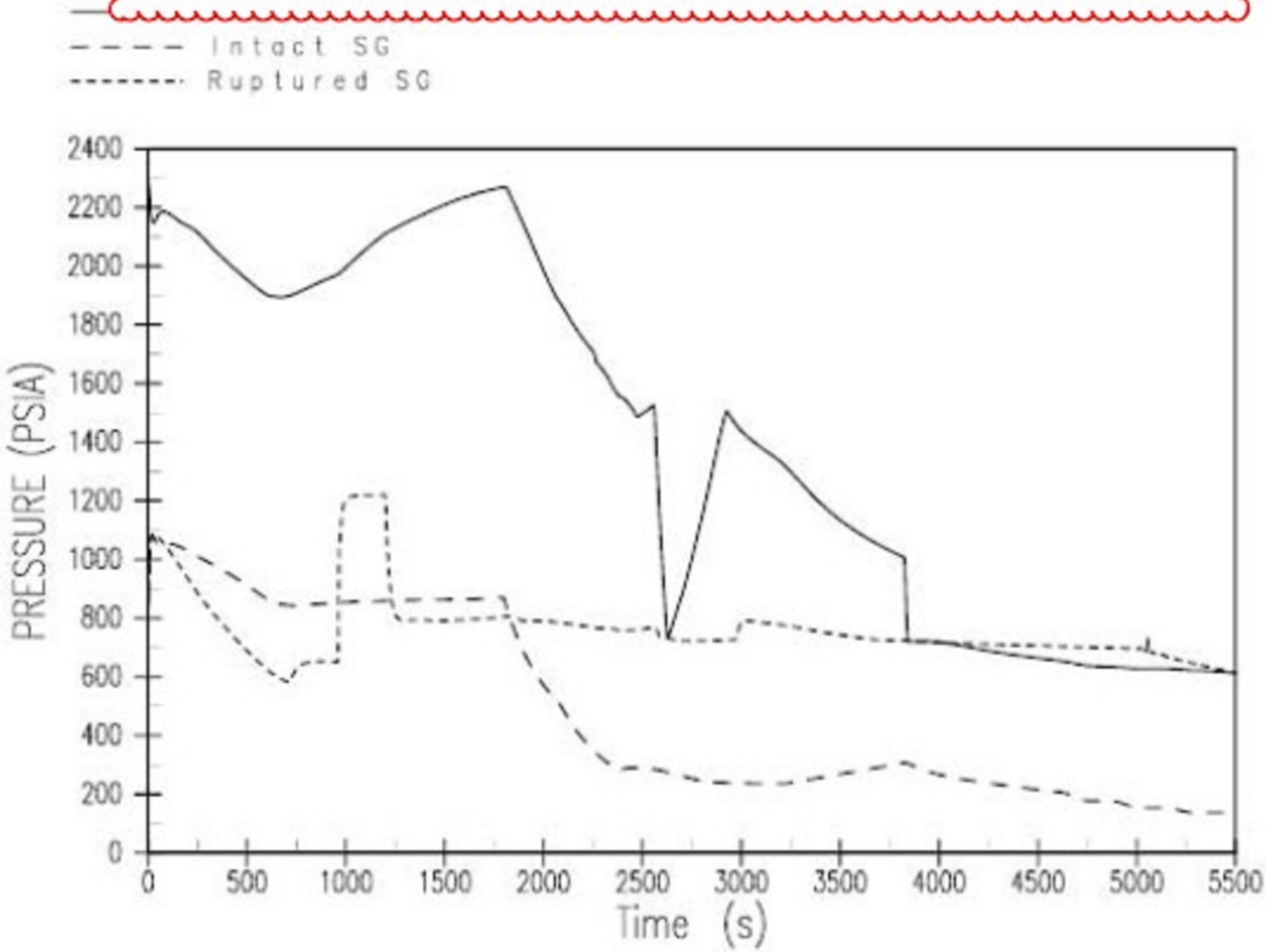
FIGURE 15.6-3P

FEEDWATER FLOW RATE (INTACT GENERATORS) TRANSIENT FOR STEAM GENERATOR TUBE RUPTURE EVENT

REV. 16 5/11


Intact Steam Generators (Total)
4000
3000
2000
1000
0
FW FLOW RATE (LBM/SEC)
0
1000
2000
3000
4000
5000
Time (s)

Replace with graph on Tab A from spreadsheet "Overfill_FSAR_Plots.xlsx"

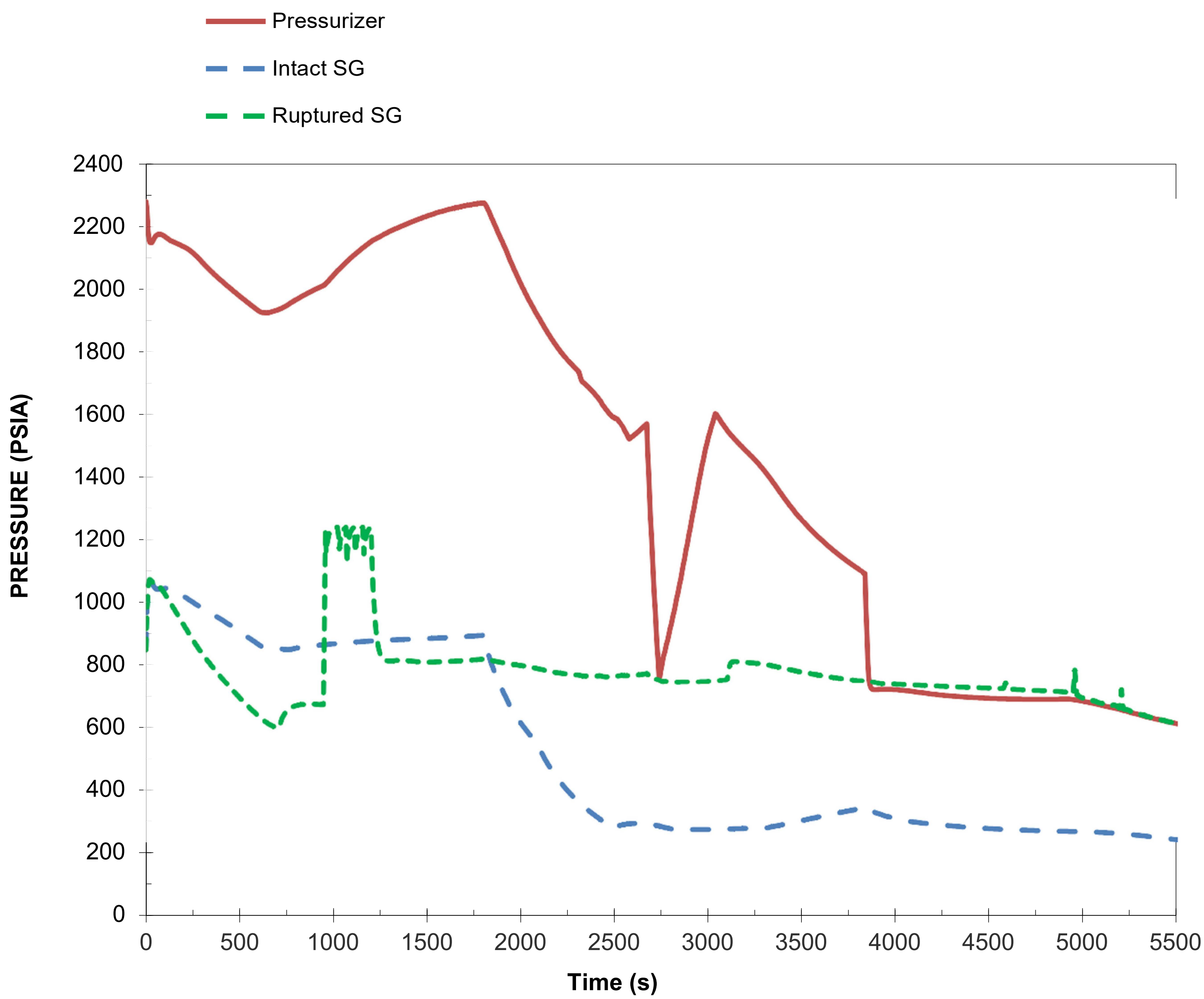


~~**Note:**~~

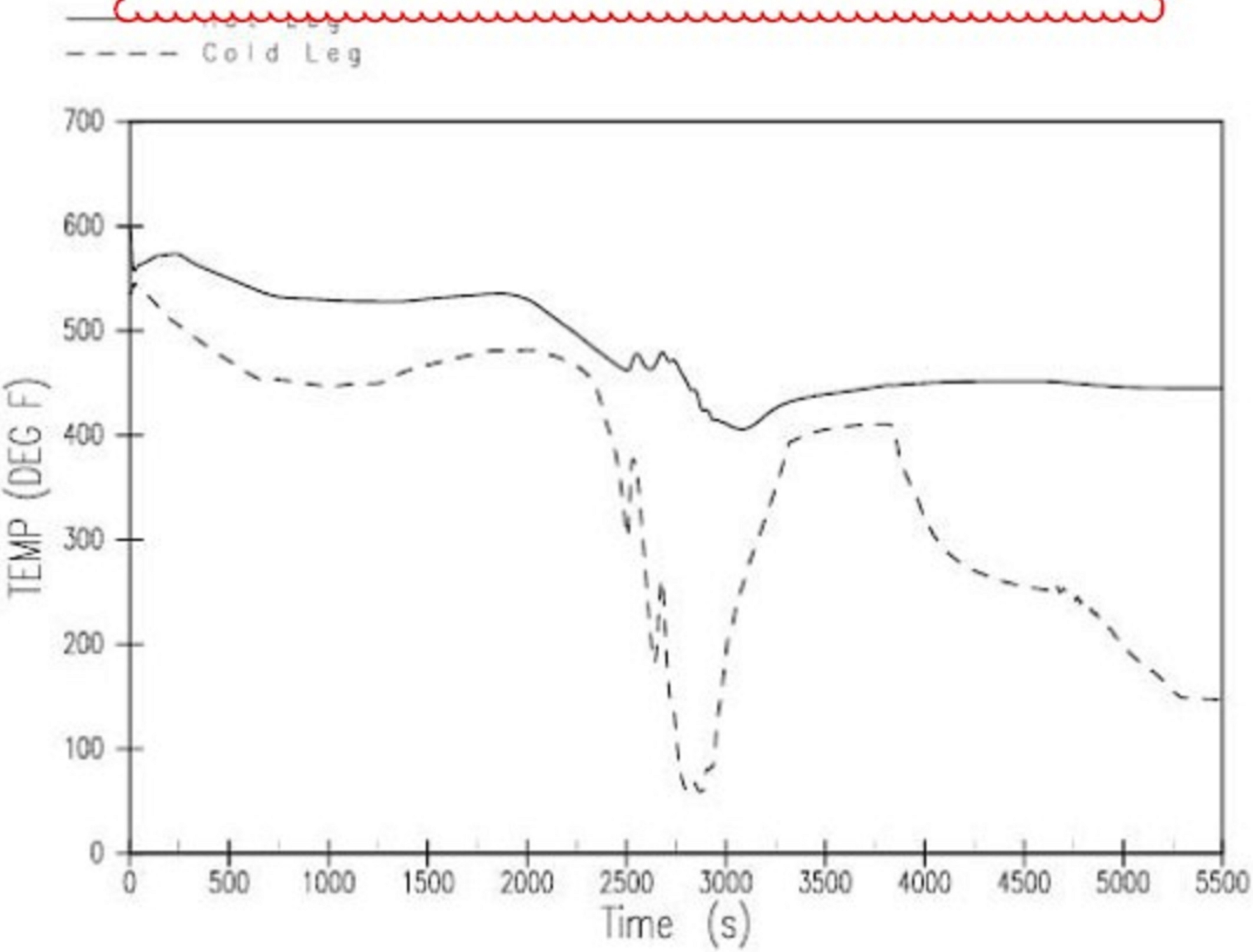
~~The thermal-hydraulic m~~ delete Note ~~figure is based on an analysis th ... SDs on the intact steam generators ... pport the RCS rapid cooldown. Th ... th respect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.~~

~~Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.~~

CALLAWAY PLANT
FIGURE 15.6-33A **PRESSURIZER AND STEAM GENERATOR (RUPTURED AND INTACT GENERATORS) PRESSURE TRANSIENTS FOR SGTR EVENT WITH OVERFILL** **REV. 16 6/11**


Pressurizer
Intact SG
Ruptured SG
PRESSURE (PSIA)
2400
2200
2000
1800
1600
1400
1200
1000
800
600
400
200
0
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000
5500
Time (s)

Replace with graph on Tab B from spreadsheet "Overfill_FSAR_Plots.xlsx"



Note:

The thermal-hydraulic [illegible] is based on an analysis [illegible] the intact steam generat[illegible] the RCS rapid cooldown. [illegible]pact to the thermal-hydra[illegible] the potential effects of [illegible]

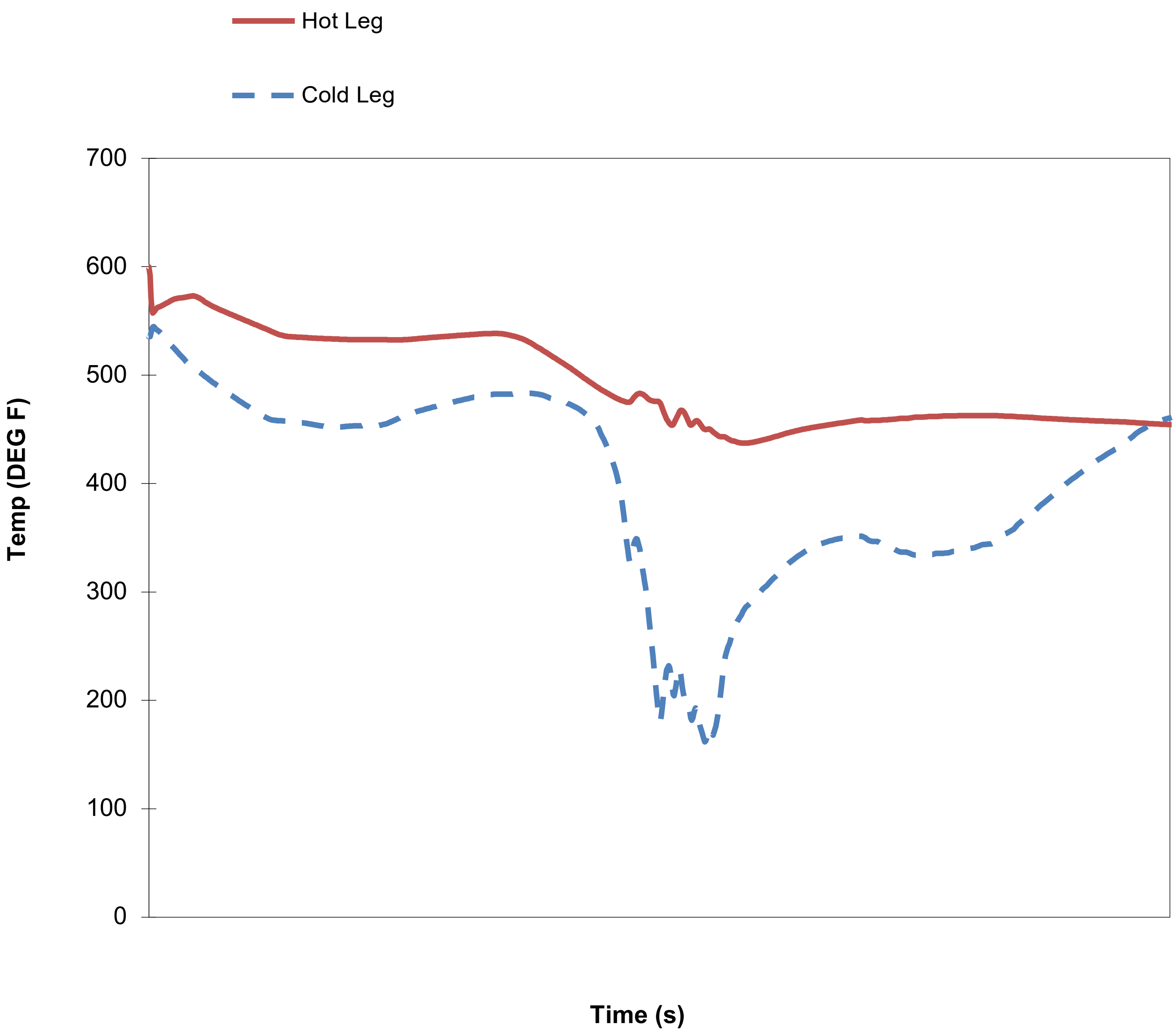
Delete Note

Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.

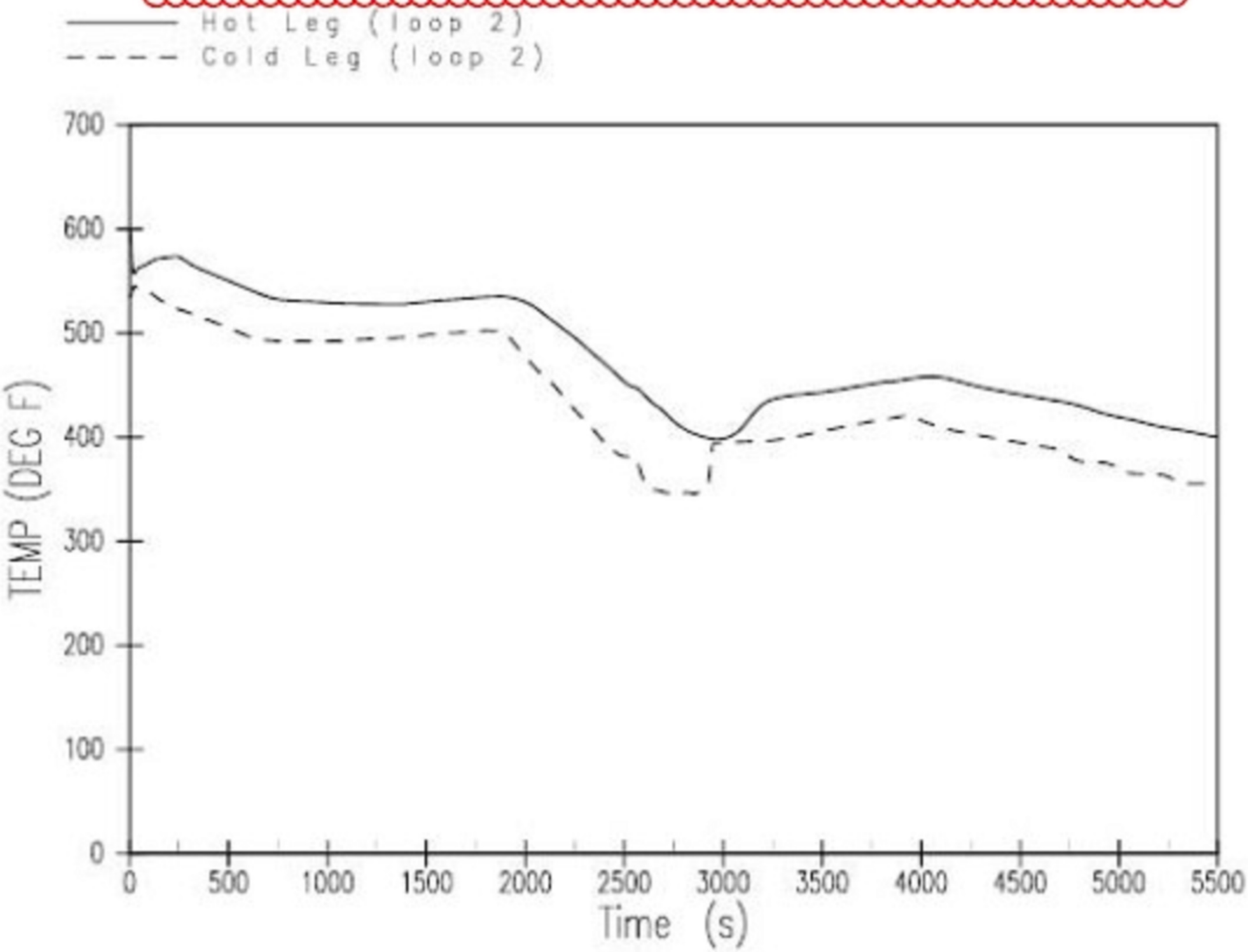
CALLAWAY PLANT

FIGURE 15.6-33B

REACTOR COOLANT SYSTEM TEMPERATURE (RUPTURED LOOP) TRANSIENT FOR SGTR EVENT WITH OVERFILL
REV. 16 6/11


Hot Leg
Cold Leg
700
600
500
400
300
200
100
0
Temp (DEG F)
Time (s)

Replace with graph on Tab C from spreadsheet "Overfill_FSAR_Plots.xlsx"



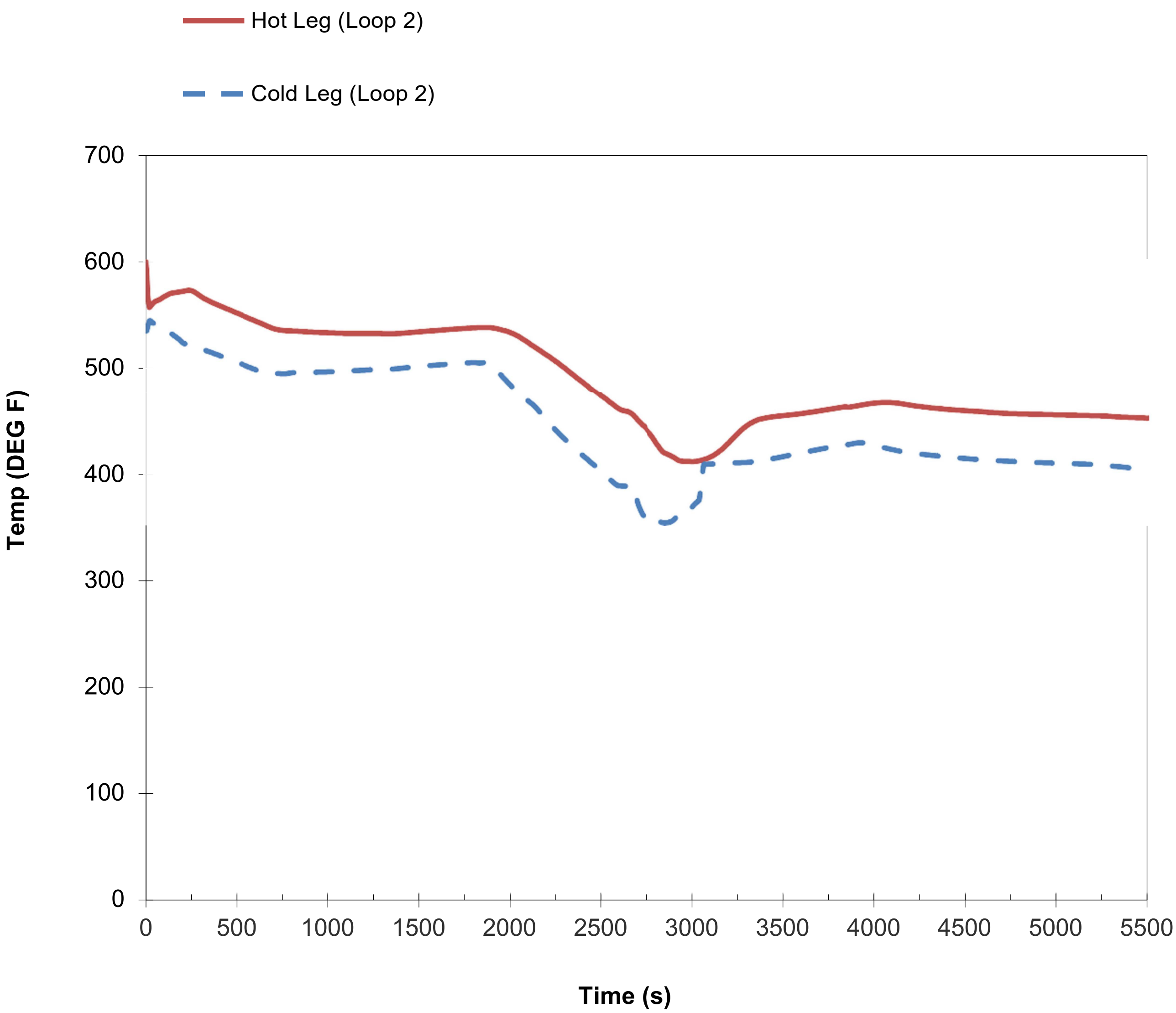
Note:

The thermal-hydraulic model presented in this figure is based on an analysis that on the intact steam generators to the RCS rapid cooldown. This respect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.

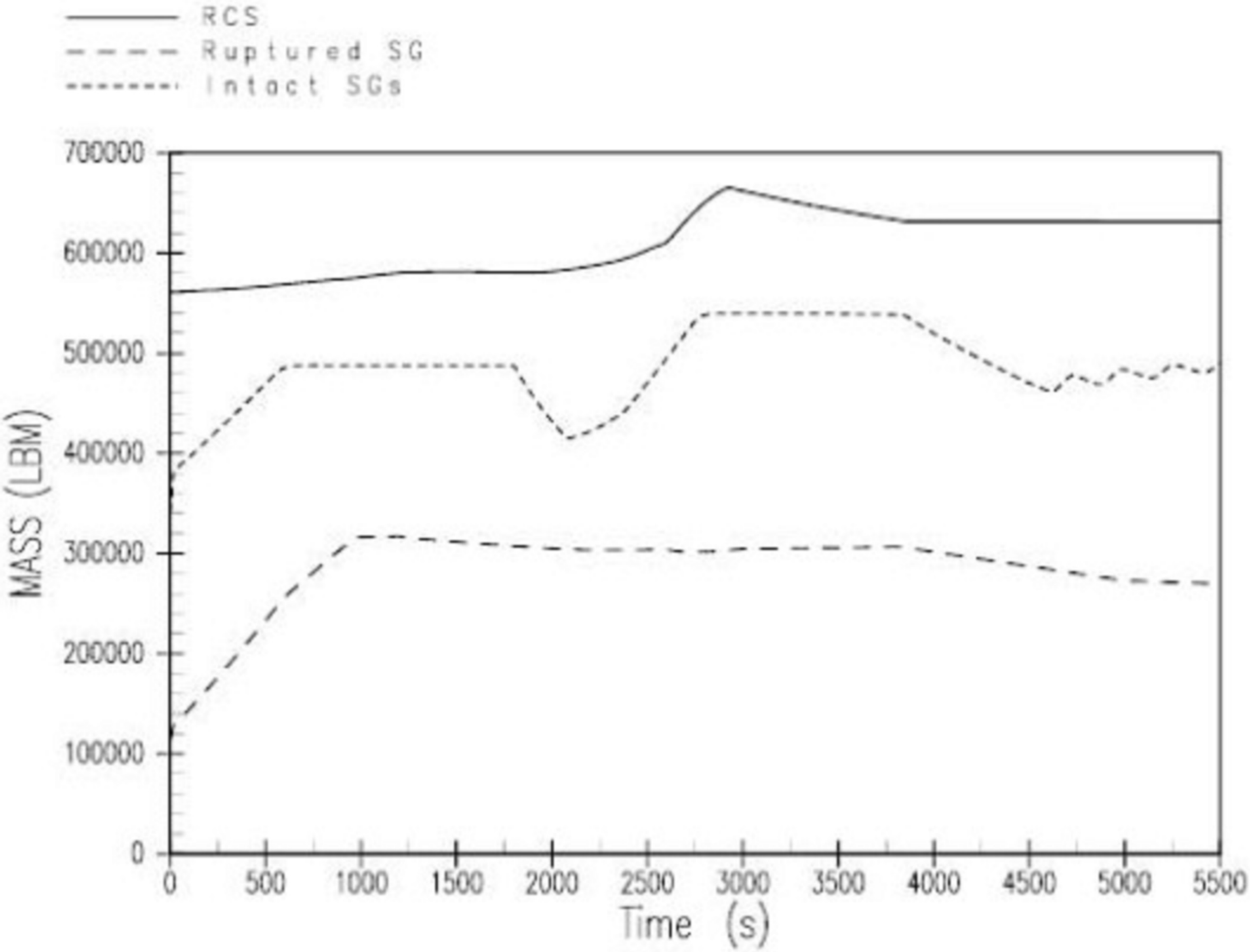
Delete Note

Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.

CALLAWAY PLANT
FIGURE 15.6-33C REACTOR COOLANT SYSTEM TEMPERATURE (INTACT LOOPS) TRANSIENT FOR SGTR EVENT WITH OVERFILL REV. 16 6/11


Hot Leg (Loop 2)
Cold Leg (Loop 2)
Temp (DEG F)
700
600
500
400
300
200
100
0
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000
5500
Time (s)

Replace with graph on Tab D from spreadsheet "Overfill_FSAR_Plots.xlsx"



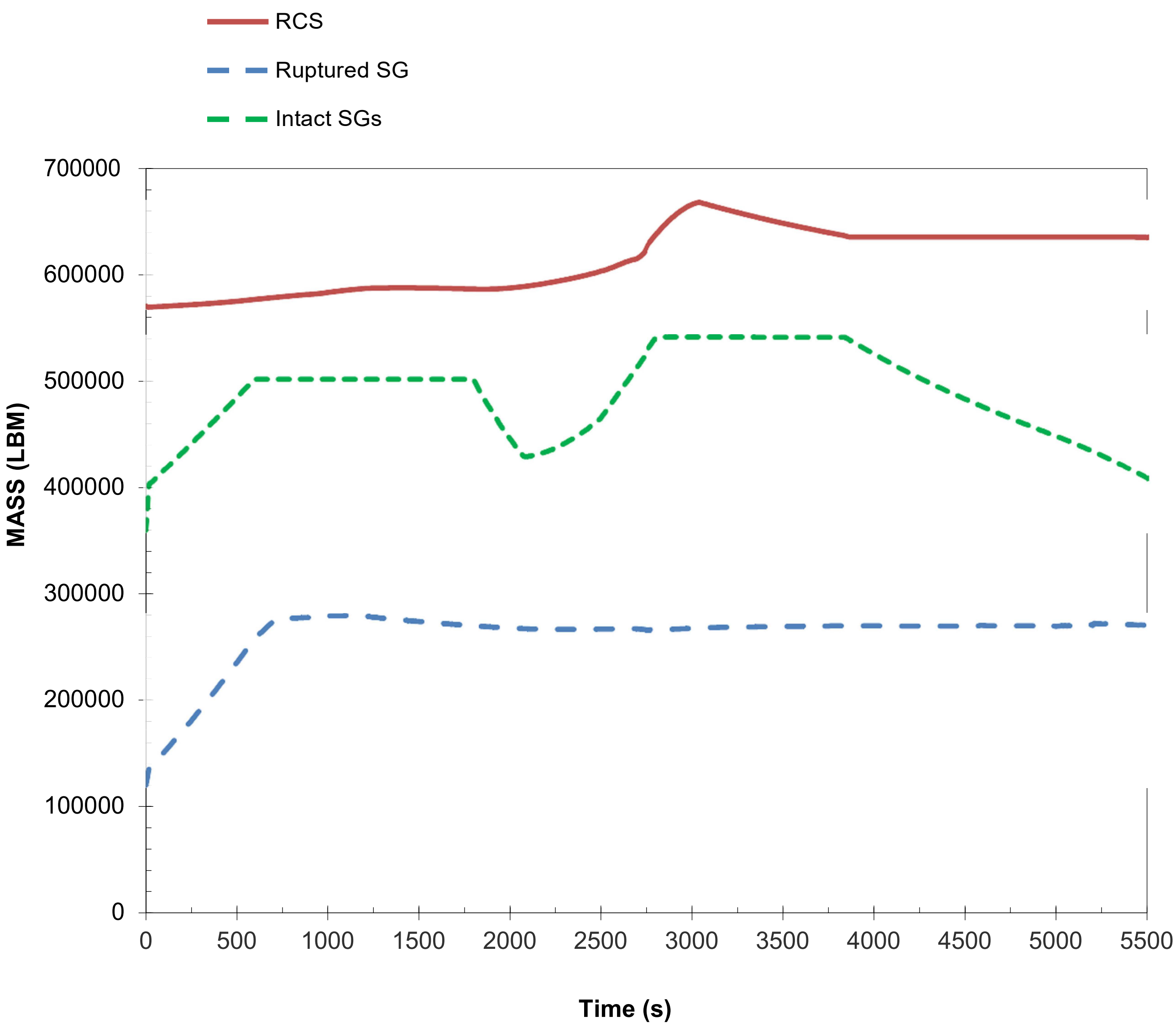
Note:

The thermal-hydraulic model presented in this figure is based on an analysis that credits all three ASDs on the intact steam genera[illegible]ort the RCS rapid cooldown[illegible] respect to the thermal-hydr[illegible]and the potential effects of the associated transient.

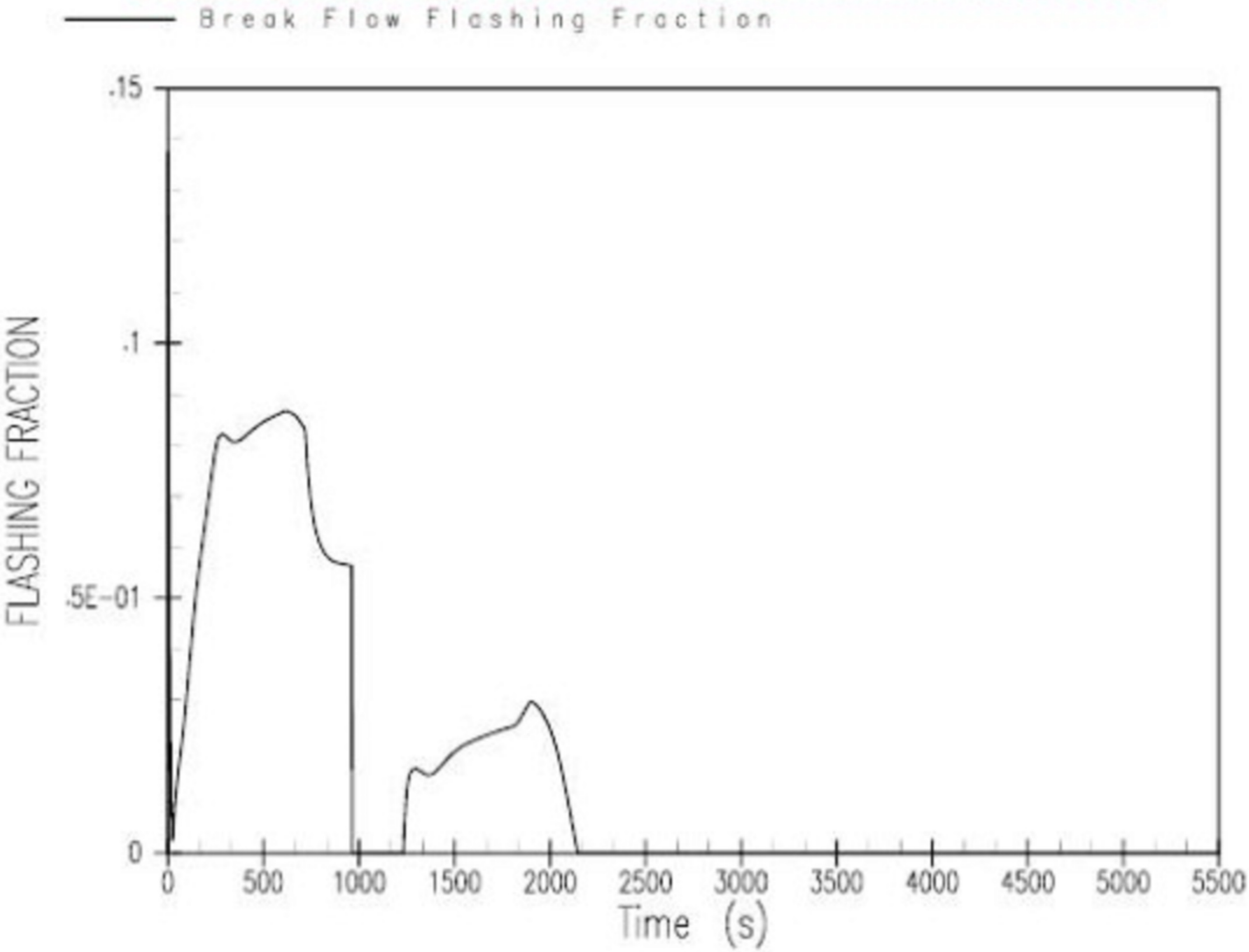
Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.

Delete Note

CALLAWAY PLANT
FIGURE 15.6-33D REACTOR COOLANT SYSTEM AND STEAM GENERATOR (RUPTURED AND INTACT GENERATORS) WATER MASS TRANSIENT FOR SGTR EVENT WITH OVERFILL REV. 16 5/11


RCS
Ruptured SG
Intact SGs
700000
600000
500000
400000
300000
200000
100000
0
MASS (LBM)
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000
5500
Time (s)

Replace with graph on Tab E from spreadsheet "Overfill_FSAR_Plots.xlsx"



~~Note:~~

~~The thermal-hydraulic m[illegible] based on an analysis th[illegible] intact steam generators [illegible] RCS rapid cooldown. T[illegible] to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.~~

Delete Note

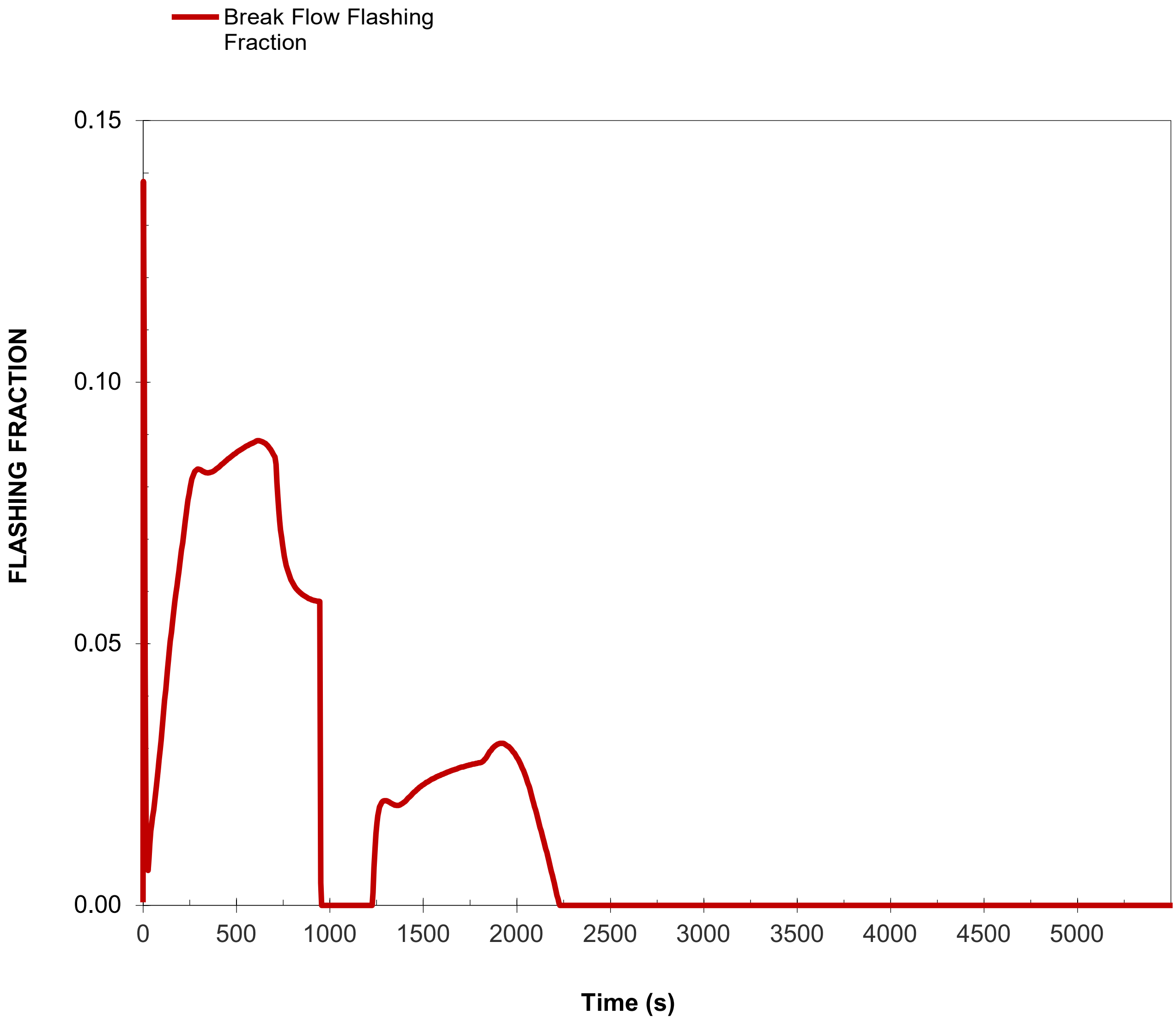
~~Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.~~

CALLAWAY PLANT

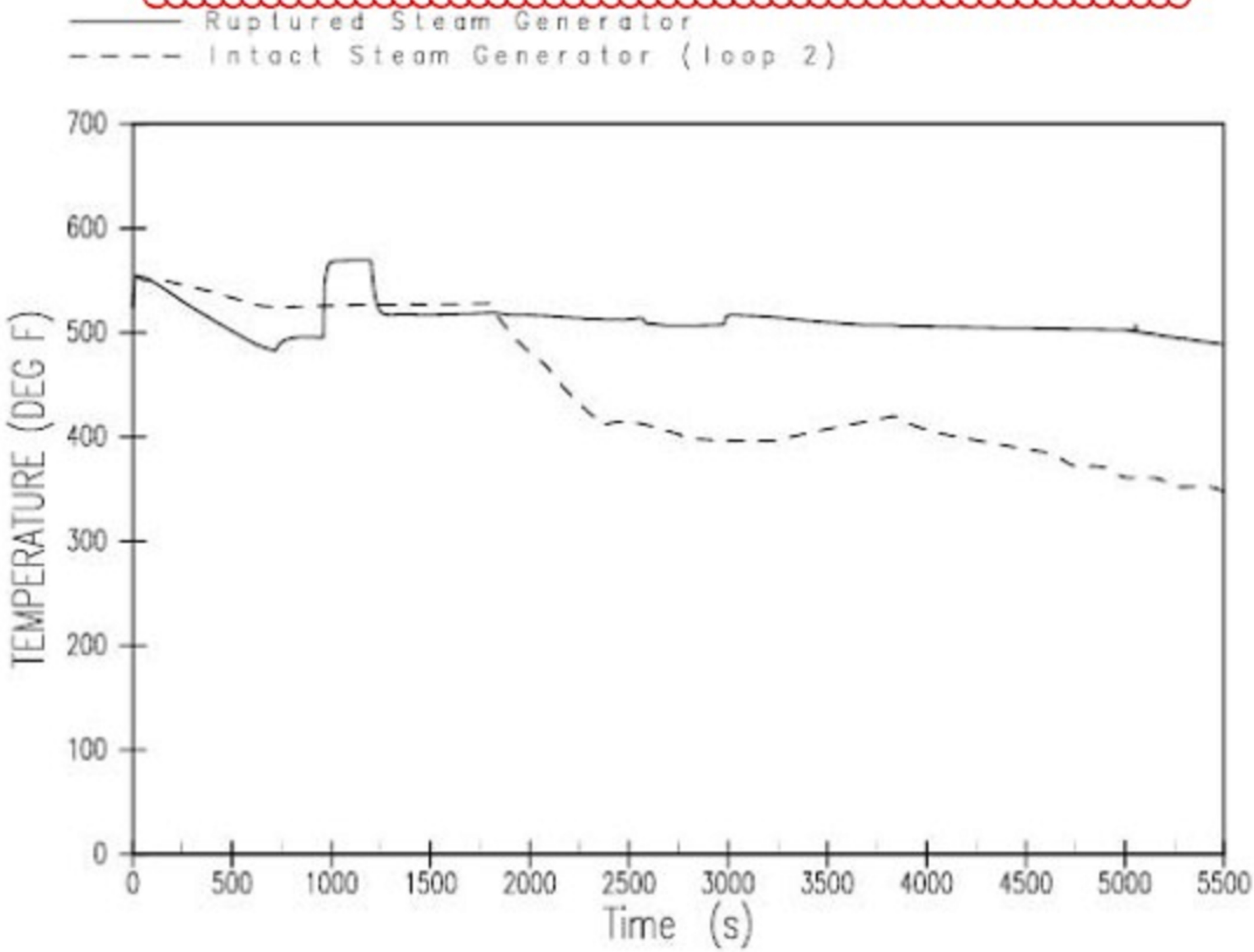
FIGURE 15.6-33E

RUPTURED STEAM GENERATOR BREAK FLOW FLASHING FRACTION TRANSIENT FOR SGTR EVENT WITH OVERFILL

REV. 16 6/11


Break Flow Flashing Fraction
0.15
0.10
0.05
0.00
FLASHING FRACTION
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000
Time (s)

Replace with graph on Tab F from spreadsheet "Overfill_FSAR_Plots.xlsx"



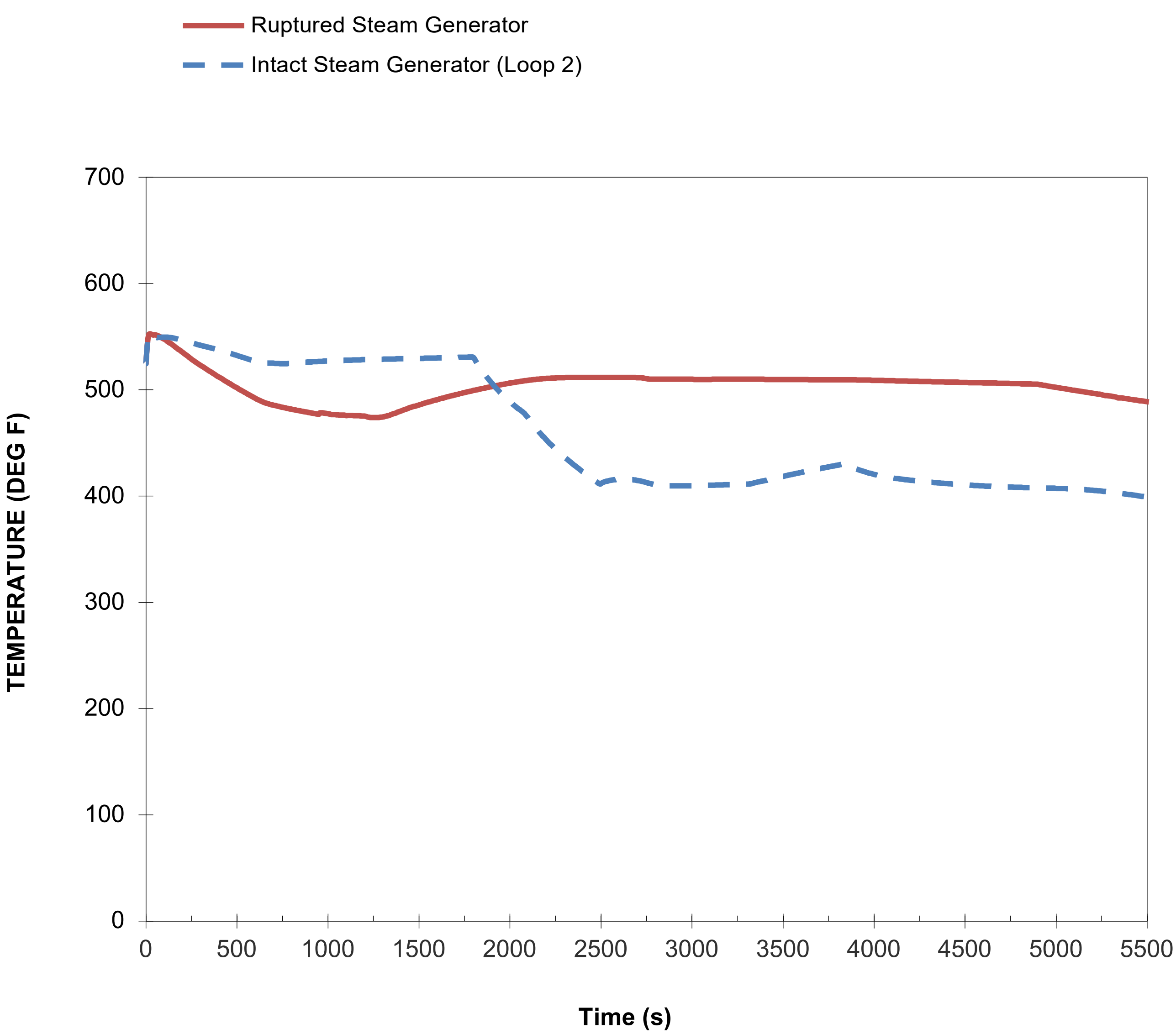
~~Note:~~

~~The thermal-hydraulic model presented in this figure is based on an analysis [illegible] ASDs on the intact steam generat[illegible] pport the RCS rapid cooldown. [illegible] th respect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.~~

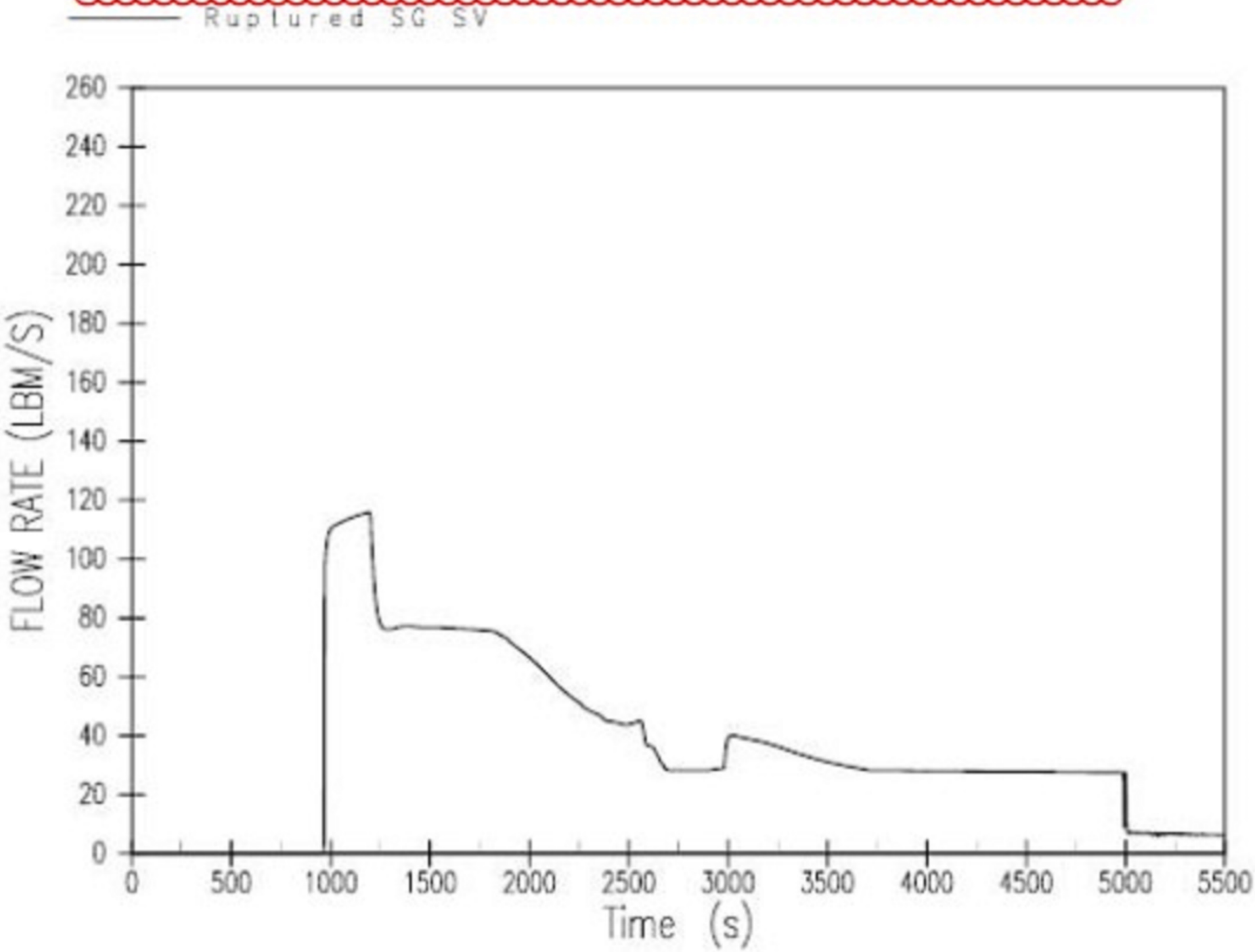
Delete Note

~~Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.~~

CALLAWAY PLANT
FIGURE 15.6-33F STEAM GENERATOR TEMPERATURE (RUPTURED AND INTACT GENERATORS) TRANSIENT FOR SGTR EVENT WITH OVERFILL REV. 16 6/11


Ruptured Steam Generator
Intact Steam Generator (Loop 2)
TEMPERATURE (DEG F)
0
100
200
300
400
500
600
700
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000
5500
Time (s)

Replace with graph on Tab G from spreadsheet "Overfill_FSAR_Plots.xlsx"



~~Note:~~

~~The thermal-hydraulic model presented in this figure is based on an analysis th[illegible] the intact steam generators [illegible]he RCS rapid cooldown. Th[illegible]pact to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.~~

Delete Note

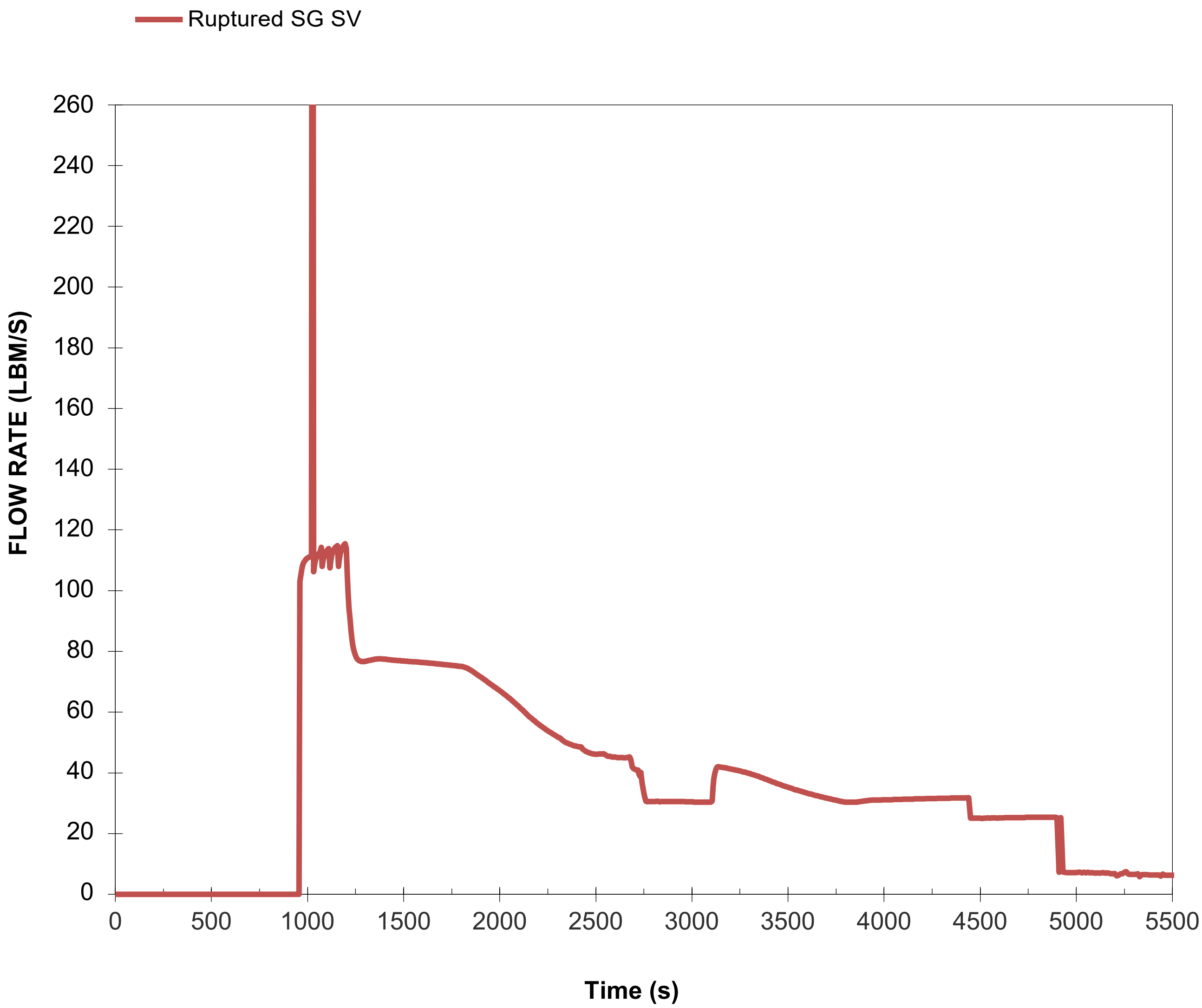
~~Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.~~

CALLAWAY PLANT

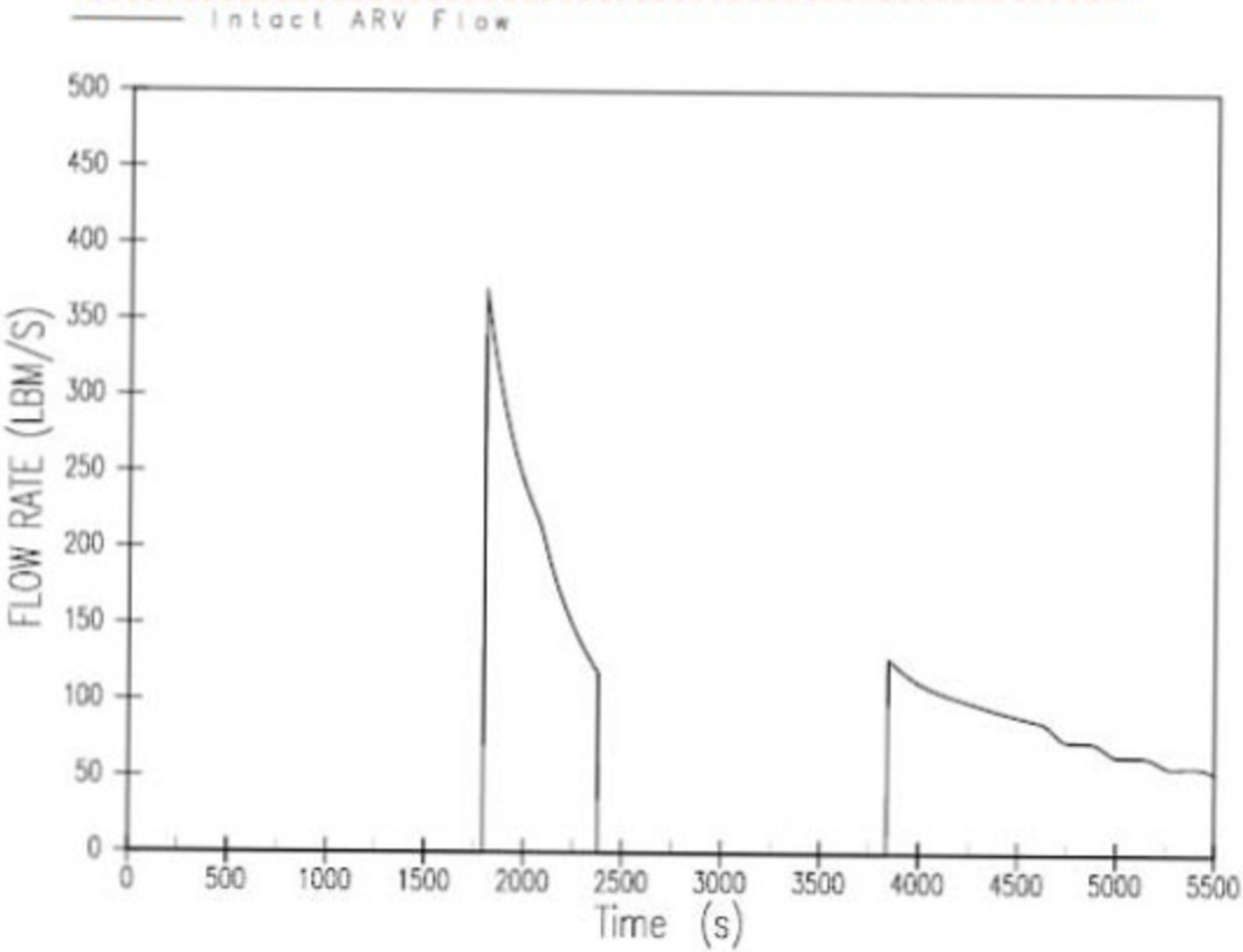
FIGURE 15.6-33G

STEAM GENERATOR ATMOSPHERIC RELEASE FLOW RATE (RUPTURED GENERATOR) TRANSIENT FOR SGTR EVENT WITH OVERFILL

REV. 16 6/11


Ruptured SG SV
FLOW RATE (LBM/S)
260
240
220
200
180
160
140
120
100
80
60
40
20
0
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000
5500
Time (s)

Replace with graph on Tab H from spreadsheet "Overfill_FSAR_Plots.xlsx"



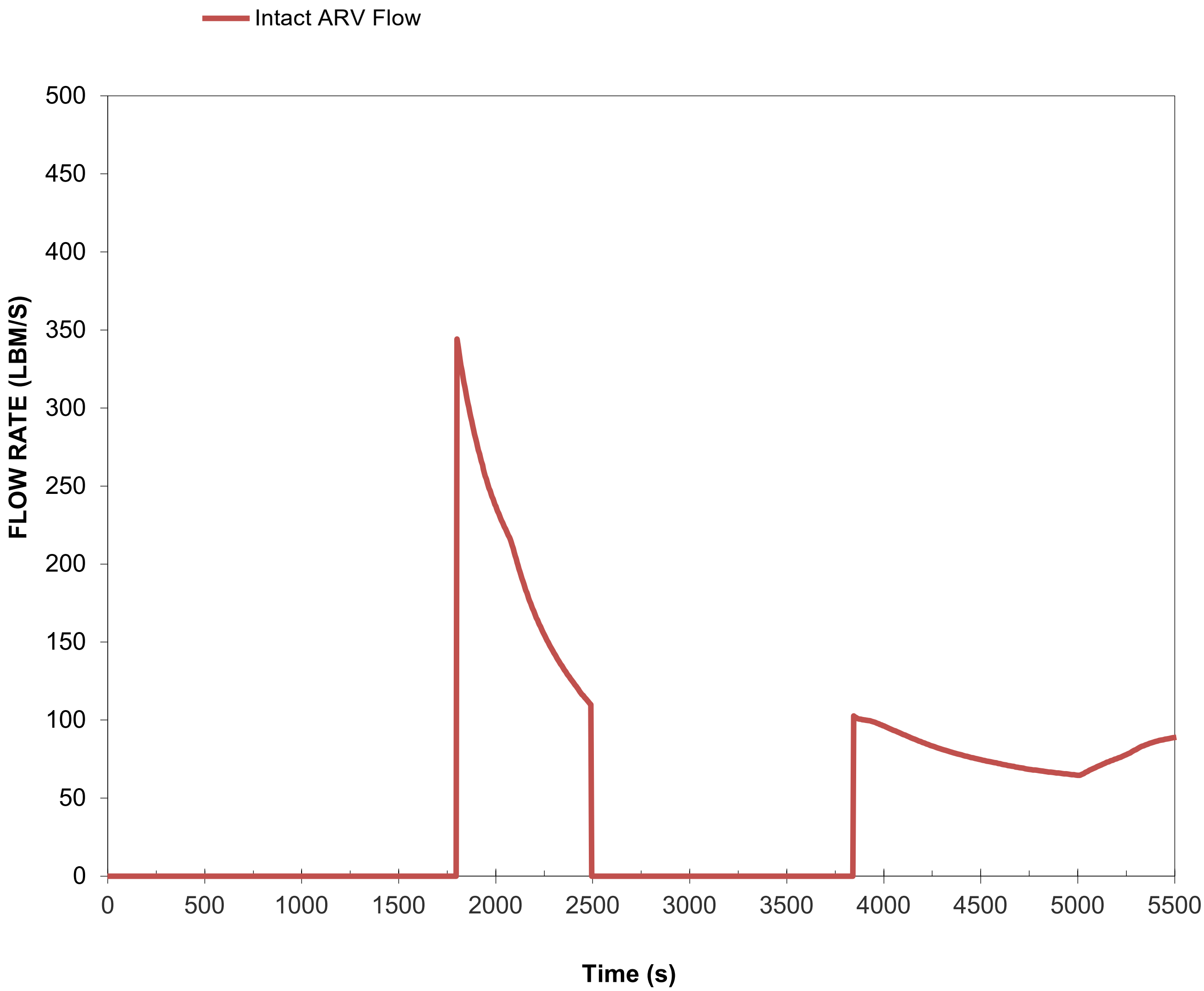
~~Note:~~

~~The thermal-hydraulic model presented in this figure is based on an analy[illegible] on the intact steam gener[illegible] the RCS rapid cooldow[illegible] respect to the thermal-hyd[illegible] the potential effects of the associated transient.~~

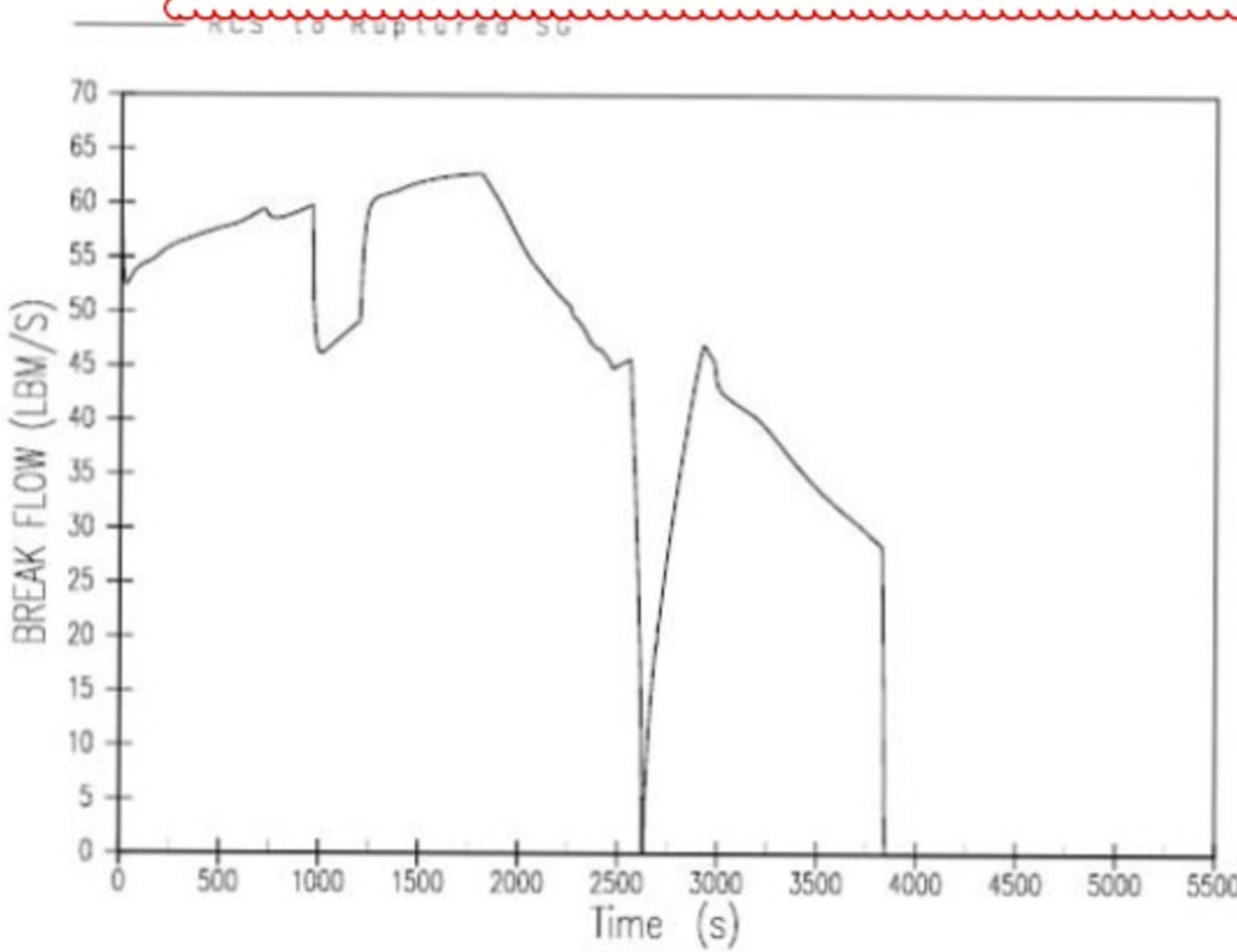
Delete Note

~~Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.~~

CALLAWAY PLANT
FIGURE 15.6-33H STEAM GENERATOR ATMOSPHERIC RELEASE FLOW RATE (INTACT GENERATORS) TRANSIENT FOR SGTR EVENT WITH OVERFILL REV. 16 6/11


Intact ARV Flow
500
450
400
350
300
250
200
150
100
50
0
FLOW RATE (LBM/S)
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000
5500
Time (s)

Replace with graph on Tab I (Indigo) from spreadsheet "Overfill_FSAR_Plots.xlsx"



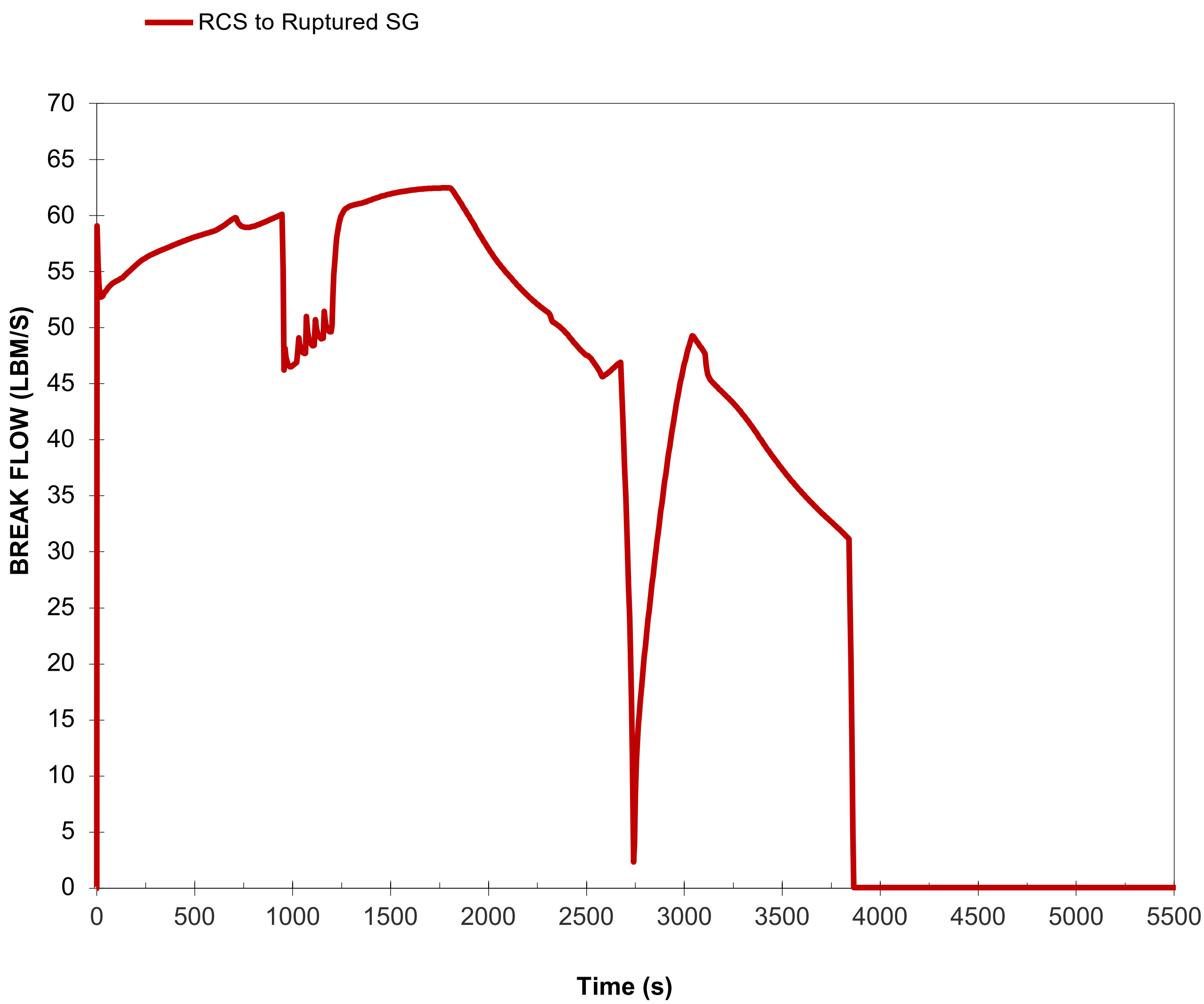
Note:

The thermal [illegible] this figure is based on [illegible] ASDs on the intact steam [illegible] support the RCS rapid [illegible] with respect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.

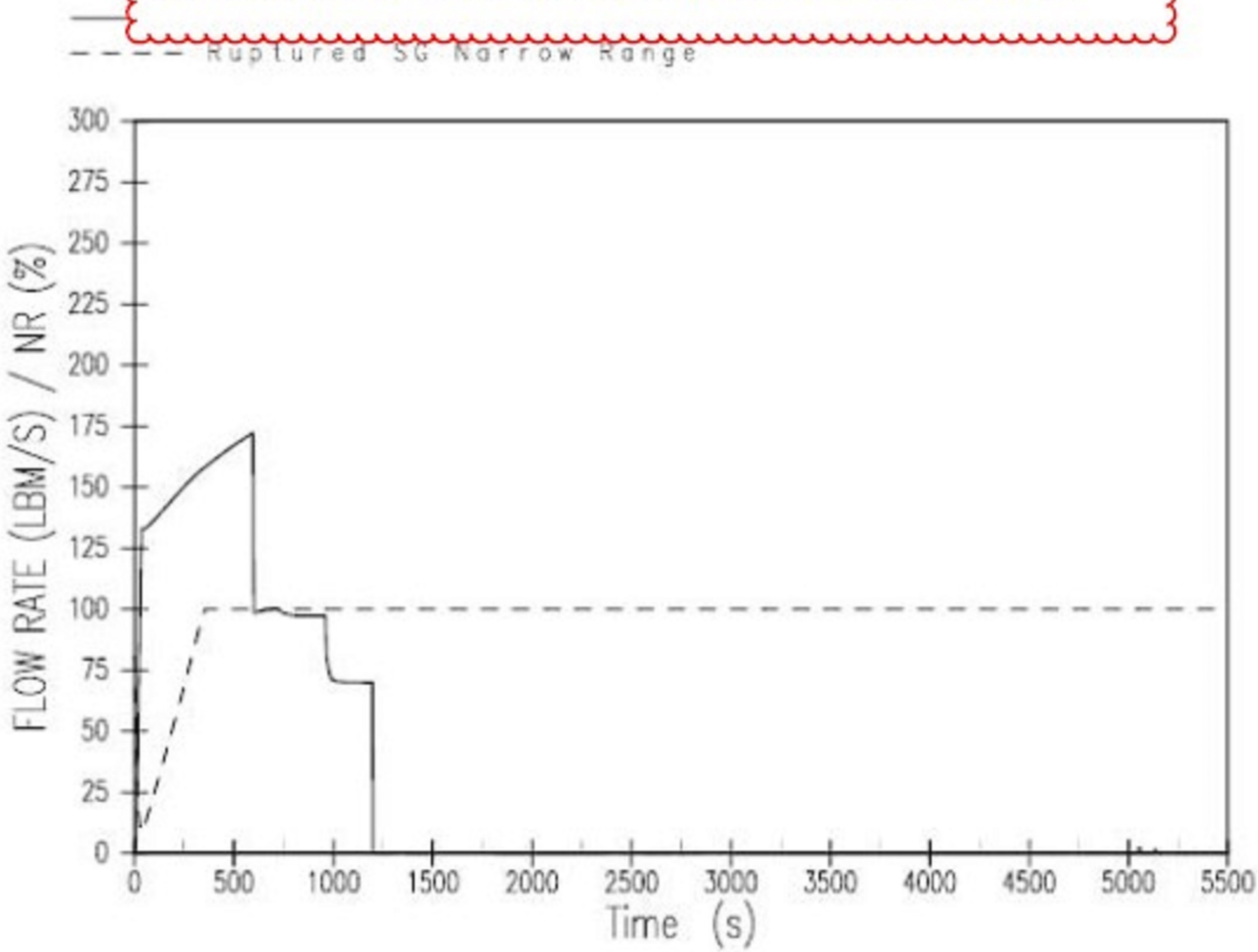
Delete Note

Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.

CALLAWAY PLANT
FIGURE 15.6-33I RUPTURED STEAM GENERATOR BREAK FLOW RATE TRANSIENT FOR SGTR EVENT WITH OVERFILL REV. 18 5/11


RCS to Ruptured SG
BREAK FLOW (LBM/S)
0
5
10
15
20
25
30
35
40
45
50
55
60
65
70
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000
5500
Time (s)

Replace with graph on Tab J from spreadsheet "Overfill_FSAR_Plots.xlsx"



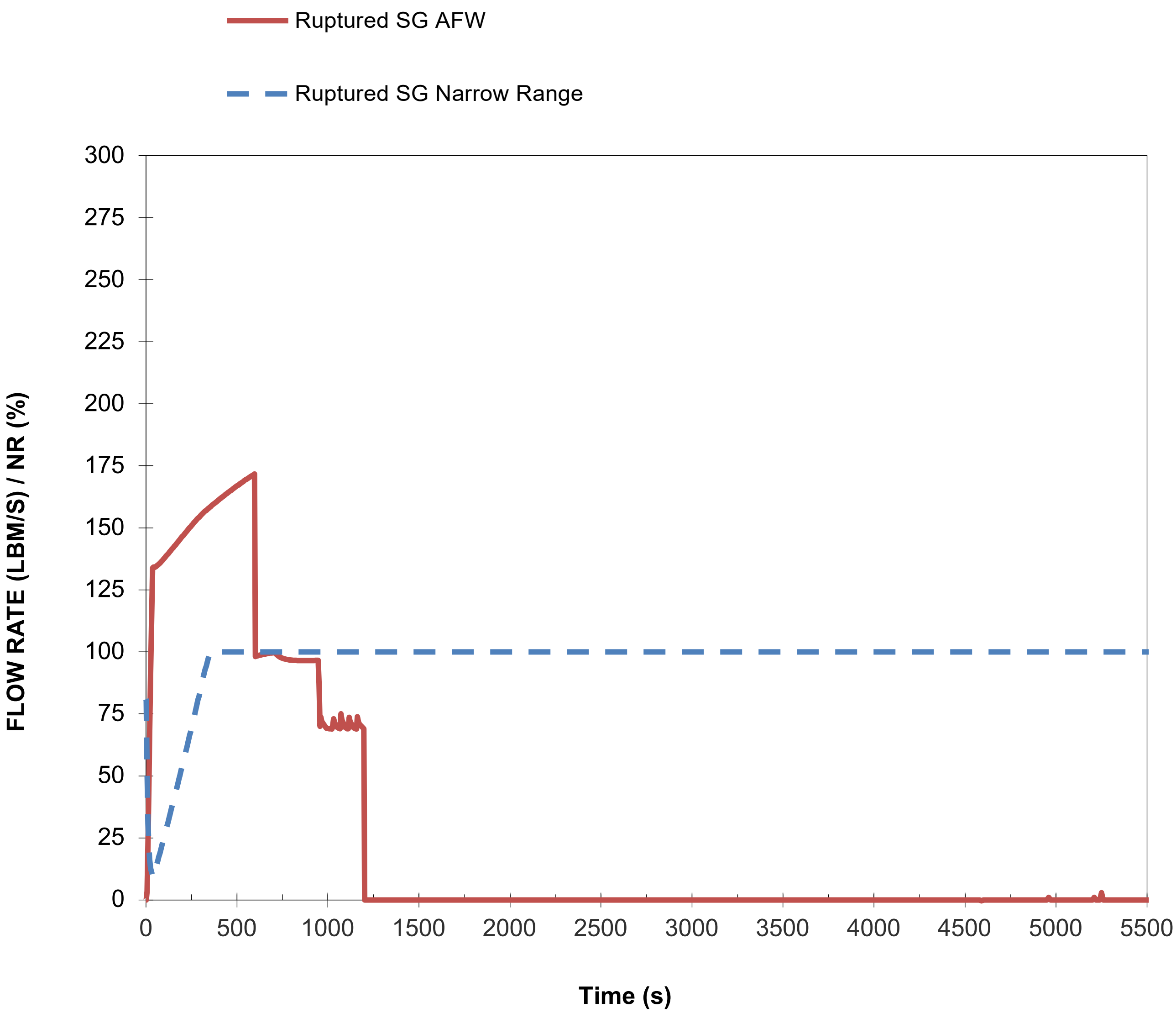
~~**Note:**~~

~~**The thermal-hydraulic** [illegible] **is based on an analysis** [illegible] **the intact steam generator** [illegible] **the RCS rapid cooldown.** [illegible] **respect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.**~~

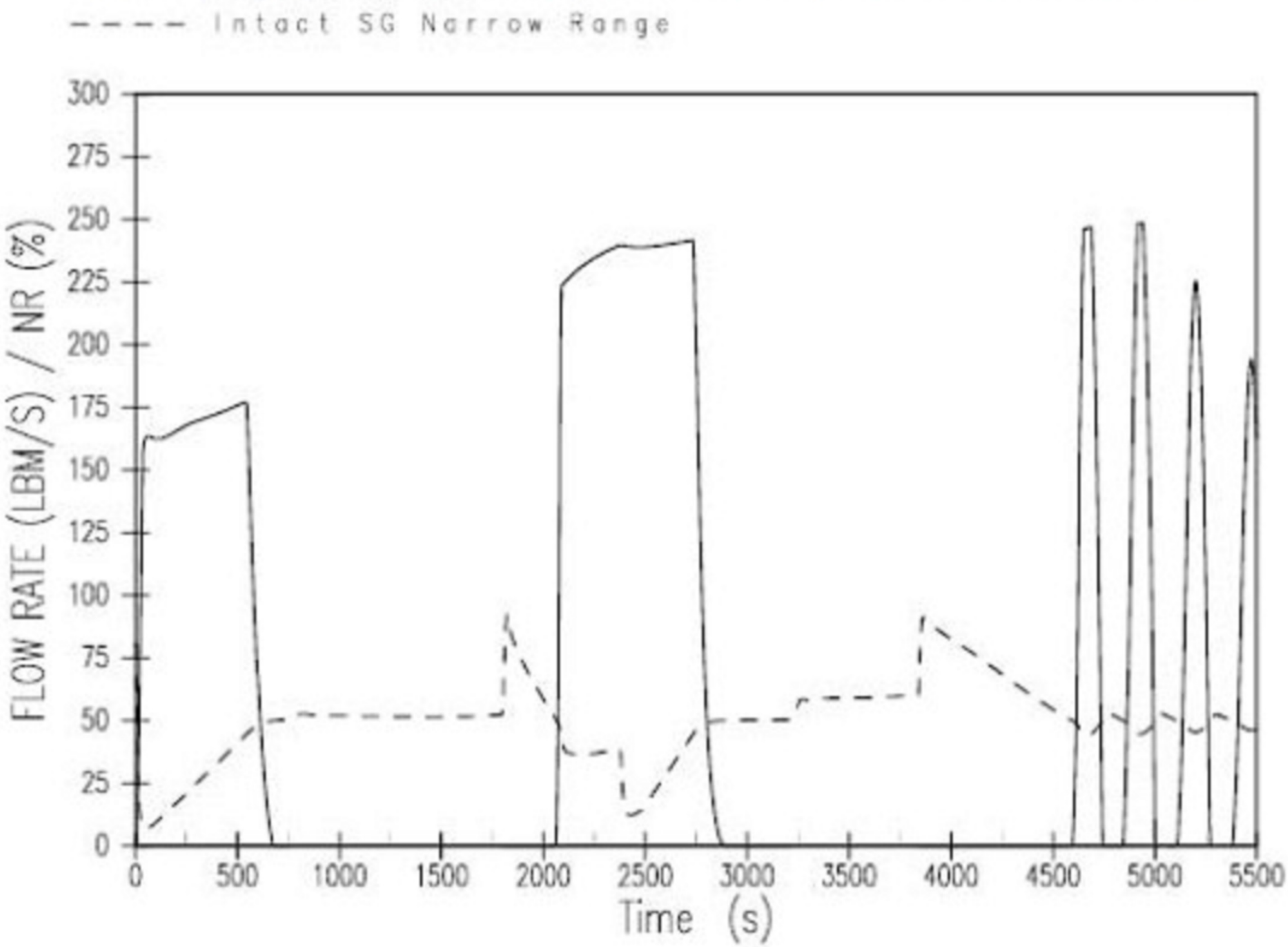
~~**Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.**~~

Delete Note

CALLAWAY PLANT
FIGURE 15.6-33J   **AUXILIARY FEEDWATER FLOW RATE AND NARROW RANGE LEVEL (RUPTURED GENERATOR) TRANSIENTS FOR SGTR EVENT WITH OVERFILL**   **REV. 18 5/11**


Ruptured SG AFW
Ruptured SG Narrow Range
FLOW RATE (LBM/S) / NR (%)
300
275
250
225
200
175
150
125
100
75
50
25
0
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000
5500
Time (s)

Replace with graph on Tab K from spreadsheet "Overfill_FSAR_Plots.xlsx"

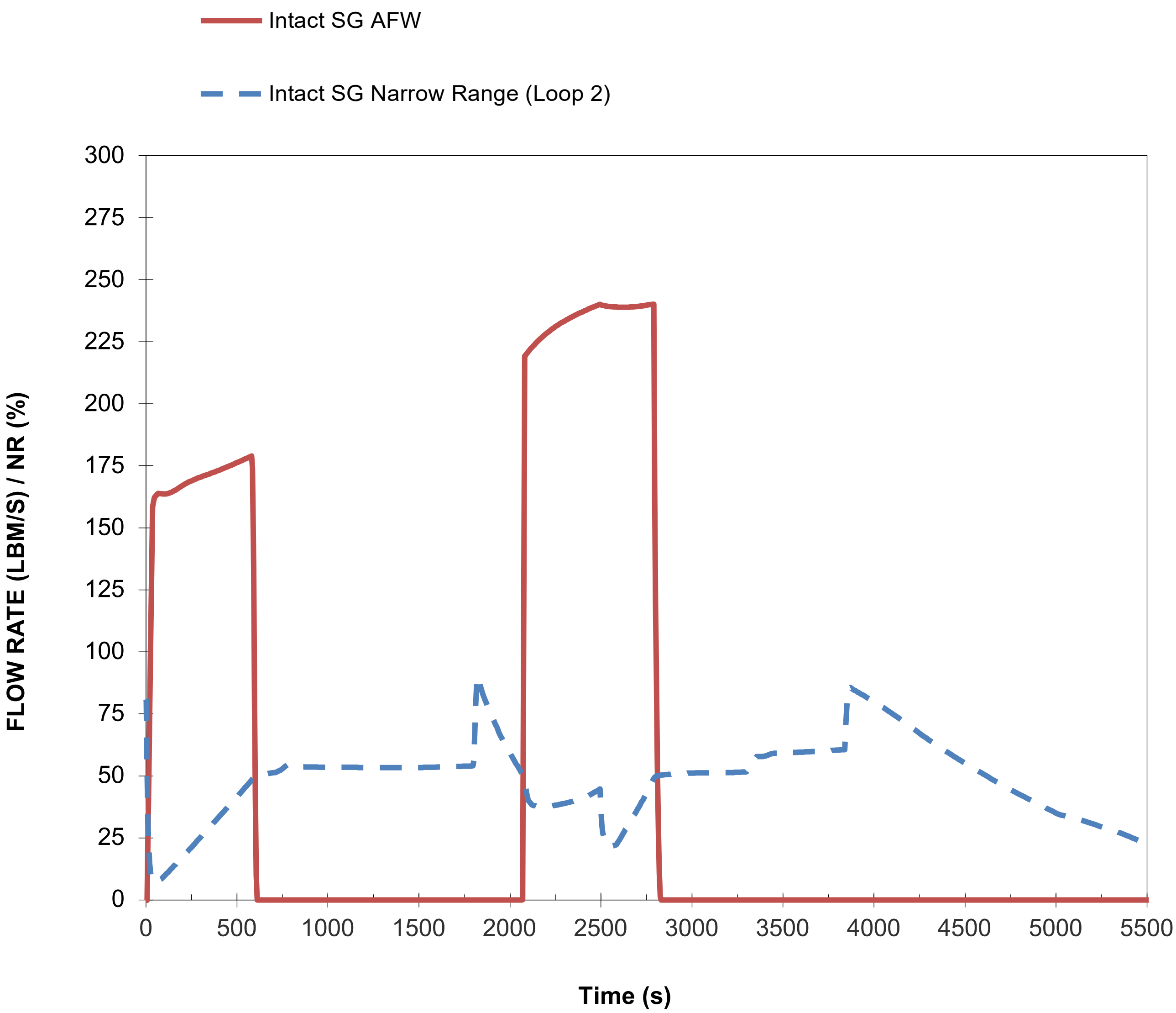


~~**Note:**~~

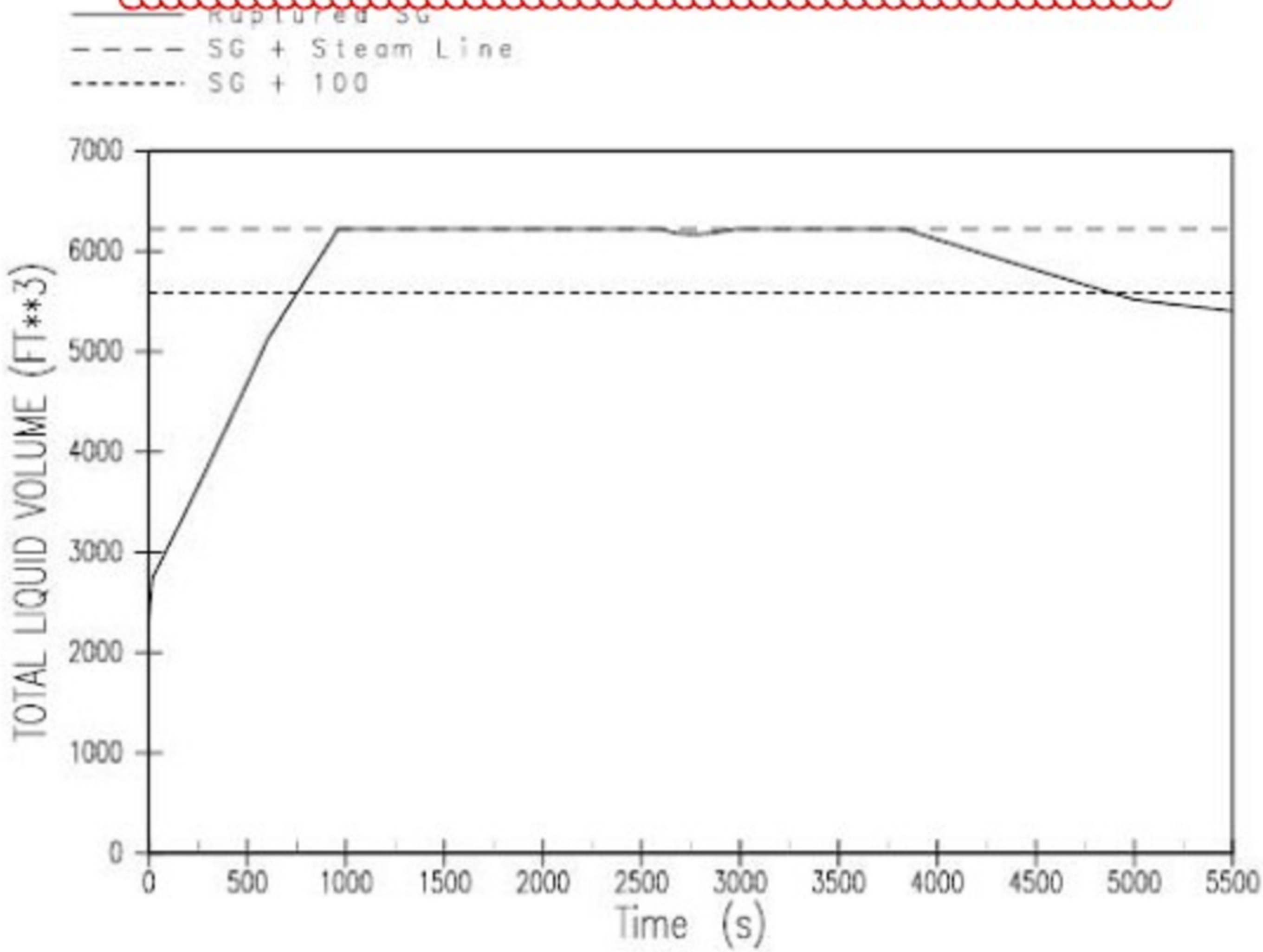
~~**The thermal-hydraulic model presented in this figure is based on an analysis**~~ Delete Note ~~**Ds on the intact steam generator**~~ ~~**port the RCS rapid cooldown. This is conservative with respect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.**~~

~~**Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.**~~

CALLAWAY PLANT
FIGURE 15.6-33K AUXILIARY FEEDWATER FLOW RATE AND NARROW RANGE LEVEL (INTACT GENERATORS) TRANSIENTS FOR SGTR EVENT WITH OVERFILL REV. 16 5/11


Intact SG AFW
Intact SG Narrow Range (Loop 2)
FLOW RATE (LBM/S) / NR (%)
300
275
250
225
200
175
150
125
100
75
50
25
0
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000
5500
Time (s)

Replace with graph on Tab L from spreadsheet "Overfill_FSAR_Plots.xlsx"



~~**Note:**~~

~~The thermal-hydrau[illegible] figure is based on an analys[illegible]Ds on the intact steam gener[illegible]port the RCS rapid cooldow[illegible]th respect to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.~~

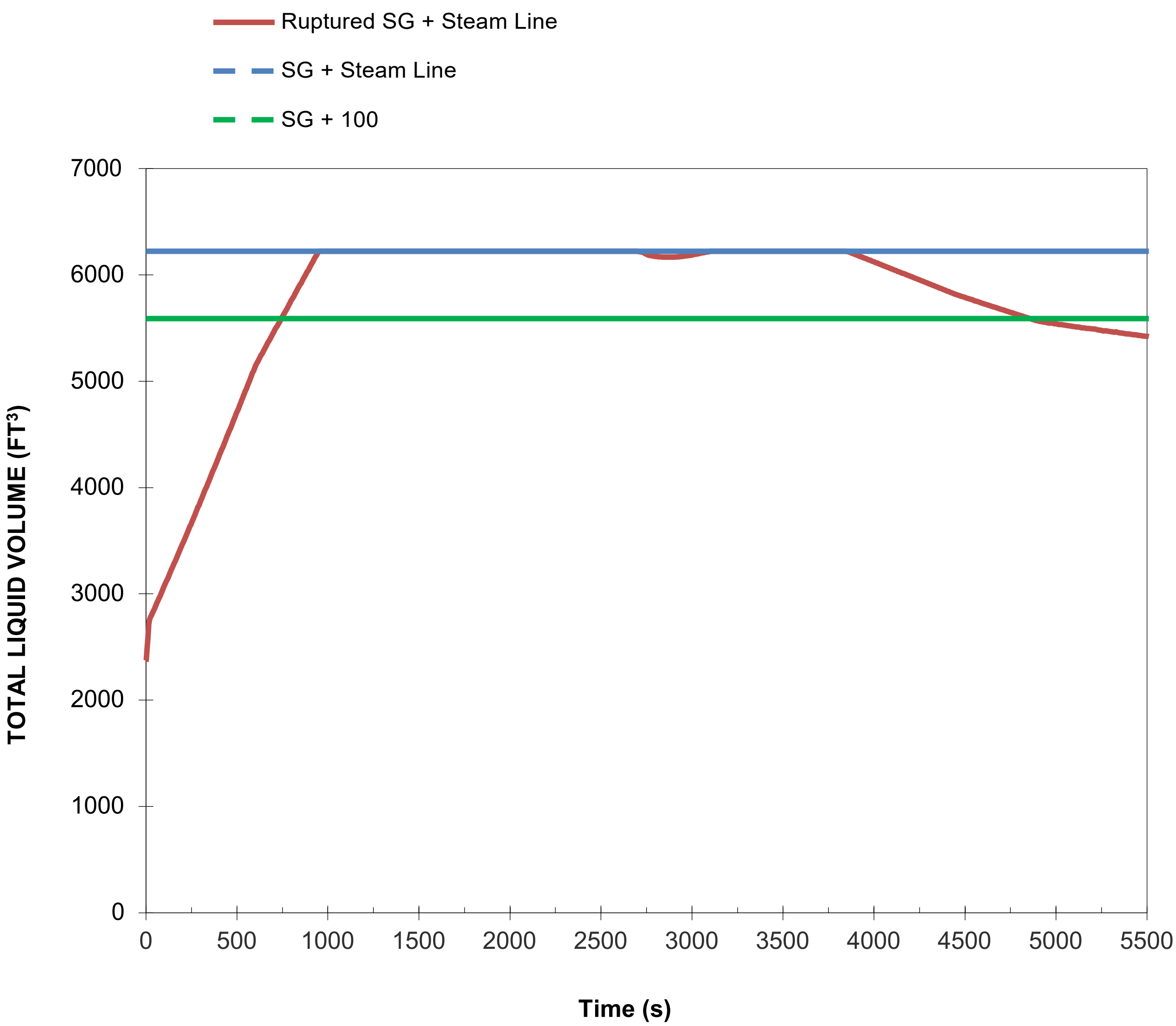
Delete Note

~~Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.~~

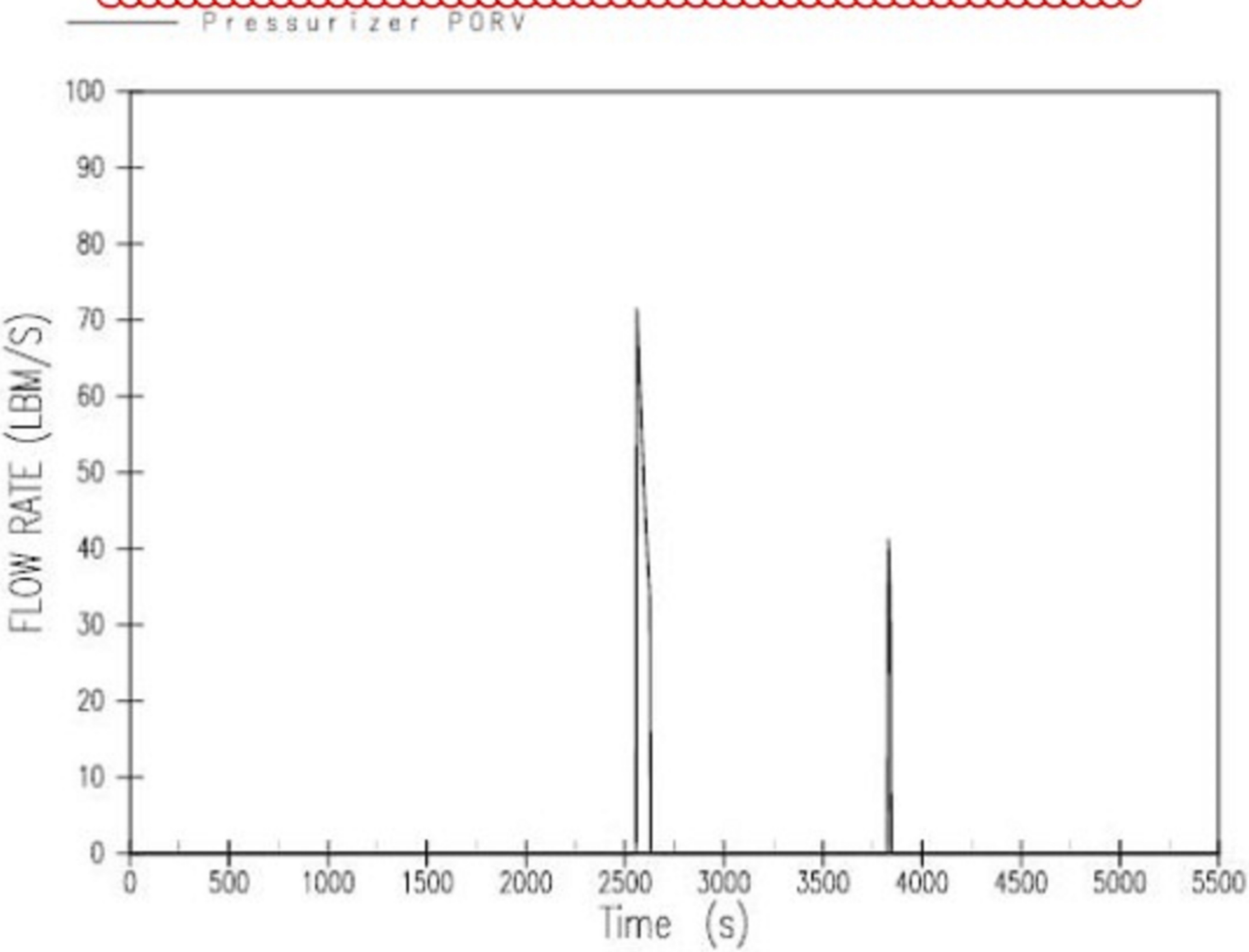
CALLAWAY PLANT

FIGURE 15.6-33L

RUPTURED STEAM GENERATOR LIQUID VOLUME TRANSIENT FOR SGTR EVENT WITH OVERFILL
REV. 16 6/11


Ruptured SG + Steam Line
SG + Steam Line
SG + 100
7000
6000
5000
4000
3000
2000
1000
0
TOTAL LIQUID VOLUME (FT³)
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000
5500
Time (s)

Replace with graph on Tab M from spreadsheet "Overfill_FSAR_Plots.xlsx"



~~**Note:**~~

~~The thermal-hydraulic m[illegible] is based on an analysis th[illegible] the intact steam generators [illegible] the RCS rapid cooldown. Th[illegible] pact to the thermal-hydraulic analysis of the SGTR and the potential effects of the associated transient.~~

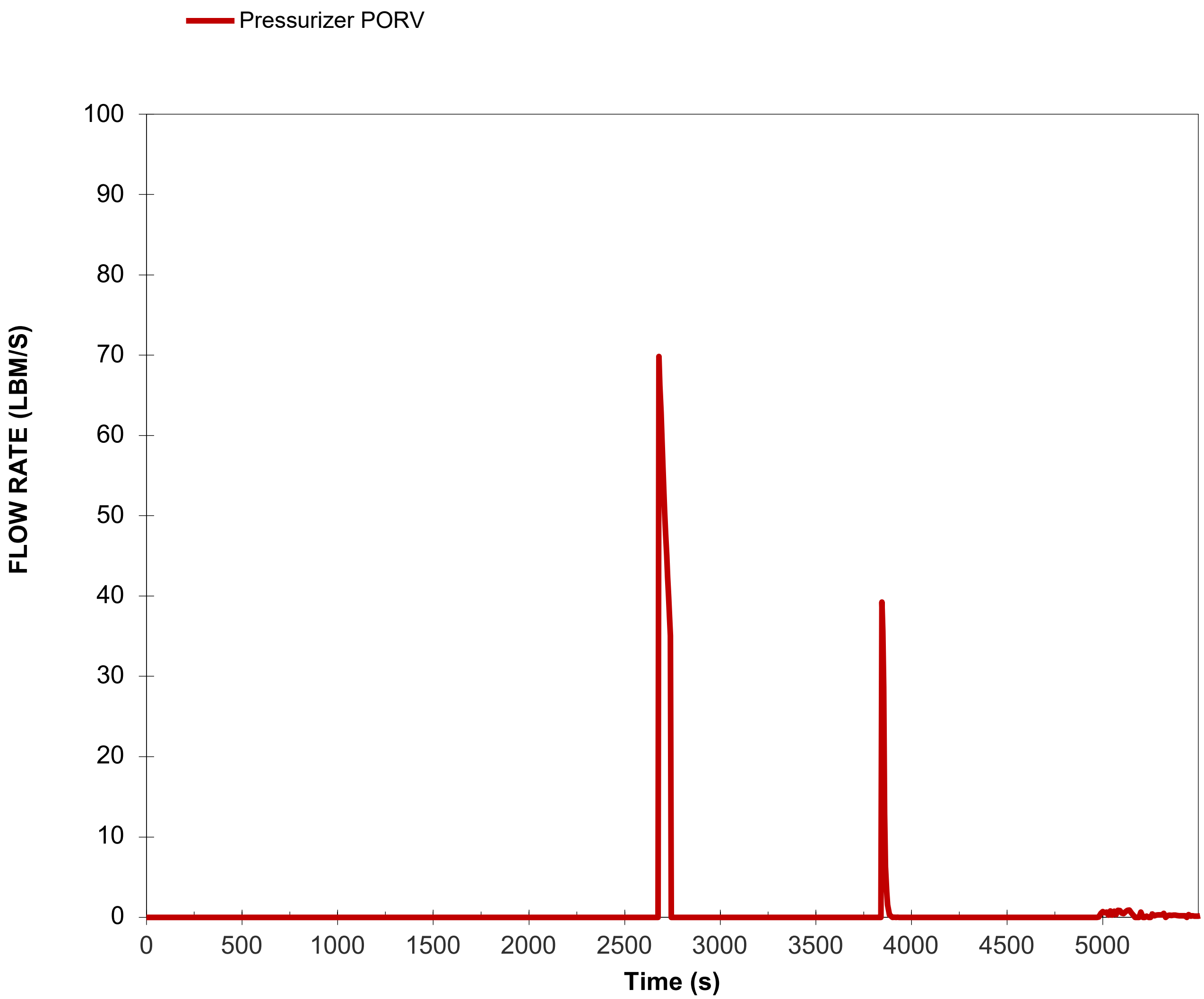
Delete Note

~~Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.~~

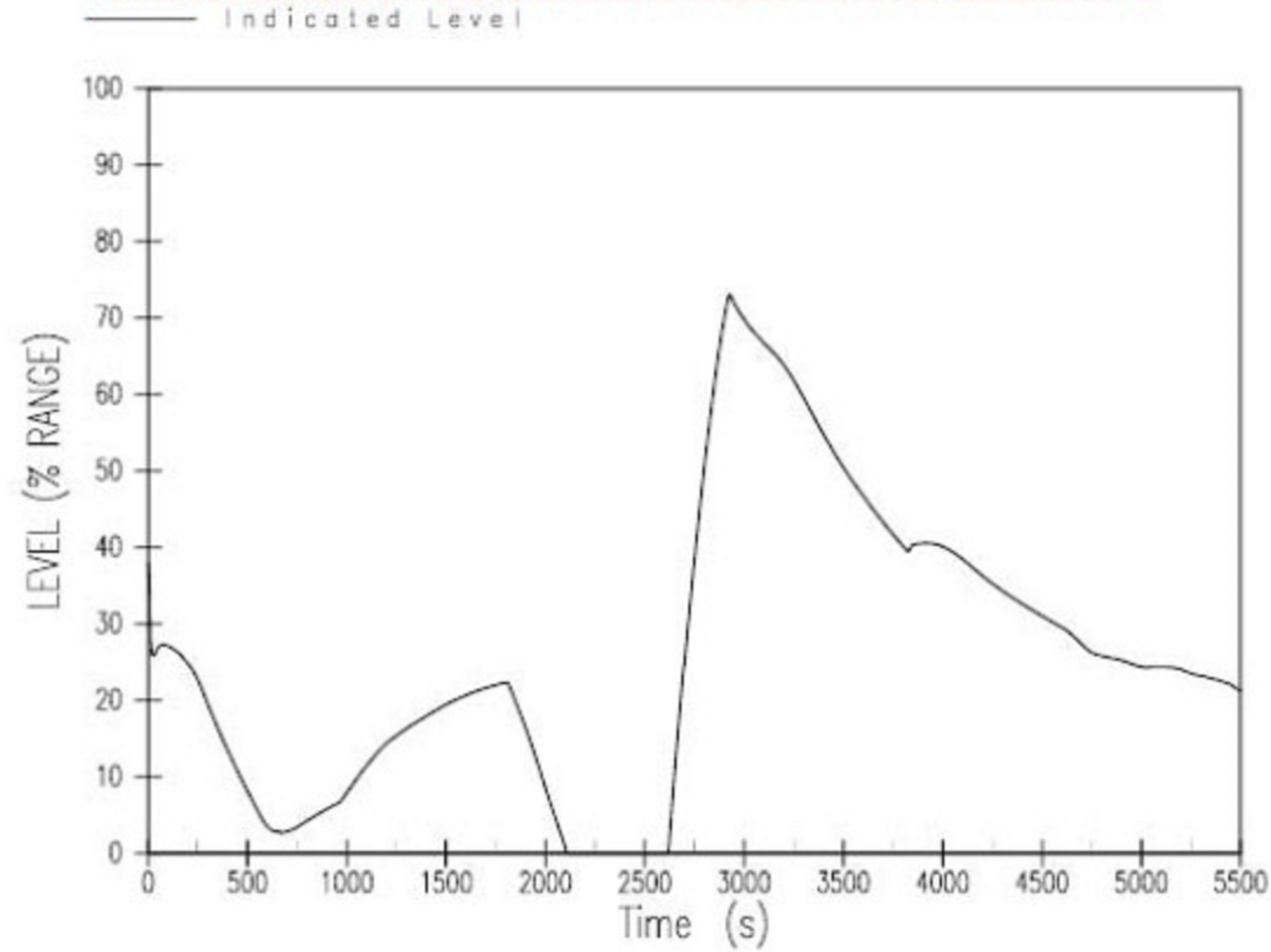
CALLAWAY PLANT

FIGURE 15.6-33M

PRESSURIZER PORV FLOW RATE TRANSIENT FOR SGTR EVENT WITH OVERFILL
REV. 16 6/11


Pressurizer PORV
100
90
80
70
60
50
40
30
20
10
0
FLOW RATE (LBM/S)
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000
Time (s)

Replace with graph on Tab N from spreadsheet "Overfill_FSAR_Plots.xlsx"



~~Note:~~

~~The thermal-hydraulic model presented in this figure is based on an analysis [illegible] ASDs on the intact steam generator [illegible] port the RCS rapid cooldown. [illegible] respect to the thermal-hydraul[illegible] and the potential effects of the associated transient.~~

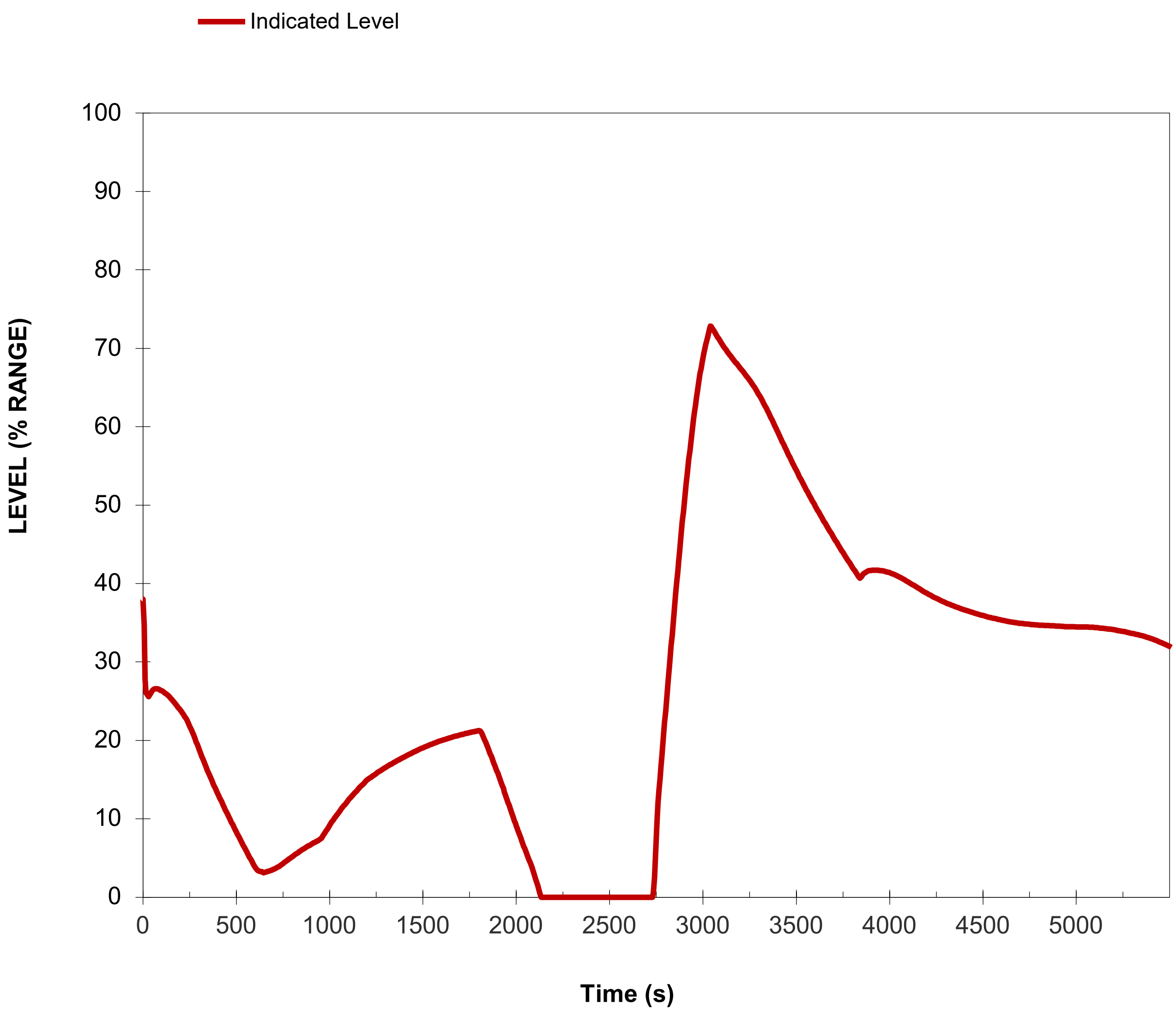
Delete Note

~~Additional calculations have been performed that conservatively reduce the number of credited ASDs available for the rapid cooldown from three to two. This results in a longer rapid cooldown duration than is shown on this figure. The analyses that use the reduced ASD availability were performed to quantify radiological consequences of the prolonged cooldown. Conservative flowrates and durations were used in the radiological consequence analyses.~~

CALLAWAY PLANT

FIGURE 15.6-33N

PRESSURIZER LIQUID VOLUME
TRANSIENT FOR SGTR EVENT
WITH OVERFILL
REV. 16 6/11


Indicated Level
100
90
80
70
60
50
40
30
20
10
0
LEVEL (% RANGE)
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000
Time (s)

Attachment C - FSAR Addendum Markups

TABLE OF CONTENTS (Continued)

LIST OF TABLES (Continued)

Insert AddendumChapter2LOT

Columbia annual mean and the 3-year annual mean site dew-point measurements are identical.

Monthly variation in wind direction amounted to no more than three 22.5-degree sectors, and the annual means of the two data sources (Columbia and on site) were within one 22.5degree sector. Mean monthly wind speed was as much as 1.7 m/sec lower on site than at Columbia (during the month of February) and was an average of 1.2 m/sec lower on site on an annual basis. Since the tendency toward significantly lower wind speed measurements on meteorological towers using stateof-the-art instrumentation compared with airport measurements has been noted in several cases, the disparity between the measurements may be attributed to difference in instrument accuracy rather than actual wind speed differences. On-site data were measured at 10 meters, while the anemometer height at Columbia was 6 meters. Whatever reason for the disparity, the lower speeds measured on site are conservative with respect to dispersion calculations.

The parameter of paramount importance other than wind speed and direction to dispersion calculations, atmospheric stability, is not routinely measured by the NWS. The NWS STAR computer program approximates stability measurements by computing Pasquill stability classes on the basis of cloudiness, sun angle, and time of day. This approximation of long-term regional stability, based on Columbia, Missouri, data, 1960 through 1969, is compared with stability measured on site in Table 2.3-55. It is apparent that the on-site data provide a somewhat greater frequency of stable conditions than does the STAR approximation. The difference is probably due to the crudeness of the STAR method
Columbia data

Sections 2.3.4.1, 2.3.4.2 and 2.3.4.3 describe historical calculations of the short-term diffusion estimates. Chapter 15 dose consequences for accidents, as described in Sections 15A.1 through 15A.4, were re-analyzed using the Alternative Source Term (AST) analysis.

Annual joint fr
atmospheric stability for the 10- and 60-meter wind levels and 60-10 meter ΔT (or 90-10 meter ΔT when 60-10 meter are missing) for the data periods, May 4, 1973 to May 4, 1974 and May 4, 1974 to May 4, 1975 are provided in Tables 2.3-56 and 2.3-57, respectively. Annual JFDs at 10, 60, and 90 meters for the period March 16, 1978 to March 16, 1979 are provided in Table 2.3-58. Table 2.3-59 provides annual JFDs at 10 and 60 meters for the three data periods combined. Monthly JFDs, at 10 and 60 meters, for the three data periods combined are provided in Table 2.3-60.

2.3.4 SHORT-TERM DIFFUSION ESTIMATES

2.3.4.1 <u>Objective</u>

Conservative and realistic estimates of atmospheric diffusion χ/Q at the site boundary (exclusion area) and the outer boundary of the LPZ were performed for time periods up to 30 days after an accident. Diffusion evaluations for short-term accidents are based on the assumption of release points or areas which are effectively lower than 2-1/2 times the height of adjacent solid structures. Description of models used and assumptions made are discussed in section 2.3.4.2.2.

on-site wind direction persistence were discussed in Sections 2.3.2.2.2.4 and 2.3.2.2.2.5, respectively. It is concluded that on-site and regional persistence are similar.

The topography in the vicinity of the site is similar to that in the vicinity of Columbia. Low rolling hills without significant relief occur in both ares, as shown in Figure 2.3-12.

A direct comparison of diffusion estimates based on the on-site data and the long-term (Columbia, Missouri) data would be quite meaningless, because the long-term data do not contain measurements of vertical temperature difference or wind direction variability. In addition, long-term wind speed data are based on anemometer starting thresholds of approximately 2 to 2.5 mph versus starting thresholds of 0.75 mph for the on-site anemometers. The Pasquill-Turner approximation, used to obtain stability classification for long-term meteorology data based on sun angle, cloudiness, and time of day (described in Section 2.3.2 and in Table 2.3-31), is too crude to yield stability values comparable to those based on vertical temperature difference and low-threshold wind speed measurements for determination of stability classification for on-site meteorology data.

2.3.5 LONG-TERM DIFFUSION ESTIMATES

Insert Addendum2.3.4.4

2.3.5.1 Objective

The objective of Section 2.3.5 is to provide realistic long-term diffusion estimates at distances up to 80 km (50 miles) from the plant for annual average release limit calculations and man-rem estimates. The terrain within 80 km (50 miles) of the site is gently rolling; no important ranges of hills or mountains are within the region. There are several small lakes and reservoirs in the region; however, no substantial water bodies are present, which are large enough to affect ambient dispersion parameters.

The analyses were based on on-site meteorological data over the periods, May 4, 1973 to May 4, 1975 and March 16, 1978 to March 16, 1979.

2.3.5.2 Calculations

Both the variable trajectory plume segment atmospheric transport model, MESODIF-II (NUREG/CR-0523), and the straight-line Gaussian dispersion model, XOQDOQ (NUREG/CR-2919), were used to determine for the long-term (annual average) diffusion estimates.

2.3.5.2.1 Plume Segment Atmospheric Transport Model (MESODIF-II)

MESODIF-II is a variable trajectory plume segment atmospheric transport model. It is designed to predict relative atmospheric dispersion factors, χ/Q and deposition factors, D/Q, of radioactive, but otherwise non-reactive material. In such a model, calculated

REFERENCES - SECTION 2.3 (Continued)

Dames & Moore, 1980, Final safety analysis report, Wolf Creek, for Kansas Gas & Electric Co.

Dunn, W., Boughton, B., and Policastro, A., 1978, Evaluation of droplet evaporation formulations employed in drift deposition models. Proceedings of Symposium on Envir[illegible] (May 2-4).

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Egan, B.A., [illegible] and envir[illegible] environmental assessment, Hagen, D., Amer. Meteor. Soc., Boston, Massachusetts, p. 123-124.

Englemann, R.J., 1968, The calculation of precipitation scavenging, in Meteorology and atomic energy. U.S. Dept. of Commerce, Springfield, Virginia, p. 208-220.

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Frost, R., 1948, Quart. J. Roy. Met. Soc., vol. 74.

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APPENDIX 3.A - CONFORMANCE TO NRC REGULATORY GUIDES

This appendix briefly discusses the extent to which Union Electric conforms to NRC published regulatory guides for the site related portions of Callaway Plant. The Standard Plant FSAR Appendix 3A may refer to the Addendum Appendix 3A or the Union Electric Operational Quality Assurance Manual (OQAM) for the specific regulatory commitment for certain regulatory guides. However in cases where a reference is not made to the Addendum Appendix 3A or the OQAM, the commitment is as stated in the Standard Plant Appendix 3A and the same regulatory position is not repeated in the Addendum Appendix 3A or in the OQAM. The statement of specific regulatory commitment for the following regulatory guides is located as indicated:

Callaway FSAR, Standard Plant - Regulatory Guides 1.1, 1.2, 1.3, ~~1.4,~~ 1.5, 1.6, 1.7, 1.9, 1.10, 1.11, 1.12, 1.13, 1.14, 1.15, 1.18, 1.20, 1.22, 1.24, ~~1.25,~~ 1.26, 1.29, 1.31, 1.32, 1.34, 1.35, 1.36, 1.40, 1.41, 1.42, 1.43, 1.44, 1.45, 1.46, 1.47, 1.48, 1.49, 1.50, 1.51, 1.52, 1.53, 1.54, 1.55, 1.56, 1.57, 1.59, 1.60, 1.61, 1.62, 1.63, 1.65, 1.66, 1.67, 1.68, 1.68.1, 1.68.2, 1.69, 1.70, 1.71, 1.72, 1.73, 1.75, 1.76, 1.77, 1.78, 1.79, 1.80, 1.81, 1.82, 1.83, 1.84, 1.85, 1.87, 1.89, 1.90, 1.92, 1.93, 1.95, 1.96, 1.97, 1.98, 1.99, 1.100, 1.101, 1.102*, 1.103, 1.104, 1.105, 1.106, 1.107, 1.108, 1.110, 1.112, 1.115, 1.117, 1.118, 1.119, 1.120, 1.121, 1.122, 1.124, 1.126, 1.128, 1.129, 1.130, 1.131, 1.133, 1.136, 1.137, 1.139, 1.140, 1.141, 1.142, 1.143, 1.147, 1.150, 1.152, 1.155, 1.158, 1.160, 1.163, 1.181, 1.182, 1.183, 1.187, and 1.195.

Callaway FSAR, Site Addendum - Regulatory Guides 1.17, 1.21, 1.23, 1.27, 1.59, 1.86, 1.91, 1.102*, 1.109, 1.111, 1.113, 1.114, 1.125, 1.127, 1.132, 1.134, 1.138, and 1.145 and 1.194.

Union Electric Operational Quality Assurance Manual - Regulatory Guides 1.8, 1.28, 1.30, 1.33, 1.37, 1.38, 1.39, 1.58, 1.64, 1.74, 1.88, 1.94, 1.116, 1.123, 1.144, and 1.146.

Clarifications, alternatives, and exceptions to these guides are identified and justification is presented or referenced. In the discussion of each guide, the sections or tables of the FSAR where more detailed information is presented are referenced. The referenced tables provide a comparison of Union Electric's position to each regulatory position of section C of the regulatory guides. All statements within the Regulatory Position Section (C) of the Regulatory Guides are considered requirements unless a specific exception or clarification has been committed to by Union Electric. This is true regardless of the qualifier (i.e., "shall" or "should") which prefaces the statement. As regards to standards endorsed by the Regulatory Guide, unless further qualified within the Regulatory Guide, "shall" statements denote requirements while "should" statements denote recommendations. A glossary of definitions is provided in the Quality Assurance Procedures Manual.

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* Refer to both the Standard Plant and the Site Addendum for the Complete statement of regulatory commitment.

REGULATORY GUIDE 1.144

REGULATORY GUIDE 1.145, Revision 1, DATED 11/82
Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plant
DISCUSSION
The recommendations of this regulatory guide are met as described in Site Addendum Section 2.3.4.4 for Alternative Source Term Short-Term Diffusion Estimates.

Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants

DISCUSSION:

UE complies with the recommendations described in the Draft Regulatory Guide 1.XXX (1978). Refer to Site Addendum Section 2.3.4.2.1 for a discussion of short-term diffusion estimates.

REGULATORY GUIDE 1.146

Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants

DISCUSSION:

Refer to the Union Electric Company Operational Quality Assurance Manual.

REGULATORY GUIDE 1.160 REVISION 2 DATED 3/97

Monitoring the Effectiveness of Maintenance at Nuclear Power Plants

DISCUSSION:

Refer to Appendix 3A of the Standard Plant FSAR.

REGULATORY GUIDE 1.194 DATED 6/03
Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants
DISCUSSION:
The recommendations of this regulatory guide are met as described in Site Addendum Section 2.3.4.4 for Alternative Source Term Short-Term Diffusion Estimates.

Attachment D - Inserts for FSAR and Addendum

Insert 2.3.4.2.2

2.3.4.2.2.2 Alternative Source Term (AST Analysis)

The short-term atmospheric dispersion factors (χ/Qs) are based on onsite meteorological data for the Callaway Plant site. The diffusion equations and assumptions used in the calculations are those outlined in NRC Regulatory Guide 1.194, "USNRC, Regulatory Guide 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” Revision 1, June 2003." Table 2.3-1 lists the limiting χ/Qs for the Callaway site. The detailed procedures used in the calculations are given in Section 2.3.4.4 of the Site Addendum.

Insert 6.5.5

4. NUREG-0800, Standard Review Plan Section 6.5.2, Revision 4, “Containment Spray as a Fission product Cleanup System,” March 2007.

Insert T6.5-1

Intake Pathways – Typical Accident Alignment

		Filter Removal Efficiency			
		Aerosol HEPA	Organic Iodine	Elemental Iodine	Nominal Flow (scfm)
Control Building (Compartment 12)					
Path 16	Filtered Intake from Env	95%	0%	0%	900.0 (30 Min) 450.0 (to end)
Path 17	Unfiltered Inleakage from Env	0%	0%	0%	0 (up to 6000 maximum)(1)
Path 30	Unfiltered from CBER via HVAC	0%	0%	0%	1018.8 (flow balance) 688.8 (to end)
Path 40	Unfiltered from Normal HVAC	0%	0%	0%	0 (N/A for accident)
Control Room Filter Unit (CRFU) Intake Plenum (Compartment 13)					
Path 18	Unfiltered Inleakage from CB	0%	0%	0%	440
Path 26	Unfiltered Inleakage from CR	0%	0%	0%	1030 (flow balance)
Path 28	Unfiltered from CBER	0%	0%	0%	330
Control Room Air Conditioning (CRAC) Mixing Plenum (Compartment 14)					
Path 21	Filtered Intake from CRFU	95%	95%	95%	1800
Path 31	Unfiltered Inleakage from CB	0%	0%	0%	16200 (flow balance)
Control Building Equipment Room (Compartment 15)					
Path 22	Unfiltered from CRAC	0%	0%	0%	385.0
Path 25	Unfiltered from Env	0%	0%	0%	663.8 (30 min)
		0%	0%	0%	333.8 (to end)
Path 29	Unfiltered from Env	0%	0%	0%	300. 0 (Maximized)
Path 41	Unfiltered from Normal HVAC	0%	0%	0%	0 (N/A for accident)
Main Control Room (Compartment 16)					
Path 19	Unfiltered from CB	0%	0%	0%	440 (30 Min) 0 (to end)
Path 23	Unfiltered from CRAC	0%	0%	0%	17615
Path 24	Unfiltered Inleakage from Env	0%	0%	0%	60 maximum(1)
Path 39	Unfiltered from Normal HVAC	0%	0%	0%	0 (N/A for accident)
Normal CB/CBER/CR HVAC (Compartment 19)					
Path 36	Unfiltered Intake from Env	0%	0%	0%	18,000 until control room isolation, then 0

Environment (Compartment 11)

The environment compartment receives all exhaust/outleakage that does not enter other compartments. There are no “dummy” exhaust compartments in the dose model. All exhaust to the environment is unfiltered.

LOCA					
Path 13	Aux Bldg Emer Exhaust to Env	90%	90%	90%	Flowrate Not Used (no mixing/no holdup in Aux Bldg)
FHA					
Path 13	Aux Bldg Emer Exhaust to Env	0%	0%	0%	
Path 20	Exhaust from CB	0%	0%	0%	6993.8 (30 min) 6653.8 (to end)
Path 27	Exhaust from CR (accident)	0%	0%	0%	206.3 (30 min) 96.3 (to end)
Path 37	Exhaust from CR (normal)	0%	0%	0%	0 (N/A for accident)
Path 38	Exhaust from normal HVAC	0%	0%	0%	0 (N/A for accident)

Note: Compartment and pathway names and numbering are arbitrarily assigned, and are generally consistent with the dose analysis computer code model. Individual dose analysis models used in calculations may differ slightly in name or numbering scheme. See FSAR Figure 15A-3 for pathway number context.

(1) The limiting Control Building and Control Room inleakage values for radiological consequences are obtained from Figure 15A-2.

Insert 6.5A.3

Section III.4.c (1) of Reference 14 specifies the following formula for the spray removal of elemental iodine:

$$\lambda_S = \frac{6K_g\,TF}{VD}$$

Where: λ_S = spray removal rate coefficient for elemental iodine
K_g = gas-phase mass transfer coefficient
T = Fall time for spray droplets
F = volumetric flow rate of the spray
V = net free (air) volume of sprayed region
D = mass-mean diameter of the spray drops

From Table 6.5-2:

Containment volume:	2.70E6 ft^3
Fraction containment sprayed:	85%
Average fall height:	131.4 ft
Spray flow rate:	3086 gpm

Gas phase mass transfer coefficient, K_g: 9.5 ft/min
Terminal Velocity: 790 ft/min:
Mean drop diameter: 831 microns

The spray flow, F = (3086 gal/min)*(0.13368 ft^3/gal) = 412.5 ft^3/min
In this application, a smaller flow produces a slower removal of iodine; therefore, the lowest expected flow value is used. As a conservative simplification, the increase in flow associated with the recirculation phase is neglected.

Volume sprayed, V = (2.7 million ft^3 total) (0.85 as fraction sprayed) = 2.3 million ft^3
In this application, a larger volume produces a slower removal of iodine.

Drop diameter, D = (831 microns)(1 m/10^6 microns)(3.281 ft/m) = 0.00273 ft

The fall time (T) may be calculated as the ratio of the average fall height to the terminal velocity.
Fall time, T = 131.4 ft / 790 ft/min = 0.166 minutes

$$\lambda_s = \frac{6(9.5\,ft/\min)(0.166\,\min)(412.5\,ft^3/\min)(60\,\min/hr)}{(2.30E+06\,ft^3)(0.00273\,ft)} = 37.3\ hr^{-1}$$

The Standard Review Plan (SRP) 6.5.2 (Reference 14) limits λ_S to 20; therefore, 20 is used for the elemental iodine spray removal coefficient in the dose calculations of Section 15.6.5. In accordance with the SRP, the effectiveness of the spray in removing elemental iodine is required to end when the amount has been reduced by a factor of 200 (DF=200).

Insert 6.5A.4

6.5A.4 PARTICULATE MODEL FOR OFFSITE AND CONTROL ROOM DOSE CALCULATIONS

Section III.4.c (4) of Reference 14 specifies the following formula for the spray removal of aerosols (particulates):

$$\lambda_P = \frac{3hFE}{2VD}$$

Where: λ_P = spray removal rate coefficient for aerosols
h = spray fall height
F = volumetric flow rate of the spray
E/D = ratio of dimensionless collection efficiency to spray drop diameter
V = volume of sprayed region

From Table 6.5-2:
Containment volume: 2.70E6 ft^3
Spray flow rate: 3086 gpm
Average Fall Height: 131.4 ft

Spray Flow, F = (3086 gal/min)(0.13368 ft^3/gal) = 412.5 ft^3/min
Volume sprayed, V = (2.7 million ft^3 total) (0.85 as fraction sprayed) = 2.3 million ft^3

SRP 6.5.2 (Reference 14) specifies that E/D = 10 m^{-1} initially and E/D = 1 m^{-1} after the aerosol mass has been reduced by a factor of 50.

$$\lambda_p = \frac{3(131.4\,ft)(412.5\,ft^3/\min)(10/m)(60\min/\,hr)}{2(3.281\,ft/m)(2.30E+06\,ft^3)} = 6.46\ hr^{-1}$$

Appendix A of Reference 15 requires that the particulate spray removal coefficient be reduced by a factor of 10 when a Decontamination Factor of 50 is reached for the aerosols (particulates).

Insert 6.5A.5

14. NUREG-0800, Standard Review Plan Section 6.5.2, Revision 4, “Containment Spray as a Fission product Cleanup System,” March 2007.

15. Regulatory Guide 1.183, original version, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000

Insert Chapter15TOC-A

Insert Chapter15List_of_Tables

Insert 15.0.9

The calculation of the core fission product inventory employs ORIGEN-S of the Oak Ridge National Laboratory (ORNL) SCALE 6.1.3 code package (Reference 22). ORIGEN-S is an isotopic depletion and decay code which allows the user to specify fuel type, enrichment and periods of irradiation/decay and uses the latest cross-section data from ORNL to determine the existing nuclide inventory at specified intervals.

The core is modeled as an eight batch enveloping cycle core with a core power level of 3636 MWt (3565 MWt plus 2% postulated calorimetric error).

The Batch 1 assembly operated at an average power of 58.58 MW/MTU for 501.5 EFPD and 39.35 MW/MTU for 546.6 EFPD.

The Batch 2 assemblies operated at an average power of 55.23 MW/MTU for 502.5 EFPD, 21.17 MW/MTU for 498.5 EFPD, and 13.10 MW/MTU for 546.6 EFPD.

The Batch 3 assemblies operated at an average power of 53.25 for 498.5 EFPD and 34.39 MW/MTU for 546.6 EFPD.

The Batch 4 assemblies operated at an average power of 54.24 MW/MTU for 498.5 EFPD and 44.17 MW/MTU for 546.6 EFPD.

The Batch 5 assemblies operated at an average power of 51.42 MW/MTU for 498.5 EFPD and 46.39 MW/MTU for 546.6 EFPD. Batch 6 assemblies operated at an average power of 52.07 MW/MTU for 546.6 EFPD.

The Batch 7 assemblies operated at an average power of 47.15 MW/MTU for 546.6 EFPD.

The Batch 8 assemblies operated at an average power of 109.78 MW/MTU for 546.6 EFPD.

The total burnups in Batches 1 through 8 at the end of the analyzed cycle (MWD/MTU) are as follows:

Batch	Number of assemblies	Exposure (MWd/MTU)
1	1	50,885
2	16	45,462
3	60	45,340
4	12	51,176
5	4	50,985
6	60	28,460
7	32	25,770
8	8	60,000

The isotopic yields utilize data for fissioning of U-235, U-238, and Pu-239 and account for the depletion of the fuel. Radiological consequences are evaluated with source terms based on the 3636 MWt core rating (Table 15A-3), Callaway-specific meteorology based on four years of combined meteorological data (Table 15A-2), and appropriate dose conversion factors (Table 15A-4)

.

Insert 15.0.11.8

RETRAN-3D evolved from continued development of RETRAN-02. Both the steady-state and transient numerical solutions methods in RETRAN-3D have been revised to use an implicit solution. This results in much improved steady-state initialization convergence for two-phase systems and a more stable transient solution. RETRAN-3D retains the analysis capabilities of RETRAN-02 and also has improved modeling capability for small break loss-of-coolant accidents and anticipated transients without scram. RETRAN-3D also has model extensions designed to provide analysis capabilities for long-term transients and transients with limited thermodynamic nonequilibrium phenomena. RETRAN-3D is used by a large number of domestic and foreign electric utilities and research organizations. RETRAN-3D has been reviewed by the USNRC and was issued a generic SER in 2001 that removed many of the conditions for RETRAN-02. RETRAN-3D is implemented in the RETRAN-02 mode for the analyses supporting the Alternate Source Term (AST) steam release calculations.

Insert 15.1.5.3.1.2A

Case 1 The initial reactor coolant concentrations of radioactive isotopes are assumed to be the dose equivalent of 1.0 µCi/gm of I-131 with an iodine spike that increases the rate of iodine release into the reactor coolant by a factor of 500. This increased rate of transfer to the coolant is assumed to be for the first 8 hours.

Case 2 An assumed reactor coolant concentration of radioactive isotopes with a dose equivalent of 60µCi/gm of I-131 as a result of a pre-accident iodine spike.

Insert 15.1.5.3.1.2B

f. The reactor coolant concentrations of alkali metals correspond to 1-percent of the fuel having cladding defects as provided in Table 11.1-5.

Insert 15.1.5.3.2

d. Reactor coolant activities based on iodine spiking effects are conservatively high.

e. The most conservative return to power, decay heat, and metal mass sensible heat are all included in the steam release calculation to provide the most conservative result.

f. It is assumed that AFW is isolated to one of the intact loops as well as the faulted loop as part of the operator actions taken to isolate the faulted steam generator. Thus the ASDs on only two loops are used for the cooldown to RHR conditions which further extends the cooldown and therefore increases the total steam release

Insert T15.1-3A

Case 1 The initial reactor coolant concentrations of radioactive isotopes are assumed to be the dose equivalent of 1.0 µCi/gm of I-131 with an iodine spike that increases the rate of iodine release into the reactor coolant by a factor of 500. This increased rate of transfer to the coolant is assumed 8 hours.

Case 2 An assumed reactor coolant concentration of radioactive isotopes with a dose equivalent of 60µCi/gm of I-131 as a result of a pre-accident iodine spike.

Insert T15.1-3B

e. Reactor coolant alkali metal activity:

1) Case 1 Based on 1-percent fuel having cladding defects as provided in Table 11.1-5

2) Case 2 Based on 1-percent fuel having cladding defects as provided in Table 11.1-5

Insert T15.1-3C

h. Alkali metal partition factors

1)	Faulted steam generator	1
2)	Intact steam generators	0.01

Insert 15.2.6.3.1.2A

a. The reactor coolant initial iodine activity is determined by two methods, and both cases are analyzed. These are:

Case 1 The initial reactor coolant concentrations of radioactive isotopes are assumed to be the dose equivalent of 1.0 µCi/gm of I-131 with an iodine spike that increases the rate of iodine release fuel into the into the reactor coolant by a factor of 500. This increased rate of transfer to the coolant is assumed to be for the first 8 hours.

Case 2 An assumed reactor coolant concentration of radioactive isotopes with a dose equivalent of 60µCi/gm of I-131 as a result of a pre-accident iodine spike.

b. The reactor coolant activity assumed for noble gas is the Technical Specification limit of 225 µCi/gm Xe-1 33 dose equivalent

c. The reactor coolant system activity assumed for alkali metals is based on 1% fuel defects, as provided in Table 11.1-5.

Insert 15.2.6.3.1.2B

e. The alkali particulates are conservatively combined with, and treated as, halogens for transport through the steam generators.

Insert 15.2.6.3.1.2C

The partition fraction for iodine and alkali metals in the steam generators is taken as:

i.) 0.01 for bulk boiling of the water in the secondary, and

ii.) Equal to the fraction of primary-to-secondary leakage that flashes to steam. This flashing fraction is conservatively held at an initial value of 5% for the first 2.667 hours and decreased to zero thereafter.

Insert T15.2-2A

c. The reactor coolant initial iodine activity is determined by two methods, and both cases are analyzed. These are:

Case 1 The initial reactor coolant concentrations of radioactive isotopes are assumed to be the dose equivalent of 1.0 µCi/gm of I-131 with an iodine spike that increases the rate of iodine release fuel into the into the reactor coolant by a factor of 500. This increased rate of transfer to the coolant is assumed to be for the first 8 hours.

Case 2 An assumed reactor coolant concentration of radioactive isotopes with a dose equivalent of 60µCi/gm of I-131 as a result of a pre-accident iodine spike.

Insert 15.3.3.3.1.1

The assumptions used to determine the initial concentrations of isotopes in the reactor coolant and secondary coolant prior to the accident are as follows:

a. Based upon inclusion of fuel failure in this event, the dose contribution from the initial RCS activity was neglected.
b. The secondary side coolant initial concentrations are assumed to be the dose equivalent of 10% of 1.0 μCi/gm dose equivalent of I-131.

Insert 15.3.3.3.1.2

g. The partition factor for iodine released by bulk boiling in the steam generators is taken as 0.01 for secondary side releases.

h. Five percent of the primary-to-secondary leakage flashes to vapor during the first 2.4 hours and has no mitigation when released to the environment. Ninety-five percent of the primary-to-secondary leakage mixes with the secondary water. These assumptions are conservatively based on a leak in the upper tubes which are assumed to be uncovered for the first 2.48 hours.

Insert T15.3-3

c. Core Inventories — See Table 15A-3
d. Radial leaking factor — 1.65
e. Extent of core damage — 5%
f. Percent of core inventory initially present in the fuel gap with a maximum of 35 rods per assembly exceeding Regulatory Guide 1.183 burnup limits:

Isotope	Burnup >54 GWD/MTU*	Burnup <54 GWD/MTU
I-131	0.12	0.08
Kr-85	0.30	0.10
Other Noble Gases	0.10	0.05
Other Halogens	0.10	0.05
Alkali Metals	0.17	0.12

g. Iodine and alkali metal partition factor in the steam generators for secondary side releases — 0.01

h. Primary-to secondary leakage flashing to vapor — 5%*

i. Primary-to-secondary leakage mixing with secondary water — 95%*

*: for the first 2.48 hours

Insert 15.4.8.3.1.2A

This is the percent of core inventory initially present in the fuel gap:

Isotope	Burnup >54 GWD/MTU	Burnup <54 GWD/MTU
I-131	0.12	0.10
Kr-85	0.30	0.10
Other Noble Gases	0.10	0.10
Other Halogens	0.10	0.10
Alkali Metals	0.17	0.12

As a result of fuel failure, 10% of the fuel gap activity is released (in separate cases) to the reactor and to the containment atmosphere with adjustment for the radial power distribution. This release is in addition to that released from the assumed 0.25% fuel melt.

Insert 15.4.8.3.1.2B

100% of the noble gases in the melted fuel is released. 50% of the iodines in the melted fuel is released to the reactor coolant. In a separate case, 25% of the iodines in the melted fuel is released to the containment atmosphere.

Insert 15.4.8.3.1.2C

g. The partition factor for iodine in the steam generators is taken as 0.01 for secondary side releases.

h. Five percent of the primary-to-secondary leakage flashes to vapor for the first 2.622 hours and has no mitigation when released to the environment. Ninety-five percent of the primary-to-secondary leakage mixes with the secondary water for the first 2.622 hours and 100% thereafter. These assumptions are conservatively based on a leak in the upper tubes which are assumed to be uncovered for the accident duration.

Insert T15.4-3

k.	Primary-to secondary leakage flashing to vapor	5%*
l.	Primary-to-secondary leakage mixing with secondary water	95%*

*: for the first 2.622 hours

Insert 15.6.5.4.1

The analysis of the radiological consequences of a postulated LOCA uses the recommended dose conversion factors (DCFs) as follows:

1. The analysis uses Committed Effective Dose Equivalent (CEDE) dose conversion factors (DCFs) for inhalation of radionuclides based on the date provided on 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." (Reference 34)
2. The Effective Dose Equivalent (EDE) dose conversion factors, provided in Table III.1 of Federal Guidance Report 12, “External Exposure to Radionuclides in Air, Water, and Soil,” (Reference 35) are used to calculate the external exposure doses.

Insert 15.6.5.4.1.4

This calculation assumes:

1. The initial activity concentration in the Reactor Coolant System corresponds to the 1 μCi/gm Dose Equivalent Iodine-131 and 225 μCi/gm Dose Equivalent Xe-133 equilibrium limits. Iodine spikes need not be considered.
2. The release of all the fission products in the RCS to the containment is assumed to occur instantaneously following the break in reactor coolant piping.
3. Because the 11 second duration is less than the 30 second delay for onset of the gap release phase in Table 4 of Regulatory Guide 1.183, no direct release of activity from the fuel is applicable.
4. Two initially open 18” mini-purge lines are isolated at 11 seconds.
5. The maximum flow rate through the mini-purge lines is calculated as a function of containment pressure until isolation at 11 seconds. See Part “g.” of Table 15.6-6.
6. Containment pressure is calculated in response to the mass and energy release from a Double Ended Cold Leg pipe break with two open 18” mini-purge lines.
7. From FSAR Figure 11.3-2, the mini-purge exhaust is routed through the Unit Vent to the environment.
8. While the mini-purge flow leaving containment is normally filtered, this filtration is not included in the Engineered Safety Feature portion of system and so is not modeled.
9. Because the 11 sec duration of the release is faster/shorter than the time delay associated with control room isolation, no filtration of air flow to the control room is modeled.

Insert 15.6.5.4.2

d. Atmospheric Dispersion

Exclusion Area Boundary and LPZ. The meteorological conditions assumed to be present at the site during the course of the accident are based on χ/Q values, which are the larger of the 5 percent overall site values and the 0.5 percent maximum sector values. This condition results in the poorest values of atmospheric dispersion calculated for the exclusion area boundary and the LPZ outer boundary. Furthermore, no credit has been taken for the transit time required for activity to travel from the point of release to the exclusion area boundary and LPZ outer boundary. Hence, the radiological consequences evaluated under these conditions are conservative.

Control Room. The meteorological conditions assumed to be present at the site during the course of the accident are based on χ/Q values, which are expected to be exceeded 5 percent of the time. No credit has been taken for the transit time required for activity to travel from the point of release to the exclusion area boundary and LPZ outer boundary. Hence, the radiological consequences evaluated under these conditions are conservative.

Insert T15.6-1

Accident	Event	Time (sec)
.		
.		
Steam generator tube rupture with stuck-open atmospheric steam dump (ASD) valve	Tube rupture occurs	0.0
	Reactor trip and Safety Injection Injection Signal	600*
	Loss of offsite power	600*
	Ruptured steam generator atmospheric steam dump valve opens	600*
	Delivery of Safety Injection begins	~620
	Auxiliary feedwater injection begins	~660
	Operator isolates ruptured steam generator by manually closing block valve	1800 = 600 + 1200
	Operator initiates RCS cooldown via intact steam generator atmospheric steam dump valves	2400 = 1800 + 600
	Operator completes RCS cooldown	3160
	Operator initiates RCS depressurization via pressurizer PORVs	3340 = 3160 + 180 sec
	Operator completes RCS depressurization	3400 = 3340 + 60 sec
	Operator terminates safety injection	3700 = 3400 + 300 sec
	Operator equalizes primary-secondary pressure	4600 = 3700 + 900 sec
	Operator opens intact SG ASDs	4800
	Operator opens block valve on ruptured SG	19,000
	RHR cut-in conditions reached	21,100 = 5.86 hours

***** This value reflects application of a slight conservative bias.

Insert T15.6-4sheet2

III. Activity Release Data	stuck open ASD	overfill case
a. Ruptured steam generator		
1. Reactor coolant discharged to steam generator, lbs	300,000 [(1)]	240,000 [(2)]
2. Flashed reactor coolant, %	16.1 for first 3000 sec (3)	zero [(4)]
3. Iodine partition factor for flashed fraction of reactor coolant	1.0	not applicable
4. Steam release to atmosphere, lbs		
0 - 2 hours	114,107	
1.361 - 2 hours		13,598
2 – 5.86 hours	111,380	
2 – 9.39 hours		83,769 [(5)]
5. Iodine carry over factor for the non-flashed fraction of reactor coolant that mixes with the initial iodine activity in the steam generator (as bulk boiling) for steam release	0.01	0.01
6. Liquid release to atmosphere, lbs		
0.264 – 1.361 hours		194,985
7. Iodine partition factor for liquid release from SG		0.5
8. Iodine partition factor for normal steam flow to condenser prior to reactor trip and loss of offsite power	0.01	not applicable
b. Unaffected steam generators		
1. Primary-to-secondary leakage, lbs	2932 [(6)]	3198 [(7)]
2. Flashed reactor coolant, %	16.1 for first 3000 sec [(3)]	4.0 between 6000 and 9000 sec
3. Total steam release, lbs		
0 - 2 hours	321,930	396,435
2 – 5.86 hours	647,375	
2 – 6.4 hours		581,117
4. Iodine carry over factor for bulk boiling	0.01	0.01
5. RHR Cut-in time, hours	5.86	6.4

Insert T15.6-4sheet3

Notes:

(1) While RETRAN calculates a total integrated break flow of 220,243 lbs, this is conservatively biased to a value of (60 lb/sec) x (5000 sec) = 300,000 lbs for use as the basis for offsite and control room dose.

(2) While RETRAN calculates a total integrated break flow of 193,200 lbs, this is conservatively biased to a value of (60 lb/sec) x (4000 sec) = 240,000 lbs for use as the basis for offsite and control room dose.

(3) The 16.1% flashing fraction is based on initial conditions in the RCS and steam generator secondary side. As a conservative simplification, this fraction is applied for the first 3000 seconds in both the ruptured and intact steam generators. After that, the RETRAN computer code model used to calculate steam releases to the environment shows that the collapsed water level in the secondary side covers the entire tube bundle. The SG tubes remain covered with water for the remainder of the accident.

(4) For the overfill case, the collapsed liquid level in the secondary side of the ruptured SG remains above the top of the tube bundle for the duration of the accident. Since the tube break is covered with water, credit is taken for scrubbing by liquid in the secondary side.

(5) While RHR cut-in conditions are reached at 6.4 hours, additional time is required to reduce the ruptured steam generator secondary side temperature below 212˚F.

(6) All of the normal allowable steam generator tube leakage is conservatively assumed to occur within the three intact steam generators. The total leakage is 1 gpm = 8.34 lb/min for a duration of 5.86 hours.

(7) The total leakage is 1 gpm = 8.333 lb/min for a duration of 6.397 hours.

Insert T15.6-6A

c. Release fraction and timing of core activity in the containment:

	Group	Gap Release Phase Fraction (30 sec-0.5 hour)	Early In-vessel Phase Fraction (0.5 hour-1.3 hour)
1	Noble gases	0.05	0.95
2	Halogens	0.05	0.35
3	Alkali metals	0.05	0.25
4	Tellurium metals	0.00	0.05
5	Barium and Strontium	0.00	0.02
6	Noble metals	0.00	0.0025
7	Cerium group	0.00	0.0005
8	Lanthanides	0.00	0.0002

Insert T15.6-6B

g. Equilibrium sump pH > 7.0

h. Reactor Coolant Activity (mini-purge only)

1. Iodine — Dose equivalent of 1.0 µCi/gm of I-131
2. Noble gas — Dose equivalent of 225 µCi/gm of Xe-133
3. Alkali metal — Based on 1% failed fuel as provided in Table 11.1-5

Insert T15.6-6C

g. Mini-purge initially in operation

initial air mass in Containment = 172,222 lb.

total RCS fluid mass (released to containment) = 551,068 lb.

Flow from Containment to Environment

Time after break	Containment pressure	Mini-purge flow rate to environment
(sec)	(psia)	(lbm/min)
0	14.7	0.00E+00
0.6	19.7	1.09E+04
0.8	21.1	1.22E+04
1	22.5	1.32E+04
2	28.4	1.65E+04
3	32.8	1.85E+04
4	36.3	2.01E+04
5	39.3	2.15E+04
5.5	40.7	2.21E+04
6	42	2.27E+04
6.5	43.2	2.33E+04
7	44.3	2.38E+04
7.5	45.3	2.43E+04
8	46.2	2.47E+04
8.5	47	2.50E+04
9	47.6	2.53E+04
9.5	48.2	2.56E+04
10	48.7	2.58E+04
10.5	49.2	2.60E+04
10.999	49.5	2.62E+04
11		0.00E+00

Insert 15.7.4.5.1.2.e

is defined by Table 3 of Reg Guide 1.183 (Reference 1) that lists the fraction of the fission product inventory that is in the fuel gap subject to the following provisions:

> "The 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"

For the assumed maximum of 32 fuel rods per assembly that does not meet the above criteria (total of 264 rods per assembly), release fractions from NUREG/CR-5009 (Reference 2) have been applied. The table below lists the Reg. Guide 1.183 and NUREG/CR-5009 gap fractions used for Non-LOCA accidents.

Non-LOCA Fraction of Fission Products in Fuel Gap

Group	RG1.183 Table 3 Fraction	NUREG/CR-5009 Fraction
I-131	0.08	0.12
Kr-85	0.10	0.30
Other Noble Gases	0.05	0.10
Other Halogens	0.05	0.10
Alkali Metals	0.12	0.17

The gap release from the fuel during the accident has been adjusted to account for the higher release fraction in the portion of high burnup fuel in each assembly. For the FHA in the containment, the 20% of fuel rods damaged in the additional assembly are conservatively assumed to include all of the high burnup rods in that assembly.

Insert 15.7.4.5.1.2.g

As noted in Item 8 of Reference 3, the elemental iodine decontamination factor to be used is 285. The organic decontamination factor is 1.0, for an overall iodine decontamination factor of 200.

Insert 15.7.4.5.1.2.h

Per Section 1.3 of Appendix B of Regulatory Guide 1.183, the chemical form of radioiodine released from the fuel to the spent fuel pool is 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 re-evolve as elemental iodine. Therefore, the chemical form of the released iodine is 99.85% elemental and 0.15% organic.

Following application of the iodine decontamination factors discussed in 15.7.4.5.1.2.g the resulting chemical composition of the iodine release above the pool is 70% elemental and 30% organic.

Insert 15.7.5

15.7.5 REFERENCES

1. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors”, July 2000.
2. NUREG/CR-5009, “Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors” February 1988
3. NRC Regulatory Issue Summary (RIS) 2006-04: Experience with Implementation of Alternative Source Terms.

Insert 15A.2.2

15A.2.2 Governing Equations

It is assumed that any activity released to the holdup system instantaneously diffuses to uniformly occupy the system volume. The following equation, from Reference 10, applies to the storage, source, transport and removal of radionuclides in compartment *i*, including the transport from the previous compartment *h* and to the next compartment *j*. This calculational model applies to both single and multiple region release models. Potential removal mechanisms considered include containment spray, natural deposition, filters, suppression pools, decay and release. The RADTRAD computer code applies the calculational model to all compartments during the event to determine the integrated offsite TEDE dose.

$$
\begin{aligned}
\frac{d}{dt}N_{n,i} = {} & \sum_{\nu=1}^{n-1}\beta_{n,\nu}N_{\nu,i}\lambda_{\nu} + S_{n,i} \\
& - \Bigg[\sum_{\substack{j=1\\ j\neq i}}^{L}\left[F_{i,j(conv)} + F_{i,j(pfil)} + \frac{Q_{i,j(supp)}}{Vol_i} + \frac{Q_{i,j(pipe)}}{Vol_i}\right] + \lambda_n + \lambda_{spr,n} + \lambda_{dep,n} \\
& + \frac{\eta_{n:i,i}}{100}F_{i,i(rfil)}\Bigg] N_{n,i} \\
& + \sum_{\substack{h=1\\ h\neq i}}^{L}\left[\left(1-\frac{\eta_{n:i,i}}{100}\right)F_{h,i(pfil)} + F_{h,i(conv)} + \frac{Q_{h,i(supp)}}{Vol_h DF_{n(supp)}} + \frac{Q_{h,i(pipe)}}{Vol_h DF_{n(pipe)}}\right] N_{n,h}
\end{aligned}
\tag{1}
$$

where

$$\lambda_n = \frac{\ln(2)}{T_n^{1/2}} \tag{2}$$

and

$T_n^{1/2}$ = half life of nuclide n [s]
$N_{n,}$ = number of atoms of nuclide n in compartment i [dimensionless]
$\beta_{n,}$ = fraction of nuclide ν that decays to nuclide n [dimensionless]
λ = radiological decay constant for nuclide n [s^{-1}]
$S_{n,}$ = source rate of nuclide n in compartment i [atoms/s]
$F_{i,j(conv)}$ = volume-normalized flow rate from compartment i to compartment *j* through a convection pathway [s^{-1}]
$F_{i,(pfil)}$ = volume-normalized filtered flow rate from compartment i to compartment *j* through a pipe [s^{-1}]
$Q_{i,(supp)}$ = volumetric flow rate from compartment i to compartment *j* through a suppression pool [ft^3/s]
$Q_{i,(pipe)}$ = volumetric flow rate from compartment i to compartment *j* through a pipe [ft^3/s]
$DF_{(supp)}$ = suppression pool decontamination factor for nuclide n [dimensionless]

$DF_{(pipe)}$ = piping decontamination factor for nuclide n [dimensionless]
$\lambda_{spr,}$ = spray removal coefficient for nuclide n [s^{-1}]
$\lambda_{dep,}$ = natural deposition removal coefficient for nuclide n [s^{-1}]
$\eta_{n:i,}$ = filter efficiency for nuclide n for a recirculating filter in compartment i [%]
$\eta_{n:h,}$ = filter efficiency for nuclide n for a filter in the pathway from compartment h to compartment i [%]
L = number of compartments defined in the model [dimensionless]
Vol_h = volume of compartment h [ft^3]

Insert 15A.4,5

15A.4 MODEL FOR RADIOLOGICAL CONSEQUENCES DUE TO RADIOACTIVE SHINE AND TRANSIT DOSE TO CONTROL ROOM OPERATORS

The shine dose is calculated based on the semi-infinite radioactive cloud surrounding the Control Room, the shine from the radioactivity inside Containment, and the shine from the Control Room Filtration Unit filter loading. The three source terms (environment, containment, and filter) from the LOCA analysis are used in combination with the MicroShield computer code to determine the total shine contribution to the Control Room dose.

The transit dose is calculated in the LOCA analysis based on additional dose points in the RADTRAD-NAI models representing the operators' path to and from the Control Room. The four components of the transit dose (inhalation, immersion, containment shine, and ground deposition) are calculated using the RADTRAD-NAI code and/or its output to determine the total transit dose contribution to the operator. Transit dose is added to the separately determined Control Room dose and control room shine dose.

15A.5 WASTE GAS DECAY TANK RUPTURE ANALYSIS RADIOLOGICAL CONSEQUENCES EVALUATION MODELS AND PARAMETERS

This section is historical as the Alternative Source Term (AST), as described in Regulatory Guide 1.183 (Reference 6), was not applied to the waste gas decay tank rupture (WGDTR) accident.

15A.5.1 GENERAL ACCIDENT PARAMETERS

This section contains the parameters used in analyzing the radiological consequences of a waste gas decay tank rupture (WGDTR). Refer to Section 15.7 for more details on WGDTR parameters. The site specific, ground-level release, short-term dispersion factors (for accidents, ground-level releases are assumed) are based on Regulatory Guide 1.XXX (Reference 7) methodology and the 0.5 percent worst-sector meteorology and these are given in Table 15A-2 (see Section 2.3.4 and the Site Addendum for additional details on meteorology). The core and gap inventories for WGDTR are given in Table 15A-3. The thyroid (via inhalation pathway), beta skin, and total-body (via submersion pathway) dose factors based on References 2 and 3 are given in Table 15A-4.

15A.5.2 OFFSITE RADIOLOGICAL CONSEQUENCES CALCULATIONAL MODELS

This section presents the models and equations used for calculating the integrated activity released to the environment, the accident flow paths, and the equations for dose calculations. One major release models are considered: a single holdup system with no internal cleanup.

15A.5.2.1 ACCIDENT RELEASE PATHWAYS

The activity release to the environment due to waste gas decay tank rupture (WGDTR) will be direct and unfiltered, with no holdup. The total activity release in this case is therefore assumed to be the initial source activity itself.

15A.5.2.2 SINGLE - REGION RELEASE MODEL

It is assumed that any activity released to the holdup system instantaneously diffuses to uniformly occupy the system volume.

The following equations are used to calculate the integrated activity released from postulated accidents.

A1(0) = initial source activity at time to, Ci

$$A_1(t) = A_1(0)e^{-\lambda_1 t} \quad (1)$$

where λ_1 = total removal constant from primary holdup system, sec^{-1}

$$\lambda_1 = \lambda_d + \lambda_{1\ell} + \lambda_r \quad (2)$$

where λ_d = decay removal constant, sec^{-1}

$\lambda_{1\ell}$ = primary holdup leak or release rate, sec^{-1}

λ_r = internal removal constant (i.e., sprays, plateout, etc.), sec^{-1}

Thus, the direct release rate to the atmosphere from the primary holdup system

$$R_u(t) = \lambda_{1\ell} A_1(t)$$

$$R_u(t) = \text{unfiltered release rate (Ci/sec)} \quad (3)$$

The integrated activity release is the integral of the above equation.

$$IAR(t) = \int_o^t R_u(t) = \int_o^t \lambda_{1\ell} A_1(o)e^{-\lambda_1 t} \quad (4)$$

This yields:

$$IAR(t) = (\lambda_{1\ell} A_1(o)/\lambda_1)(1 - e^{-\lambda_1 t}) \quad (5)$$

15A.5.2.3 OFFSITE THYROID DOSE CALCULATION MODEL

Offsite thyroid doses are calculated using the equation:

$$D_{TH} = \sum_i DCF_{Thi} \quad \sum_j (IAR)_{ij} \quad (BR)_j \quad (\chi/Q)_j \tag{14}$$

where

$(IAR)_{ij}$ = integrated activity of isotope i released* during the time interval j in Ci

and $(BR)_j$ = breathing rate during the time interval j in meter^3/second

$(\chi/Q)_j$ = offsite atmospheric dispersion factor during time interval j in second/meter^3

$(DCF)_{Thi}$ = thyroid dose conversion factor via inhalation for isotope i in rem/Ci

D_{Th} = thyroid dose via inhalation in rems

* No credit is taken for cloud depletion by ground deposition and radioactive decay during transport to the exclusion area boundary or the outer boundary of the low-population zone.

15A.5.2.4 OFFSITE TOTAL-BODY DOSE CALCULATIONAL MODEL

Assuming a semi-infinite cloud of gamma emitters, offsite total-body doses are calculated using the equation:

$$D_{TB} = \sum_i DCF_{\gamma i} \quad \sum_j (IAR)_{ij} \quad (\chi/Q)_j$$

where

$(IAR)_{ij}$ = integrated activity of isotope i released* during the jth time interval in Ci

and $(\chi/Q)_j$ = offsite atmospheric dispersion factor during time interval j in second/meter^3

$(DCF)_{\gamma i}$ = total-body gamma dose conversion factor for the ith isotope in rem-meter^3/Ci-sec

D_{TB} = total-body dose in rems

15A.5.3 CONTROL ROOM RADIOLOGICAL CONSEQUENCES CALCULATIONAL MODELS in WGDTR

The WGDTR analysis does not provide control room doses.

Insert 15A.6

6. USNRC Regulatory Guide 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000

7. DELETED.

8. EPA Federal Guidance Report No. 11, EPA-520/1-88-020, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988.

9. EPA Federal Guidance Report No. 12, EPA-402-R-93-081, "External Exposure to Radionuclides in Air, Water, and Soil," 1993.

10. NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation," December 1997.

Insert T15A-2

Note: The WGDTF (Section 15.7.1) and Radioactive Liquid Waste System Leak or Failure (Section 15.7.2) use the Atmospheric Dispersion Factors on Sheet 5

χ/Qs for Alternative Source Term (AST) Radiological Consequences

Location Type/ Event(s) (Release Location)/ Time Interval (hours)	χ/Q (Sec/meters^3)
Exclusion Area Boundary	
RWST Vent	
0 – 720	2.05E-04
Reactor Building/Other Onsite Release Locations	
0 – 720	2.00E-04
Low Population Zone	
RWST Vent	
0 – 2	6.87E-05
2 – 8	3.57E-05
8 – 24	2.57E-05
24 – 96	1.26E-05
96 – 720	4.54E-06
Reactor Building/Other Onsite Release Locations	
0 – 2	6.87E-05
2 – 8	3.42E-05
8 – 24	2.42E-05
24 – 96	1.13E-05
96 – 720	3.83E-06
Control Room	
LOCA Containment Leakage, Rod Ejection (Diffuse Containment)	
0 – Isolation	7.12E-03
Isolation – 2	7.49E-04
2 – 8	5.32E-04
8 – 24	2.29E-04
24 – 96	1.50E-04
96 – 720	9.56E-05

TABLE 15A-2 (Sheet 2)

Release Location/ Time Interval (hours)	χ/Q (Sec/meters^3)
LOCA Mini-Purge[1] & ECCS Leakage, FHA in FHB[2] (Unit Vent Exhaust)	
0 – Isolation (LOCA Mini-Purge & ECCS Leakage)	1.90E-03
0-Isolation (FHA in FHB)	2.23E-03
Isolation – 2	6.86E-04
2 – 8	5.72E-04
8 – 24	2.32E-04
24 – 96	1.42E-04
96 – 720	9.57E-05
Letdown Line Break[1] (Unit Vent Exhaust)	
0 – 2	1.90E-03
2 – 8	1.58E-03
8 – 24	6.67E-04
24 – 96	3.90E-04
96 – 720	2.29E-04
LOCA RWST Backleakage (RWST Vent)	
0 – Isolation	9.28E-04
Isolation – 2	7.47E-04
2 – 8	6.55E-04
8 – 24	2.71E-04
24 – 96	1.52E-04
96 – 720	9.17E-05
FHA in Containment (Emergency Personnel Access Hatch[3])	
0 – Isolation[3]	7.12E-03
Isolation – 2	8.61E-04
2 – 8	7.54E-04
8 – 24	3.22E-04
24 – 96	1.84E-04
96 – 720	1.43E-04

[1] In this accident, the control room never isolates, so the normal intake receptor location is used for the entire accident.
[2] The closest point of the FHB is used before isolation, since it has a higher χ/Q value.
[3] Diffuse leakage through the containment wall is used before isolation instead, since it has a higher χ/Q value.

TABLE 15A-2 (Sheet 3)

Release Location/ Time Interval (hours)	χ/Q (Sec/meters^3)
Locked Rotor, SGTR (Closest ASD[4])	
0 – Isolation[4]	1.76E-02
Isolation – 2	1.74E-03
2 – 8	1.33E-03
8 – 24	6.50E-04
24 – 96	3.62E-04
96 – 720	2.96E-04
LOOP[1] (MSSV)	
0 – 2	1.76E-02
2 – 8	1.46E-02
8 – 24	6.74E-03
24 – 96	3.81E-03
96 – 720	3.05E-03
MSLB (Closest MSL Point[5])	
0 – Isolation[5]	1.76E-02
Isolation – 2[5]	1.74E-03
2 – 8	1.56E-03
8 – 24	6.61E-04
24 – 96	3.83E-04
96 – 720	3.22E-04

[4] The closest MSSV is used before isolation instead, since it has a higher χ/Q value.
[5] The closest MSSV is used before isolation instead, since it has a higher χ/Q value. Additionally, the closest ASD is used for the first two hours instead, since it has a higher χ/Q value.

TABLE 15A-2 (Sheet 5)
LIMITING SHORT-TERM ATMOSPHERIC DISPERSION FACTORS (χ/QS) FOR WGDTR ANALYSIS

χ/Qs Applicable to: Waste Gas Decay Tank Failure (WGDTF, Section 15.7.1) and Radioactive Liquid Waste System Leak or Failure (Section 15.7.2)

Location Type/ Time Interval (hrs)	χ/Q (sec/meter^3)
Site boundary	1.5E-4
0-2	
Low-population zone	
0-8	1.5E-5
8-24	1.0E-5
24-96	4.6E-6
96-720	1.5E-6

Note that the WGDTR χ/Qs were not revised for the alternative source term (AST) analysis.

Insert T15A-3

TABLE 15A-3 CORE INVENTORY (Ci)

Isotope	Core Activity (Ci)	Isotope	Core Activity (Ci)	Isotope	Core Activity (Ci)
Kr-85	9.677E+05	Cs-134	1.405E+07	Te-125m	1.590E+05
Kr-85m	2.469E+07	Cs-136	4.531E+06	Te-133m	9.382E+07
Kr-87	4.866E+07	Cs-137	1.008E+07	Ba-141	1.587E+08
Kr-88	6.507E+07	Ba-139	1.807E+08	Ba-137m	9.583E+06
Rb-86	1.834E+05	Ba-140	1.715E+08	Pd-109	3.214E+07
Sr-89	9.252E+07	La-140	1.777E+08	Rh-106	5.544E+07
Sr-90	7.220E+06	La-141	1.594E+08	Rh-103m	1.540E+08
Sr-91	1.151E+08	La-142	1.515E+08	Tc-101	1.680E+08
Sr-92	1.235E+08	Ce-141	1.620E+08	Eu-154	5.726E+05
Y-90	7.816E+06	Ce-143	1.492E+08	Eu-155	2.368E+05
Y-91	1.214E+08	Ce-144	1.213E+08	Eu-156	2.182E+07
Y-92	1.250E+08	Pr-143	1.460E+08	La-143	1.475E+08
Y-93	1.416E+08	Nd-147	6.421E+07	Nb-97	1.662E+08
Zr-95	1.651E+08	Np-239	1.907E+09	Nb-95m	1.894E+06
Zr-97	1.651E+08	Pu-238	2.539E+05	Pm-147	1.575E+07
Nb-95	1.659E+08	Pu-239	2.836E+04	Pm-148	1.720E+07
Mo-99	1.811E+08	Pu-240	3.890E+04	Pm-149	5.969E+07
Tc-99m	1.603E+08	Pu-241	1.171E+07	Pm-151	1.881E+07
Ru-103	1.541E+08	Am-241	1.130E+04	Pm-148m	3.619E+06
Ru-105	1.080E+08	Cm-242	3.128E+06	Pr-144	1.222E+08
Ru-106	4.835E+07	Cm-244	2.900E+05	Pr-144m	1.699E+06
Rh-105	9.707E+07	I-130	1.809E+06	Sm-153	4.468E+07
Sb-127	8.907E+06	Kr-83m	1.164E+07	Y-94	1.493E+08
Sb-129	2.816E+07	Xe-138	1.697E+08	Y-95	1.591E+08
Te-127	8.717E+06	Xe-131m	1.280E+06	Y-91m	6.664E+07
Te-127m	1.443E+06	Xe-133m	6.204E+06	Br-82	3.049E+05
Te-129	2.590E+07	Xe-135m	4.176E+07	Br-83	1.154E+07
Te-129m	4.971E+06	Cs-138	1.785E+08	Br-84	2.094E+07
Te-131m	1.886E+07	Cs-134m	4.054E+06	Am-242	6.242E+06
Te-132	1.390E+08	Rb-88	6.626E+07	Np-238	3.962E+07
I-131	9.758E+07	Rb-89	8.706E+07	Pu-243	4.014E+07
I-132	1.414E+08	Sb-124	7.037E+04		
I-133	1.996E+08	Sb-125	7.418E+05		
I-134	2.240E+08	Sb-126	4.691E+04		
I-135	1.893E+08	Te-131	8.272E+07		
Xe-133	1.995E+08	Te-133	2.890E+05		
Xe-135	4.704E+07	Te-134	1.772E+08		

Table 15A-3 (Sheet 2)
WGDTR Fuel and Rod Gap Activities

	Core	
Isotope	Fuel	Gap
I-131	9.95E+7	9.95E+6
I-132	1.44E+8	1.44E+7
I-133	2.04E+8	2.04E+7
I-134	2.25E+8	2.25E+7
I-135	1.91E+8	1.91E+7
Kr-83m	1.27E+7	1.27E+6
Kr-85m	2.72E+7	2.72E+6
Kr-85	8.61E+5	2.58E+5
Kr-87	5.24E+7	5.24E+6
Kr-88	7.38E+7	7.38E+6
Kr-89	9.03E+7	9.03E+6
Xe-131m	1.12E+6	1.12E+5
Xe-133m	6.35E+6	6.35E+5
Xe-133	1.99E+8	1.99E+7
Xe-135m	3.96E+7	3.96E+6
Xe-135	4.38E+7	4.38E+6
Xe-137	1.78E+8	1.78E+7
Xe-138	1.70E+8	1.70E+7

*For WGDTR, the gap activity is assumed to be 10 percent of fuel activity for all isotopes except for Kr85; for Kr-85 it is assumed to be 30 percent of the fuel activity.

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Table 15A-4 DOSE CONVERSION FACTORS USED IN ALTERNATIVE SOURCE TERM (AST) ACCIDENT ANALYSIS

FSAR sections re-analyzed for radiological consequences as part of the replacement steam generator program and, separately, the Alternative Source Term (AST) implementation include the following accidents:

15.1.5 STEAM SYSTEM PIPING FAILURE
15.2.6 LOSS OF NONEMERGENCY AC POWER TO THE STATION AUXILIARIES
15.3.3 REACTOR COOLANT PUMP SHAFT SEIZURE (LOCKED ROTOR)
15.4.8 SPECTRUM OF ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENTS
15.6.2 BREAK IN INSTUMENT LINE OR OTHER LINES FROM REACTOR COOLANT PRESSURE BOUNDARY THAT PENETRATE CONTAINMENT
15.6.3 STEAM GENERATOR TUBE FAILURE

Additional FSAR sections re-analyzed for AST radiological consequences include the following accidents:

15.6.5 LOSS-OF-COOLANT ACCIDENTS RESULTING FROM A SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY
15.7.4 FUEL HANDLING ACCIDENTS

The following DCF's are based on Federal Guidance Reports 11 (Reference 8) and 12 (Reference 9).

Isotope	Inhalation CEDE* Sv/Bq	Inhalation CEDE* Rem/Ci	Air Submersion EDE* (Sv-m^3)/(Bq-sec)	Air Submersion EDE* Rem-m^3/Ci-sec	Contaminated Ground EDE* (Sv-m^2)/(Bq-sec)	Contaminated Ground EDE* Rem-m^2/Ci-sec
Kr-85m	N/A	N/A	7.48E-15	2.77E-02	1.52E-16	5.63E-04
Kr-85	N/A	N/A	1.19E-16	4.40E-04	2.64E-18	9.78E-06
Kr-87	N/A	N/A	4.12E-14	1.52E-01	7.32E-16	2.71E-03
Kr-88	N/A	N/A	1.02E-13	3.77E-01	1.74E-15	6.44E-03
Xe-131m	N/A	N/A	3.89E-16	1.44E-03	2.06E-17	7.63E-05
Xe-133m	N/A	N/A	1.37E-15	5.07E-03	4.07E-17	1.51E-04
Xe-133	N/A	N/A	1.56E-15	5.77E-03	4.61E-17	1.71E-04
Xe-135m	N/A	N/A	2.04E-14	7.55E-02	4.24E-16	1.57E-03
Xe-135	N/A	N/A	1.19E-14	4.40E-02	2.42E-16	8.96E-04
Xe-138	N/A	N/A	5.77E-14	2.13E-01	1.03E-15	3.81E-03
I-130	7.14E-10	2.64E+03	1.04E-13	3.85E-01	2.1E-15	7.78E-03
I-131	8.89E-09	3.29E+04	1.82E-14	6.73E-02	3.76E-16	1.39E-03
I-132	1.03E-10	3.81E+02	1.12E-13	4.14E-01	2.21E-15	8.19E-03

Table 15A-4 (Sheet 2)

Isotope	Inhalation CEDE* Sv/Bq	Inhalation CEDE* Rem/Ci	Air Submersion EDE* $(Sv\text{-}m^3)/(Bq\text{-}sec)$	Air Submersion EDE* $Rem\text{-}m^3/Ci\text{-}sec$	Contaminated Ground EDE* $(Sv\text{-}m^2)/(Bq\text{-}sec)$	Contaminated Ground EDE* $Rem\text{-}m^2/Ci\text{-}sec$
I-133	1.58E-09	5.85E+03	2.94E-14	1.09E-01	5.97E-16	2.21E-03
I-134	3.55E-11	1.31E+02	1.30E-13	4.81E-01	2.53E-15	9.37E-03
I-135	3.32E-10	1.23E+03	7.98E-14	2.95E-01	1.47E-15	5.44E-03
Cs-134	1.25E-08	4.63E+04	7.57E-14	2.80E-01	1.52E-15	5.63E-03
Cs-136	1.98E-09	7.33E+03	1.06E-13	3.92E-01	2.09E-15	7.74E-03
Cs-137	8.63E-09	3.19E+04	7.74E-18	2.86E-05	2.85E-19	1.06E-06
Cs-138	2.74E-11	1.01E+02	1.21E-13	4.48E-01	2.19E-15	8.11E-03
Rb-86	1.79E-09	6.62E+03	4.81E-15	1.78E-02	9.31E-17	3.45E-04
Te-127m	5.81E-09	2.15E+04	1.47E-16	5.44E-04	1.13E-17	4.19E-05
Te-127	8.60E-11	3.18E+02	2.42E-16	8.95E-04	5.18E-18	1.92E-05
Te-129m	6.47E-09	2.39E+04	1.55E-15	5.74E-03	3.78E-17	1.40E-04
Te-129	2.42E-11	8.95E+01	2.75E-15	1.02E-02	6.01E-17	2.23E-04
Te-131m	1.73E-09	6.40E+03	7.01E-14	2.59E-01	1.37E-15	5.07E-03
Te-132	2.55E-09	9.44E+03	1.03E-14	3.81E-02	2.28E-16	8.44E-04
Sb-127	1.63E-09	6.03E+03	3.33E-14	1.23E-01	6.76E-16	2.50E-03
Sb-129	1.74E-10	6.44E+02	7.14E-14	2.64E-01	1.38E-15	5.11E-03
Sr-89	1.12E-08	4.14E+04	7.73E-17	2.86E-04	2.27E-18	8.41E-06
Sr-90	3.51E-07	1.30E+06	7.53E-18	2.79E-05	2.84E-19	1.05E-06
Sr-91	4.49E-10	1.66E+03	3.45E-14	1.28E-01	6.77E-16	2.51E-03
Sr-92	2.18E-10	8.07E+02	6.79E-14	2.51E-01	1.25E-15	4.63E-03
Ba-139	4.64E-11	1.72E+02	2.17E-15	8.03E-03	4.59E-17	1.70E-04
Ba-140	1.01E-09	3.74E+03	8.58E-15	3.17E-02	1.8E-16	6.67E-04
Ru-103	2.42E-09	8.95E+03	2.25E-14	8.33E-02	4.63E-16	1.71E-03
Ru-105	1.23E-10	4.55E+02	3.81E-14	1.41E-01	7.69E-16	2.85E-03
Ru-106	1.29E-07	4.77E+05	0	0.00E+00	0.00E+00	0.00E+00
Rh-105	2.58E-10	9.55E+02	3.72E-15	1.38E-02	7.62E-17	2.82E-04
Mo-99	1.07E-09	3.96E+03	7.28E-15	2.69E-02	1.47E-16	5.44E-04
Tc-99m	8.80E-12	3.26E+01	5.89E-15	2.18E-02	1.21E-16	4.48E-04
Ce-141	2.42E-09	8.95E+03	3.43E-15	1.27E-02	7.38E-17	2.73E-04

Table 15A-4 (Sheet 3)

Isotope	Inhalation CEDE* Sv/Bq	Inhalation CEDE* Rem/Ci	Air Submersion EDE* $(Sv\text{-}m^3)/(Bq\text{-}sec)$	Air Submersion EDE* $Rem\text{-}m^3/Ci\text{-}sec$	Contaminated Ground EDE* $(Sv\text{-}m^2)/(Bq\text{-}sec)$	Contaminated Ground EDE* $Rem\text{-}m^2/Ci\text{-}sec$
Ce-143	9.16E-10	3.39E+03	1.29E-14	4.77E-02	2.79E-16	1.03E-03
Ce-144	1.01E-07	3.74E+05	8.53E-16	3.16E-03	2.03E-17	7.52E-05
Pu-238	1.06E-04	3.92E+08	4.88E-18	1.81E-05	8.38E-19	3.10E-06
Pu-239	1.16E-04	4.29E+08	4.24E-18	1.57E-05	3.67E-19	1.36E-06
Pu-240	1.16E-04	4.29E+08	4.75E-18	1.76E-05	8.03E-19	2.97E-06
Pu-241	2.23E-06	8.25E+06	7.25E-20	2.68E-07	1.93E-21	7.15E-09
Np-239	6.78E-10	2.51E+03	7.69E-15	2.85E-02	1.63E-16	6.04E-04
Y-90	2.28E-09	8.44E+03	1.90E-16	7.03E-04	5.32E-18	1.97E-05
Y-91	1.32E-08	4.88E+04	2.60E-16	9.62E-04	5.74E-18	2.13E-05
Y-92	2.11E-10	7.81E+02	1.30E-14	4.81E-02	2.53E-16	9.37E-04
Y-93	5.82E-10	2.15E+03	4.80E-15	1.78E-02	9.12E-17	3.38E-04
Nb-95	1.57E-09	5.81E+03	3.74E-14	1.38E-01	7.48E-16	2.77E-03
Zr-95	6.39E-09	2.36E+04	3.60E-14	1.33E-01	7.23E-16	2.68E-03
Zr-97	1.17E-09	4.33E+03	9.02E-15	3.34E-02	1.74E-16	6.44E-04
La-140	1.31E-09	4.85E+03	1.17E-13	4.33E-01	2.16E-15	8.00E-03
La-142	6.84E-11	2.53E+02	1.44E-13	5.33E-01	2.46E-15	9.11E-03
Nd-147	1.85E-09	6.85E+03	6.19E-15	2.29E-02	1.39E-16	5.15E-04
Pr-143	2.19E-09	8.10E+03	2.10E-17	7.77E-05	7.01E-19	2.60E-06
Am-241	1.20E-04	4.44E+08	8.18E-16	3.03E-03	2.75E-17	1.02E-04
Cm-242	4.67E-06	1.73E+07	5.69E-18	2.11E-05	9.56E-19	3.54E-06
Cm-244	6.70E-05	2.48E+08	4.91E-18	1.82E-05	8.78E-19	3.25E-06

*: CEDE: Committed Effective Dose Equivalent for inhalation of radioactive materials

EDE: Effective Dose Equivalent for cloudshine or submergence in a semi-infinite cloud, or shine from a contaminated ground surface

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TABLE 15A-5 (Sheet 2)

Isotope	Specific Activity µCi/gm
Class 2	
Br-83	4.00E-02
Br-84	2.17E-02
Class 3[2]	
Rb-86	7.87E-04
Rb-88	1.86E+00
Cs-134	2.31E-01
Cs-136	1.20E-01
Cs-137	1.67E-01
Class 6[2]	
Co-58	1.78E-02
Co-60	2.22E-03
Sr-89	3.25E-03
Sr-90	9.26E-05
Sr-91	0.006027
Y-90	1.11E-05
Y-91m	0.003336
Y-91	0.000593
Y-93	0.000315
Zr-95	0.000556
Nb-95	0.000464
Tc-99m	0.4448

TABLE 15A-5 (Sheet 3)

Isotope	Specific Activity µCi/gm
Ru-103	0.000417
Ru-106	9.26E-05
Rh-103m	0.000417
Rh-106	9.26E-05
Te-125m	0.000269
Te-127m	0.002591
Te-127	0.007873
Te-129m	0.01301
Te-129	0.01479
Te-131m	0.02313
Te-131	0.010197
Te-132	0.2502
Ba-137m	0.147896
Ba-140	0.002035
La-140	0.00139
Ce-141	0.000648
Ce-143	0.00037
Ce-144	0.000306
Pr-143	0.000464
Pr-144	0.000306

Insert Appendix 15B

Appendix 15B: Regulatory Guide 1.183, Revision 0, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at nuclear Power Plants" – Conformance Tables

Note: In Tables 15B-1 through 15B-7. The text shown in the "RG Position" columns is taken from Regulatory Guide 1.183. Therefore, references to footnotes, tables, and numbered references may be found in the regulatory guide.

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 Main Sections

Table 15B-2 Conformance with Regulatory Guide 1.183, Revision 0, Appendix A (Loss of Coolant Accident)

Table 15B-3 Conformance with Regulatory Guide 1.183, Revision 0, Appendix B (Fuel Handling Accident)

Table 15B-4 Conformance with Regulatory Guide 1.183, Revision 0, Appendix E (PWR Main Steam Line Break Accident)

Table 15B-5 Conformance with Regulatory Guide 1.183, Revision 0, Appendix F (PWR Steam Generator Tube Rupture Accident)

Table 15B-6 Conformance with Regulatory Guide 1.183, Revision 0, Appendix G (PWR Locked Rotor Accident)

Table 15B-7 Conformance with Regulatory Guide 1.183, Revision 0, Appendix H (PWR Rod Ejection Accident)

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 – Main Sections			
RG Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
3.	**ACCIDENT SOURCE TERM**		
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. 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. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 or ORIGEN-ARP. Core inventory factors (Ci/MWt) provided in TIDI 14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuels 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 with a core thermal power of 3565 MWt. A 2% calorimetric uncertainty is applied as a multiplier on the total core inventory resulting from the ORIGEN runs. Core design parameters (enrichment, burnup, and MTU loading) are based on cycles 19-22 to model a bounding cycle.
3.1	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 term based upon full power, core average number of fuel rods in the core. To account for differences in power level conditions. The FHA source term is across the core, radial peaking factors from the facility's core operating limits derived from the core source term, the report (COLR) or technical specifications should be applied in determining number of damaged fuel rods, and a the inventory of the damaged rods.	Conforms	For the DBA LOCA, all fuel assemblies were assumed to be affected and the core average inventory was used. A peaking factor of 1.65 was used for DBA events that do not involve the entire core (fuel handling accident, rod ejection, locked rotor), with fission product inventories for damages fuel rods determined by multiplying the total core inventory by the fraction of damaged rods.
3.1	No adjustment to the fission product conservative assembly peaking factor inventory should be made for events postulated to occur during power which corresponds to the maximum fuel operations at less than full rated power or those postulated to occur at the rod peaking factor permitted at the beginning of core life.	Conforms	No adjustments for less than full power were made in any analysis. For the fuel handling accident, 76-hours of radioactive decay after shutdown was modeled.

<table>
<tr><th colspan="4">Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 – Main Sections</th></tr>
<tr><th>RG Section</th><th>Regulatory Guide 1.183 Position</th><th>Analysis</th><th>Basis of Conformance</th></tr>
<tr><td>3.2</td><td>Release Fractions

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 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1.

Table 2
PWR Core Inventory Fraction Released Into Containment
<table><tr><th></th><th>Gap Release Phase</th><th>Early In-Vessel Phase</th><th>Total</th></tr><tr><td>Noble Gases</td><td>0.05</td><td>0.95</td><td>1.0</td></tr><tr><td>Halogens</td><td>0.05</td><td>0.35</td><td>0.4</td></tr><tr><td>Alkali Metals</td><td>0.05</td><td>0.25</td><td>0.3</td></tr><tr><td>Tellurium Metals</td><td>0.00</td><td>0.05</td><td>0.05</td></tr><tr><td>Ba, Sr</td><td>0.00</td><td>0.02</td><td>0.02</td></tr><tr><td>Noble Metals</td><td>0.00</td><td>0.0025</td><td>0.0025</td></tr><tr><td>Cerium Group</td><td>0.00</td><td>0.0005</td><td>0.0005</td></tr><tr><td>Lanthanides</td><td>0.00</td><td>0.0002</td><td>0.0002</td></tr></table></td><td>Conforms</td><td>For the LOCA event, the core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases in Table 2 were utilized.</td></tr>
</table>

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 – Main Sections			
RG Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
3.2	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. **Table 3**[11] **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 Footnote 11: The 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.	Conforms	For non-LOCA events, the fraction of the core inventory assumed to be in the gap by radionuclide group in Table 3 was utilized in conjunction with the maximum core radial peaking factor of 1.65. The control rod ejection accident was evaluated per Footnote 11 (the gap fractions are assumed to be 10% for iodines and noble gases). To account for possible damage to an assembly with high burnup and rod power and to address Footnote 11, the fuel handling accident used conservatively high gap fractions of 12% for I-131, 30% for Kr-85, 10% for all other iodines and noble gases, and 17% for Alkali Metals. These gap fractions were obtained from NUREG/CR-5009.

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 – Main Sections			
RG Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
3.3	Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs. 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. **Table 4** **LOCA Release Phases** **PWRs**: **Onset** / **Duration**; **BWRs**: **Onset** / **Duration** **Phase**: Gap Release — PWRs 30 sec / 0.5 hr; BWRs 2 min / 0.5 hr Early In-Vessel — PWRs 0.5 hr / 1.3 hr; BWRs 0.5 hr / 1.5 hr	Conforms	The Table 4 PWR onset and durations for the DBA LOCA releases were utilized in the analysis. Note that the gap release was modeled beginning at 30 seconds and ending in the first half hour in order to model the early in-vessel release beginning at 0.5 hr.
3.3	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 art 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.	Not Applicable	No additional delays in gap release were assumed for the DBA analyses.

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 – Main Sections			
RG Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
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	The Table 5 elements in each radionuclide group were utilized in OBA analyses. Note that since RADTRAD is limited to modeling 63 nuclides, certain nuclides which were deemed to be insignificant from a dose perspective were not included.
3.5	**Chemical Form** Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 95% of the iodine released should be assumed to be cesium iodide (CsI), 4.85% elemental iodine, and 0.15% 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 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.	Conforms	For releases from the reactor coolant system (RCS) to the containment, 95% of the iodine released was assumed to be cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Fission products were assumed to be in particulate form with the exception of elemental and organic iodine and noble gases,

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 – Main Sections			
RG Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
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.	Conforms	The amount of fuel damage caused by non-LOCA design basis events was analyzed. The conservatively calculated values were reflected in the rod ejection and locked rotor DBA analyses.
4.	**DOSE CALCULATION METHODOLOGY**		
4.1	**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.	Conforms	The dose calculations determined the TEDE and consider all radionuclides that are significant with regard to dose consequences. Progeny was not included in the dose calculations consistent with previously approved submittals, including: Point Beach Units 1 & 2 - April 2011 (ADAMS Accession Number ML110240054) Arkansas Nuclear One, Unit 2 - April 2011 (ADAMS Accession Number ML110980197)
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” (Reference 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” (Reference 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	CEDE Conversion factors for isotopes 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."

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 – Main Sections			
RG Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
4.1.3	For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5 x 10-4 cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8 x 10-4 cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3 x 10-4 cubic meters per second.	Conforms	The breathing rates provided were utilized to calculate the offsite dose consequences. For detem1ining a limiting 2-hour EAB dose, a constant breathing rate of 3.5 x 10-4 cubic meters per second was used.
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" (Reference 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	EDE Conversion factors for isotopes were taken from Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil."
4.15	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).	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. This was performed by the RADTRAD computer code with constant inputs for atmospheric dispersion factors and breathing rates.
4.16	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).
4.17	No correction should be made for depletion of the effluent plume by deposition on the ground.	Conforms	No correction was made for the depletion of the effluent plume by deposition on the ground.

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 – Main Sections			
RG Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
4.2	**Control Room Dose Consequences**		
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, • 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.	Conforms	The TEDE analysis considered all significant sources of radiation that would cause exposure to Control Room personnel. For Callaway, the limiting Control Room dose included: • Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility, • Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from the Control Building, • 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 Control Room recirculation filters and radioactive material in the Control Building. • Radiation shine from radioactive material deposited on the ground which operators encounter in transit to and from the Control Room.
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 nonconservative results for the control room.	Conforms	The radioactive material releases and radiation levels used in the Control Room dose analyses were determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values.
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.	Conforms	The models used to transport radioactive material into and through the Control Room, and the shielding models used to deternine radiation dose rates from external sources, were developed to provide suitably conservative estimates of the exposure to Control Room personnel.

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 – Main Sections			
RG Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
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 (Reference 3) and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Reference 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 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.	Conforms	Credit for engineered safety features that mitigate airborne radioactive material within the Control Room and Control Building were assumed as appropriate. Note that no credit for Control Room isolation was modeled for events that rely solely on radiation monitors.
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 personnel 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	Conforms	The occupancy factors and breathing rate were utilized to determine the doses to the hypothetical maximum exposed individual who is present in the Control Room. Control Room X/Q values were determined utilizing the ARCON96 computer code which does not incorporate occupancy factors. Occupancy factors were included in the RADTRAD computer code for the dose evaluations.

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 – Main Sections			
RG Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
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 (Reference 22). $$DDE_{finite} = \frac{DDE_{\infty} V^{0.338}}{1173}$$	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 the given equation. This correction was performed by the RADTRAD computer code.
4.3	**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 (Reference 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.	Conforms	Exception – The current TID-14844 accident source term will remain the licensing basis for equipment qualification and NUREG-0737 evaluations other than Control Room and Technical Support Center doses.
4.4	**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 (Reference 2) items generally reference General Design	Conforms	The DBAs were updated for consistency with the TEDE criterion in Table 6 for offsite doses and in 10 CFR 50.67(b)(2)(iii) for the Control Room and Technical Support Center doses.

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 – Main Sections

RG Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
	Criteria (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 AST, the applicable criteria should be updated for consistency with the TEDE criterion in 10 CFR 50.67(b)(2)(iii). **Accident or Case** — **EAB and LPZ Dose Criteria** LOCA — 25 rem TEDE PWR Steam Generator Tube Rupture Fuel Damage or Pre-incident Spike — 25 rem TEDE Coincident Iodine Spike — 2.5 rem TEDE PWR Main Steam Line Break Fuel Damage or Pre-incident Spike — 25 rem TEDE Coincident Iodine Spike — 2.5 rem TEDE PWR Locked Rotor Accident — 2.5 rem TEDE PWR Rod Ejection Accident — 6.3 rem TEDE Fuel Handling Accident — 6.3 rem TEDE		
5.	**ANALYSIS ASSUMPTIONS AND METHODOLOGY**		
5.1	**General Considerations**		
5.1.1	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.	Conforms	The OBA analyses were prepared, reviewed, and maintained per 10 CFR 50 Appendix B and the guidance consistent with RG 1 .183.

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 – Main Sections			
RG Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
5.1.2	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 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.	Conforms	Credit was taken for Engineered Safeguard Features with failure assumptions to maximize the calculated doses. Assumptions regarding the occurrence and timing of a loss of offsite power were also selected with the objective of maximizing the postulated radiological consequences.
5.1.3	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 nonconservative in another portion of the same analysis.	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	Licensees should ensure that analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.	Conforms	Licensee has ensured that analysis assumptions and methods are compatible with the AST 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.	Conforms	Licensee has analyzed the DBAs that are affected by the specific proposed applications of an AST.

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 – Main Sections			
RG Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
5.3	**Meteorological Assumptions** Atmospheric dispersion values *(X/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 x/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 *X/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 *X/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 *X/Q* values. Additional guidance is provided in Regulatory Guide 1.23, "Onsite Meteorological Programs" (Ref. 30). All changes in *X/Q* analysis methodology should be reviewed by the NRC staff.	Conforms	The re-calculation of atmospheric dispersion factors was performed for the EAB and LPZ using the NRC computer code PAVAN according to the guidance of RG 1.145 and for the control room and TSC intakes with new release points using the NRC computer code ARCON96 according to the guidance of RG 1.194. The meteorological data used in the calculation were collected in accordance with Callaway site-specific measurements program and RG 1.23.
6.	**Assumptions for Evaluating the Radiation Doses for Equipment Qualification**	Not applicable	An AST assessment was not performed for equipment qualification. The TID-14844 assumptions will continue to be used as the radiation dose basis for equipment qualification, radiation zone maps, and shielding calculations.

Table 15B-2 Conformance with Regulatory Guide 1.183, Revision 0 Appendix A (Loss-of-Coolant Accident)			
RG Section	**Regulatory Guide 1.183 Position**	**Analysis**	**Basis of Conformance**
Appendix A	**Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident**		
	Source Term		
Appendix A 1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	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 with a core thermal power of 3636 MWt (102% of 3565 MWt nominal power). Core design parameters (enrichment, burnup, and MTU loading) are based on the cycles 19 through 22 with conservative increases in enrichment and burnup. Margin is added to the EOC core inventory, calculated with ORIGEN-S, to account for potential core design differences in future cycles. For the DBA LOCA, all fuel assemblies were assumed to be affected and a conservatively bounding core inventory was used.
Appendix A 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% elemental iodine, and 0.15% 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 equilibrium pH in the sump stays above 7.
Appendix A 3.1	The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs … as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. .... The release into the containment … should be assumed to terminate at the end of the early in-vessel phase.	Conforms	Based on relative volumes, the release from the fuel is split between the sprayed and unsprayed regions in containment. While operation of 2 of 4 containment air coolers promotes mixing between the two regions, the exchange rate is conservatively limited to two turnovers of the unsprayed region per hour. This is in accordance with section A 3.3 (below). The release to containment is assumed to terminate at the end of the early in-vessel phase.

Table 15B-2 Conformance with Regulatory Guide 1.183, Revision 0 Appendix A (Loss-of-Coolant Accident)			
RG Section	**Regulatory Guide 1.183 Position**	**Analysis**	**Basis of Conformance**
Appendix A 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 (Reference A-1) and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Reference A-2). The latter model is incorporated into the analysis code RADTRAD (Reference 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	Natural deposition is not credited in this analysis. Spray removal coefficients are calculated in accordance with Chapter 6.5.2 of the SRP.

Table 15B-2 Conformance with Regulatory Guide 1.183, Revision 0 Appendix A (Loss-of-Coolant Accident)			
RG Section	**Regulatory Guide 1.183 Position**	**Analysis**	**Basis of Conformance**
Appendix A 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 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. This simplified model is incorporated into the analysis code RADTRAD. 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.	Conforms	In accordance with Position 5.1.2, Containment spray is - an ESF system, - classified as safety related, - required to be operable by technical specifications - powered by emergency power sources, and - automatically actuated. The mixing rate between sprayed and unsprayed regions in containment is conservatively assumed to be two turnovers of the unsprayed regions per hour. The spray removal coefficient for particulate iodine is reduced by a factor of 10 when a DF of 50 is reached.
Appendix A 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.	Not used.	

Table 15B-2 Conformance with Regulatory Guide 1.183, Revision 0 Appendix A (Loss-of-Coolant Accident)			
RG Section	**Regulatory Guide 1.183 Position**	**Analysis**	**Basis of Conformance**
Appendix A 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 (Reference 7). Analyses should consider iodine re-evolution if the suppression pool liquid pH is not maintained greater than 7.	Not Applicable	
Appendix A 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 (Reference A-1).	Not Applicable	
Appendix A 3.7	The primary containment … should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50% of the technical specification leak rate. .. . Leakage from subatmospheric containments is assumed to terminate when the containment is brought to and maintained at a subatmospheric condition as defined by technical specifications.	Conforms	From Technical Specification 5.5.16.c., the maximum allowable containment leakage rate, L_a, at P_a, shall be 0.20% of the containment air weight per day. After 24 hours, this is reduced to 0.10% per day.
Appendix A 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.	Conforms.	Only the Containment Mini-purge may be in use during power operation. 100% of the RCS maximum equilibrium activity is released to containment at initiation of the LOCA. The mini-purge isolation valves automatically close within 11 seconds, well before the onset of the gap release.

Table 15B-2 Conformance with Regulatory Guide 1.183, Revision 0 Appendix A (Loss-of-Coolant Accident)			
RG Section	**Regulatory Guide 1.183 Position**	**Analysis**	**Basis of Conformance**
Appendix A 5.0	**Assumptions on ESF System Leakage**		
Appendix A 5.1	With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Tables 1 and 2 of this guide) should be assumed to instantaneously and homogeneously mix in the primary containment sump water (in PWRs) … 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 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.	Conforms	In combination with item A 5.3 below, only iodine is released to the environment from the ESF system leakage.
Appendix A 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 (Reference 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	The operational limit of 1 gpm is doubled to 2 gpm as the basis for ECCS leakage to the Aux. Building. Instead of waiting the full 11.8 minutes as the earliest time to begin recirculation, recirculation is conservatively assumed to start just after control room isolation at 62 seconds. Isolation valve seat leakage to the RWST is analyzed as a separate case with a total of 4 gpm of back-leakage to the RWST (3 gpm below water line, 1 gpm above water line).
Appendix A 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	The release from leakage to the environment is limited to iodine.

Table 15B-2 Conformance with Regulatory Guide 1.183, Revision 0 Appendix A (Loss-of-Coolant Accident)			
RG Section	**Regulatory Guide 1.183 Position**	**Analysis**	**Basis of Conformance**
Appendix A 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, 212ºF); and h_{fg} is the heat of vaporization at 212ºF.	Conforms	With a maximum sump temperature of approximately 265˚F after the beginning of recirculation, the calculated flashing fraction is less than 10%.
Appendix A 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 analysis assumes that 10% of the iodine activity in the leakage becomes airborne and is available for filtration by the Aux. Building vent/exhaust system. The back-leakage into the RWST represents a much more controlled and well-defined environment that allows the ultimate release from the leakage to be more directly evaluated. As a very conservative treatment of RWST liquid, the analysis assumes the RG 1.183 conservative airborne release of 10% of the liquid activity for the first 24 hours of the event to cover any potential flashing or elemental iodine regeneration within the piping. After 24 hours a very conservative airborne release of 8% of the iodine is assumed despite the calculated flashing fraction of 5.5% (based on sump saturated liquid enthalpy). In accordance with NUREG/CR-5950, the temperature and pH dependent iodine re-evolution from the liquid space inside the tank and incoming sump leakage is conservatively bounded by an 8% release of sump iodine directly to the airspace. Additionally, the vent at the top of the tank restricts the ventilation rate of the vapor space.

Table 15B-2 Conformance with Regulatory Guide 1.183, Revision 0 Appendix A (Loss-of-Coolant Accident)			
RG Section	**Regulatory Guide 1.183 Position**	**Analysis**	**Basis of Conformance**
Appendix A 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 (Reference A-5) and Generic Letter 99-02 (Reference A-6).	Conforms	97% elemental and 3% organic iodine was specified as the appropriate chemical form. Compliance with RG 1.52 is presented in FSAR Table 9.4-2.
Appendix A 7.0	**Assumption on Containment Purging** 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 (Reference A-5) and Generic Letter 99-02 (Reference A-6).	Limited applicability	The containment is not purged post-LOCA, but because use of the Mini-purge system is allowed during normal operation and because of the possibility of the accident occurring at this time, operation of Mini-purge is analyzed as a possible contribution to the LOCA dose. See item A 3.8. No credit is taken for the filtration portion of the Mini-purge system because these filters are not classified as safety related. Compliance with Reg. Guide 1.52 for (other) ESF filters is presented in FSAR Table 9.4-2.

Table 15B-3 Conformance with Regulatory Guide 1.183, Revision 0 Appendix B (Fuel Handling Accident)			
RG Section	**Regulatory Guide 1.183 Position**	**Analysis**	**Basis of Conformance**
Appendix B	**Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident**		
	Source Term		
Appendix B 1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	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 with a core thermal power of 3637 MWt (102% of 3565 MWt nominal power). Core design parameters (enrichment, burnup, and MTU loading) are based on the cycles 19 through 22 with conservative increases in enrichment and burnup. Margin is added to the EOC core inventory, calculated with ORIGEN-S, to account for potential core design differences in future cycles.

Table 15B-3 Conformance with Regulatory Guide 1.183, Revision 0 Appendix B (Fuel Handling Accident)

RG Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
Appendix B 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 based upon consideration of Section 5.1.4 as discussed in Basis of Compliance	The number of damaged fuel rods is carried over from the current design basis FHA analyses. For the FHA in containment, 1.2 fuel assemblies were assumed to be affected and a conservatively bounding source term was used. For the FHA in the Fuel Handling Building, 1.0 fuel assemblies were assumed to be affected and a conservatively bounding source term was used. Per Reg. Guide 1.183, Section 5.1.4 “Applicability of Prior Licensing Basis”, the prior FHA design basis “may continue as the facility’s design basis” if it is unrelated to the use of the AST or unaffected by the AST. The prior design basis for the number of damaged fuel rods is not directly affected by implementation of the AST and is compatible with the characteristics and the revised dose calculation methodology of the AST or the TEDE criteria. Additionally, retention of the facility’s design basis for this FHA analysis does not introduce any assumption that is inconsistent with the internally consistent assumptions that comprise the AST methodology as specified in Section 5.2 of Reg Guide 1.183. Use of this prior licensing/design basis fuel failure value is in accordance with Section 5.1.4 of the main body of Reg. Guide 1.183 and therefore remains applicable as the basis for AST.
Appendix B 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 release fractions specified in Regulatory Position 3.2 are applied to damaged fuel that meets the requirements specified. For high burnup fuel, higher release fractions from NUREG/CR-5009 and Reg. Guide 1.25 are applied.

Table 15B-3 Conformance with Regulatory Guide 1.183, Revision 0 Appendix B (Fuel Handling Accident)			
RG Section	**Regulatory Guide 1.183 Position**	**Analysis**	**Basis of Conformance**
Appendix B 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	In accordance with the specified instant and complete dissociation and re-evolution of the CsI, the effective chemical form specified for the radioiodine is 99.85 percent elemental (95% CsI + 4.85% elemental) and 0.15 percent organic.
	Water Depth		
Appendix B 2	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 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).	Conforms	A minimum of 23 of water is required by Technical Specifications during fuel movement. As noted in Item 8 ofNRC Regulatory Isssue Summary (RIS) 2006-04, “Experience with Implementation of Alternative Source Terms”, the decontamination factor discussion in Reg. Guide 1.183 is misleading; the elemental iodine decontamination factor to be used is 285. The organic decontamination factor is 1.0. As described, this gives an overall effective decontamination factor of 200.
	Noble Gases		
Appendix B 3	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).	Conforms	No holdup or “scrubbing” of noble gases is credited.
	Fuel Handling Accidents within the Fuel Building		
Appendix B 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	A two hour release period is specified for activity escaping the fuel pool. Holdup and dilution within the FHB are not credited.

Table 15B-3 Conformance with Regulatory Guide 1.183, Revision 0 Appendix B (Fuel Handling Accident)

RG Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
Appendix B 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 system1 should be determined and accounted for in the radioactivity release analyses.	Not Applicable	Filtration of releases from the FHB are not credited.
Appendix B 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	Holdup and dilution in the FHB are not credited.
	Fuel Handling Accidents within Containment		
Appendix B 5.1	If the containment is isolated during fuel handling operations, no radiological consequences need to be analyzed.	Not Applicable	During refueling, open containment penetrations are allowed (under administrative controls).
Appendix B 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,1 no radiological consequences need to be analyzed.	Not Applicable	Automatic isolation of the containment is not credited.
Appendix B 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.	Conforms	A two hour release period is specified for the activity escaping the cavity pool.

Table 15B-3 Conformance with Regulatory Guide 1.183, Revision 0 Appendix B (Fuel Handling Accident)			
RG Section	**Regulatory Guide 1.183 Position**	**Analysis**	**Basis of Conformance**
Appendix B 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.	Not Applicable	ESF filtration in the containment is not credited.
Appendix B 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.	Not Applicable	Dilution and mixing within the containment are not credited.

Table 15B-4 Conformance with Regulatory Guide 1.183, Revision 0 Appendix E (PWR Main Steam Line Break)			
RG Section	**Regulatory Guide 1.183 Position**	**Analysis**	**Basis of Conformance**
Appendix E	**Assumptions for Evaluating the Radiological Consequences of a PWR Main Steam Line Break Accident**		
	Source Term		
Appendix E 1	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 regulatory 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. The fuel damage estimate should assume that the highest worth control rod is stuck at its fully withdrawn position.	Not Applicable	Assuming that the highest worth control rod is stuck at its fully withdrawn position, no fuel damage was postulated to occur during the MSLB.
Appendix E 2	If no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the technical specifications. Two cases of iodine spiking should be assumed.	Conforms	The initial RCS coolant activity is such that the DE I-133 and DE Xe-133 values are at the maximum allowed by technical specifications. Since no fuel damage occurs, two cases of iodine spiking (pre-accident and accident-initiated) were modeled.
Appendix E 2.1	A reactor transient has occurred prior to the postulated main steam line break (MSLB) and has raised the primary coolant iodine concentration to the maximum value (typically 60 μCi/gm DE I-131) permitted by the technical specifications (i.e., a pre-accident iodine spike case).	Conforms	The pre-accident iodine spike was modeled with a primary coolant iodine concentration of 60 μCi/gm DE I-131, consistent with the Technical Specification limit.
Appendix E 2.2	The primary system transient associated with the MSLB causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 μCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel gap of all fuel pins.	Conforms	The accident-initiated concurrent iodine spike was modeled with a spike factor of 500 on the appearance rate and spike duration of 8 hours. The initial activity was based on 1.0 μCi/gm DE I-131, consistent with the Technical Specification limit.

Table 15B-4 Conformance with Regulatory Guide 1.183, Revision 0 Appendix E (PWR Main Steam Line Break)			
RG Section	**Regulatory Guide 1.183 Position**	**Analysis**	**Basis of Conformance**
Appendix E 3	The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	No fuel failures are postulated. The activity from pre-accident and accident-initiated iodine spikes was modeled to be released instantaneously and homogenously throughout the primary coolant.
Appendix E 4	The chemical form of radioiodine released from the fuel should be assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.	Conforms	No fuel failures are postulated. Iodine chemical fractions for steam generator releases to the environment (97% elemental and 3% organic) were modeled in the analysis.
	Transport		
Appendix E 5.1	For facilities that have not implemented alternative repair criteria (see Reference E-1, DG-1074), the primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. For facilities with traditional generator specifications (both per generator and total of all generators), the leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.	Conforms	The Technical Specification in-identified leakage of 1 gpm was assumed to be entirely to the faulted steam generator and released directly to the environment.
Appendix E 5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of the parameter being converted. The ARC leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specifications are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft3).	Conforms	A density of 1.0 gm/cc (62.4 lbm/ft3) was used.
Appendix E 5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The primary-to-secondary leakage was terminated at 22 hours when the reactor coolant system was cooled to 212°F. The release of radioactivity from the unaffected steam generators was terminated at 7.9 hours when shutdown cooling is available (350°F) to remove decay heat.

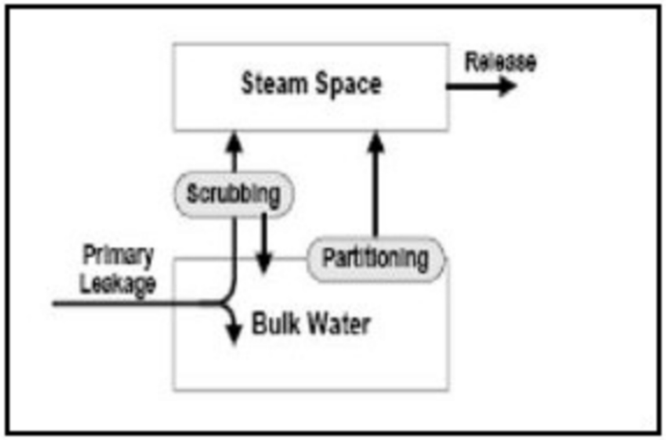
Table 15B-4 Conformance with Regulatory Guide 1.183, Revision 0 Appendix E (PWR Main Steam Line Break)			
RG Section	**Regulatory Guide 1.183 Position**	**Analysis**	**Basis of Conformance**
Appendix E 5.4	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	The noble gas radionuclides released from the primary system are released to the environment without reduction or mitigation.
Appendix E 5.5	The transport model described in this section should be utilized for iodine and particulate releases from the steam generators. This model is shown in Figure E-1 and summarized below: 	Conforms	The transport model was utilized in the analysis. See items 5.5.1 thru 5.5.4, below, for additional discussion.
Appendix E 5.5.1	A portion of the primary-to-secondary leakage will flash to vapor, based on the thermodynamic conditions in the reactor and secondary coolant. • During periods of steam generator dryout, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation. • With regard to the unaffected steam generators used for plant cooldown, the primary-to-secondary leakage can be assumed to mix with the secondary water without flashing during periods of total tube submergence.	Conforms	All primary-to-secondary leakage was assumed to be to the faulted steam generator. The faulted steam generator was assumed to blowdown to dryout conditions and the primary-to secondary leakage was modeled as a release to the environment with no mitigation. There is no primary-to-secondary leakage to the unaffected steam generators.

Table 15B-4 Conformance with Regulatory Guide 1.183, Revision 0 Appendix E (PWR Main Steam Line Break)

RG Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
Appendix E 5.5.2	The leakage that immediately flashes to vapor will rise through the bulk water of the steam generator and enter the steam space. Credit may be taken for scrubbing in the generator, using the models in NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident" (Reference E-2), during periods of total submergence of the tubes.	Not Applicable	See Item 5.5.1 above. All leakage is assumed to be to the faulted generator and released directly to the environment with no mitigation.
Appendix E 5.5.3	The leakage that does not immediately flash is assumed to mix with the bulk water.	Not Applicable	See Item 5.5.1 above. All leakage is assumed to be to the faulted generator and released directly to the environment with no mitigation.
Appendix E 5.5.4	The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 may be assumed. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.	Conforms	A partition coefficient of 100 was modeled for iodine and particulates. The moisture carryover from the steam generators for particulate retention is 0.1%. This is equivalent to a partition factor of 1000 and therefore, assuming a partition factor of 100 for particulates is conservative.
Appendix E 5.6	Operating experience and analyses have shown that for some steam generator designs, tube uncovery may occur for a short period following any reactor trip (Reference E-3). The potential impact of tube uncovery on the transport model parameters (e.g., flash fraction, scrubbing credit) needs to be considered. The impact of emergency operating procedure restoration strategies on steam generator water levels should be evaluated.	Not Applicable	See Item 5.5.1 above. All primary-to-secondary leakage was assumed to be to the faulted steam generator. The faulted steam generator was assumed to blowdown to dryout conditions and the primary-to secondary leakage was modeled as a release to the environment with no mitigation.

Table 15B-5 Conformance with Regulatory Guide 1.183, Revision 0 Appendix F (PWR Steam Generator Tube Rupture Accident)

RG Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
Appendix F	**Assumptions for Evaluating the Radiological Consequences of a PWR Steam Generator Tube Rupture Accident**		
	Source Term		

Table 15B-5 Conformance with Regulatory Guide 1.183, Revision 0 Appendix F (PWR Steam Generator Tube Rupture Accident)

RG Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
Appendix F 1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are 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	The inventory of fission products in the reactor core and available for release from the fuel into the RCS was based on the 1% fuel defects described in the FSAR and scaled up according to DE I-131 and DE Xe-133 Technical Specifications
Appendix F 2	If no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by technical specification. Two cases of iodine spiking should be assumed.	Conforms	No fuel damage is postulated. The maximum coolant activities allowed by Tech Spec 3.4.16 [1 μCi/gm DE I-131 (equilibrium), 60 μCi/gm DE I-131 (transient) and 225 μCi/gm DE XE-133] are used in the iodine spiking cases.
Appendix F 2.1	A reactor transient has occurred prior to the postulated steam generator tube rupture (SGTR) and has raised the primary coolant iodine concentration to the maximum value (typically 60 μCi/gm DE I-131) permitted by the technical specifications (i.e., a preaccident iodine spike case).	Conforms	The pre-accident iodine spike case uses the maximum coolant activities allowed by Tech Spec 3.4.16 (60 μCi/gm DE I-131 (transient) and 225 μCi/gm DE XE-133) as the source term released to the primary coolant.
Appendix F 2.2	The primary system transient associated with the SGTR causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 μCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel gap of all fuel pins.	Conforms	The concurrent iodine spike case uses the equilibrium coolant activities allowed by Tech Spec 3.4.16 (1 μCi/gm DE I-131 and 225 μCi/gm DE XE-133) and applies the 335 spiking factor to the determined release rate for the 8-hour duration as indicated. The effects of decay, purification from letdown, and leakage are accounted for over the 8 hour duration when determining the iodine release rates.
Appendix F 3	The activity released from the fuel, if any, should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	No activity released from damaged fuel. Source term is the maximum coolant activity allowed by Tech Spec 3.4.16 per Item F 2 above.
Appendix F 4	Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Conforms	The chemical form of the iodine from steam generator releases are 97% elemental and 3% organic.

Table 15B-5 Conformance with Regulatory Guide 1.183, Revision 0 Appendix F (PWR Steam Generator Tube Rupture Accident)

RG Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
Appendix F 5.1	The primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. The leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.	Conforms	The primary-to-secondary leak rate is based on Callaway FSAR SP Section 15.3.3.3.1.2, Rev. OL-18, 500 lb_m/hr (equivalent to 1gpm according to LCO 3.4.13). The leakage is apportioned equally between the SGs since all noble gases are released without reduction or mitigation and the partition coefficient in bulk water is the same for all SGs. The flashing fraction in the ruptured SG is also conservatively applied to the intact SGs to ensure the maximum activity release is modeled.
Appendix F 5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Not used.	The primary-to-secondary leak rate is based on Callaway Technical Specific 3.4.13 Basis Safety Analysis Limit, 1 gpm. FSAR SP Section 15.3.3.3.1.2, Rev. OL-18, Item d shows a leak rate of 1 gpm (500 lbm/hr).which is already in mass leak rate form.
Appendix F 5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 °C (212 °F). The release of radioactivity from the unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The release of radioactivity is based on steam release analyses which have been biased to maximize the radiological consequences. The steam release analysis documents that shutdown cooling is established at 5.9 hours for the SGTR with a stuck open ADV and at 6.4 hours with overfill.
Appendix F 5.4	The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.	Conforms	The secondary system release is through the MSSVs, ASDs, TDAFP exhaust with no credit for a partition coefficient when the secondary system release is through the condenser.
Appendix F 5.5	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	The noble gas radionuclides released in the model are released without reduction or mitigation.

Table 15B-5 Conformance with Regulatory Guide 1.183, Revision 0 Appendix F (PWR Steam Generator Tube Rupture Accident)			
RG Section	**Regulatory Guide 1.183 Position**	**Analysis**	**Basis of Conformance**
Appendix F 5.6	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms.	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E for iodine and particulates was considered as appropriate for the SGTR Accident. Additionally, it should be noted that within NL17006A060, "PG&E Letter DCL-16-124 there is a discussion on page 94 of 221 that "The effect of SG tube uncovery in intact SGs (for SGTR and non-SGTR events), has been evaluated for potential impact on dose consequences as part of a Westinghouse Owners Group (WOG) Program and demonstrated to be insignificant;…" For the SGTR (Overfill and stuck open ADV), this work for Callaway accommodates 0.83 hours of SG tube uncovery. This approach is very conservative relative to NRC acceptance of previous Westinghouse Owners Group work that shows no (complete) uncovery of SG tubes. More specifically, in response to NRC Information Notice 88-31, WOG letter OG-92-25, dated 3-31-92 summarized work on this subject and submitted WCAP-13247 to the NRC. The NRC response to the WOG submittal is a letter from Robert C. Jones, dated 3-10-93, which concludes that "the Westinghouse analyses demonstrate that the effects of partial steam generator tube uncovery on the iodine release for SGTR and non-SGTR events is negligible." In the context of more recent events, the License Amendment Request to implement AST at Diablo Canyon referenced the Westinghouse Owners Group work in saying that the effect of SG tube uncovery in SGTR and non-SGTR events has been evaluated and demonstrated to be insignificant. See pages 94 and 104 of 221 of ADAMS accession number ML17006A060. NRC approval of this approach is reflected in acceptance of (only) an iodine partition coefficient of 100 in the associated Safety Evaluation released as an attachment to the NRC letter dated 4-27-17. See pages 63 and 89 of 115 of ADAMS accession number ML17012A246.

Table 15B-6 Conformance with Regulatory Guide 1.183, Revision 0 Appendix G (PWR Locked Rotor Accident)

RG Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
Appendix G	**Assumptions for Evaluating the Radiological Consequences of a PWR Locked Rotor Accident**		
	Source Term		
Appendix G 1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this regulatory 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	The inventory of fission products in the reactor core and available for release from the fuel into the RCS was based on the maximum full power operation with a core thermal power of 3636 MWt (102% of 3565 MWt nominal power). Core design parameters (enrichment, burnup, and MTU loading) are based on the cycles 19 through 22 with conservative increases in enrichment and burnup. Margin is added to the EOC core inventory, calculated with ORIGEN-S, to account for potential core design differences in future cycles. For the DBA Locked Rotor Accident, 5% of the fuel assemblies were assumed to be affected, damaged, and 35 rods per damaged assembly were assumed to be affected by Footnote 11.
Appendix G 2	If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the main steam line break outside containment.	Conforms	For the DBA Locked Rotor Accident, 5% of the fuel assemblies were assumed to be affected, damaged, and 35 rods per damaged assembly were assumed to be affected by Footnote 11.
Appendix G 3	The activity released from the fuel should be assumed to be released instantaneously and homogeneously throughout the primary coolant.	Conforms	The radionuclides from the damaged assemblies start in the RCS compartment and not the fuel compartment, therefore being instantaneously and homogeneously distributed throughout the primary coolant.
Appendix G 4	The chemical form of radioiodine released from the fuel should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodine. Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.	Conforms	The chemical form of the iodine from steam generator releases are 97% elemental and 3%.

Table 15B-6 Conformance with Regulatory Guide 1.183, Revision 0 Appendix G (PWR Locked Rotor Accident)

RG Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
Appendix G 5.1	The primary-to-secondary leak rate in the steam generators should be assumed to be the leak-rate-limiting condition for operation specified in the technical specifications. The leakage should be apportioned between the steam generators in such a manner that the calculated dose is maximized.	Conforms	The primary-to-secondary leak rate is based on Callaway Technical Specific 3.4.13 Basis Safety Analysis Limit, 1 gpm. FSAR SP Section 15.3.3.3.1.2, Rev. OL-18, Item d shows a leak rate of 1 gpm (500 lbm/hr).
Appendix G 5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Not used.	The primary-to-secondary leak rate is based on Callaway Technaical Specific 3.4.13 Basis Safety Analysis Limit, 1 gpm. FSAR SP Section 15.3.3.3.1.2, Rev. OL-18, Item d shows a leak rate of 1 gpm (500 lb_m/hr), which is already in mass leak rate form.
Appendix G 5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 °C (212 °F). The release of radioactivity should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The release of radioactivity is based on steam release analyses which have been biased to maximize the radiological consequences. The steam release analysis documents that shutdown cooling is established at 7.3 hours for the locked rotor accident.
Appendix G 5.4	The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.	Conforms	The secondary system release is through the MSSVs, ASDs, TDAFP exhaust with no credit for a partition coefficient when the secondary system release is through the condenser.
Appendix G 5.5	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	The noble gas radionuclides released in the model are released without reduction or mitigation.

Table 15B-6 Conformance with Regulatory Guide 1.183, Revision 0 Appendix G (PWR Locked Rotor Accident)			
RG Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
Appendix G 5.6	The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms.	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E for iodine and particulates was considered as appropriate for the Locked Rotor Accident. Additionally, it should be noted that within NL17006A060, "PG&E Letter DCL-16-124 there is a discussion on page 94 of 221 that "The effect of SG tube uncovery in intact SGs (for SGTR and non-SGTR events), has been evaluated for potential impact on dose consequences as part of a Westinghouse Owners Group (WOG) Program and demonstrated to be insignificant;…" This work for Callaway accommodates 2.48 hours of SG tube uncovery. This approach is very conservative relative to NRC acceptance of previous Westinghouse Owners Group work that shows no (complete) uncovery of SG tubes. More specifically, in response to NRC Information Notice 88-31, WOG letter OG-92-25, dated 3-31-92 summarized work on this subject and submitted WCAP-13247 to the NRC. The NRC response to the WOG submittal is a letter from Robert C. Jones, dated 3-10-93, which concludes that "the Westinghouse analyses demonstrate that the effects of partial steam generator tube uncovery on the iodine release for SGTR and non-SGTR events is negligible." In the context of more recent events, the License Amendment Request to implement AST at Diablo Canyon referenced the Westinghouse Owners Group work in saying that the effect of SG tube uncovery in SGTR and non-SGTR events has been evaluated and demonstrated to be insignificant. See pages 94 and 104 of 221 of ADAMS accession number ML17006A060. NRC approval of this approach is reflected in acceptance of (only) an iodine partition coefficient of 100 in the associated Safety Evaluation released as an attachment to the NRC letter dated 4-27-17. See pages 63 and 89 of 115 of ADAMS accession number ML17012A246.

Table 15B-7 Conformance with Regulatory Guide 1.183, Revision 0 Appendix H (PWR Rod Ejection Accident)

RG 1.183 Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
Appendix H	**Assumptions for Evaluating the Radiological Consequences of a PWR Rod Ejection Accident**		
	Source Term		
Appendix H 1	Assumptions acceptable to the NRC Staff regarding core inventory are in Regulatory Position 3 of this guide. For the rod ejection 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 the assumption that 100% of the noble gases and 25% of the iodines contained in that fraction are available for release from containment. For the secondary system release pathway, 100% of the noble gases and 50% of the iodines in that fraction are released to the reactor coolant.	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 with a core thermal power of 3636 MWt (102% of 3565 MWt nominal power). Core design parameters (enrichment, burnup, and MTU loading) are based on the cycles 19 through 22 with conservative increases in enrichment and burnup. Margin is added to the EOC core inventory, calculated with ORIGEN-S, to account for potential core design differences in future cycles. For the DBA CRE, all fuel assemblies were assumed to be affected and a conservatively bounding core inventory was used.
Appendix H 2	If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the loss-of-coolant accident (LOCA), main steam line break, and steam generator tube rupture.	Conforms	As fuel damage is postulated for a rod ejection event at Callaway, this analysis is performed to confirm the dose consequences of a rod ejection are bounded by a LOCA.

Table 15B-7 Conformance with Regulatory Guide 1.183, Revision 0 Appendix H (PWR Rod Ejection Accident)

RG 1.183 Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
Appendix H 3	Two release cases are to be considered. In the first, 100% of the activity released from the fuel should be assumed to be released instantaneously and homogeneously through the containment atmosphere. In the second, 100% of the activity released from the fuel should be assumed to be completely dissolved in the primary coolant and available for release to the secondary system.	Conforms	RADTRAD does not allow the user to specify more than one NIF file for a given plant scenario file; therefore, the primary to secondary leakage case requires a third case to analyze the release of the equilibrium activity contained in the secondary coolant prior to the accident. According to Appendix A Item 7.3, the noble gases leaked form primary to secondary coolant must be released directly to environment without mitigation. RADTRAD does not have an effective mechanism for releasing the noble gases without reduction without affecting the iodine inventories; therefore, the primary to secondary leakage case is divided into two cases to analyze the full source term applicable to a primary to secondary leakage case. In total, four RADTRAD-NAI cases are analyzed to produce the dose consequences of the two release cases defined by RG 1.183 Appendix A Item 3. · Containment Leakage · Primary to Secondary Noble Gas Leakage · Primary to Secondary Non-Noble Gas Leakage (Alkalis and Halogens) · Secondary Equilibrium Initial Activity Leakage
Appendix H 4	The chemical form of radioiodine released to the containment atmosphere should be assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. If containment sprays do not actuate or are terminated prior to accumulating sump water, or if the containment sump pH is not controlled at values of 7 or greater, the iodine species should be evaluated on an individual case basis. Evaluations of pH should consider the effect of acids created during the rod ejection accident event, e.g., pyrolysis and 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 iodine chemical fractions for release to containment are 95% CsI, 4.85% elemental, and 0.15% organic. All fission products, with the exception of elemental and organic iodine and noble gases, are assumed to be in particulate form. Leakage and decay are the only removal processes modeled in containment. With no iodine being held in the sump water, sump pH has no impact.
Appendix H 5	Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Conforms	The primary to secondary leakage cases and secondary initial activity case implement the composition: 97% elemental iodine, and 3% organic iodide.
Appendix H 6.0	**Transport from Containment**		

Table 15B-7 Conformance with Regulatory Guide 1.183, Revision 0 Appendix H (PWR Rod Ejection Accident)

RG 1.183 Section	Regulatory Guide 1.183 Position	Analysis	Basis of Conformance
Appendix H 6.1	A reduction in the amount of radioactive material available for leakage from the containment that is due to natural deposition, containment sprays, recirculating filter systems, dual containments, or other engineered safety features may be taken into account. Refer to Appendix A to this guide for guidance on acceptable methods and assumptions for evaluating these mechanisms.	Conforms	Containment spray and natural deposition are available for credit at Callaway but are conservatively neglected in this analysis. Neglecting these engineered safety features provides margin for the potential that containment spray does not actuate during a rod ejection accident.
Appendix H 6.2	The containment should be assumed to leak at the leak rate incorporated in the technical specifications at peak accident pressure for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing. Leakage from subatmospheric containments is assumed to be terminated when the containment is brought to a subatmospheric condition as defined in technical specifications.	Conforms	From Technical Specification 5.5.16.c., the maximum allowable containment leakage rate, L_a, at P_a, shall be 0.20% of the containment air weight per day. After 24 hours, this is reduced to 0.10% per day.
Appendix H 7.0	**Transport from Secondary System**		
Appendix H 7.1	A leak rate equivalent to the primary-to-secondary leak rate limiting condition for operation specified in the technical specifications should be assumed to exist until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The primary-to-secondary leak rate is based on Callaway FSAR SP Section 15.4.8.3.1.2 and LCO 3.4.13, 1 gpm. The release of radioactivity is based on steam release analyses which have been biased to maximize the radiological consequences. The steam release analysis documents that shutdown cooling is established at 7.3 hours for the Rod ejection accident.
Appendix H 7.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests typically are based on cooled liquid. The facility's instrumentation used to determine leakage typically is located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft3).	Conforms	The primary-to-secondary leak rate based on Callaway FSAR SP Section 15.4.8.3.1.2, 1 gpm, is converted to 8.34 lbm/min as shown below. $1\ gpm\ (\frac{0.1337 ft^3}{gal})\ (\frac{62.4 lbm}{ft^3}) = 8.34\ lbm/min$
Appendix H 7.3	All noble gas radionuclides released to the secondary system are assumed to be released to the environment without reduction or mitigation.	Conforms	The noble gas radionuclides released in the models are released without reduction or mitigation.
Appendix H 7.4	The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms	Use of Assumptions 5.5 and 5.6 of Appendix E is documented

Chapter 15 Replacement Figures

Figure 15.6-3a: Replaced with Tab A in spreadsheet “SGTR_FSAR_Plots.xlsx”
Figure 15.6-3b: Replaced with Tab B in spreadsheet “SGTR_FSAR_Plots.xlsx”
Figure 15.6-3c: Replaced with Tab C in spreadsheet “SGTR_FSAR_Plots.xlsx”
Figure 15.6-3d: Replaced with Tab D in spreadsheet “SGTR_FSAR_Plots.xlsx”
Figure 15.6-3e: Replaced with Tab E in spreadsheet “SGTR_FSAR_Plots.xlsx”
Figure 15.6-3f: Replaced with Tab F in spreadsheet “SGTR_FSAR_Plots.xlsx”
Figure 15.6-3g: Replaced with Tab G in spreadsheet “SGTR_FSAR_Plots.xlsx”
Figure 15.6-3h: Replaced with Tab H in spreadsheet “SGTR_FSAR_Plots.xlsx”
Figure 15.6-3i: Replaced with Tab I in spreadsheet “SGTR_FSAR_Plots.xlsx”
Figure 15.6-3j: Replaced with Tab J in spreadsheet “SGTR_FSAR_Plots.xlsx”
Figure 15.6-3k: Replaced with Tab K in spreadsheet “SGTR_FSAR_Plots.xlsx”
Figure 15.6-3l: Replaced with Tab L in spreadsheet “SGTR_FSAR_Plots.xlsx”
Figure 15.6-3m: Replaced with Tab M in spreadsheet “SGTR_FSAR_Plots.xlsx”
Figure 15.6-3n: Replaced with Tab N in spreadsheet “SGTR_FSAR_Plots.xlsx”
Figure 15.6-3o: Replaced with Tab O in spreadsheet “SGTR_FSAR_Plots.xlsx”
Figure 15.6-3p: Replaced with Tab P in spreadsheet “SGTR_FSAR_Plots.xlsx”

Figure 15.6-33a: Replaced with Tab A in spreadsheet “Overfill_FSAR_Plots.xlsx”
Figure 15.6-33b: Replaced with Tab B in spreadsheet “Overfill_FSAR_Plots.xlsx”
Figure 15.6-33c: Replaced with Tab C in spreadsheet “Overfill_FSAR_Plots.xlsx”
Figure 15.6-33d: Replaced with Tab D in spreadsheet “Overfill_FSAR_Plots.xlsx”
Figure 15.6-33e: Replaced with Tab E in spreadsheet “Overfill_FSAR_Plots.xlsx”
Figure 15.6-33f: Replaced with Tab F in spreadsheet “Overfill_FSAR_Plots.xlsx”
Figure 15.6-33g: Replaced with Tab G in spreadsheet “Overfill_FSAR_Plots.xlsx”
Figure 15.6-33h: Replaced with Tab H in spreadsheet “Overfill_FSAR_Plots.xlsx”
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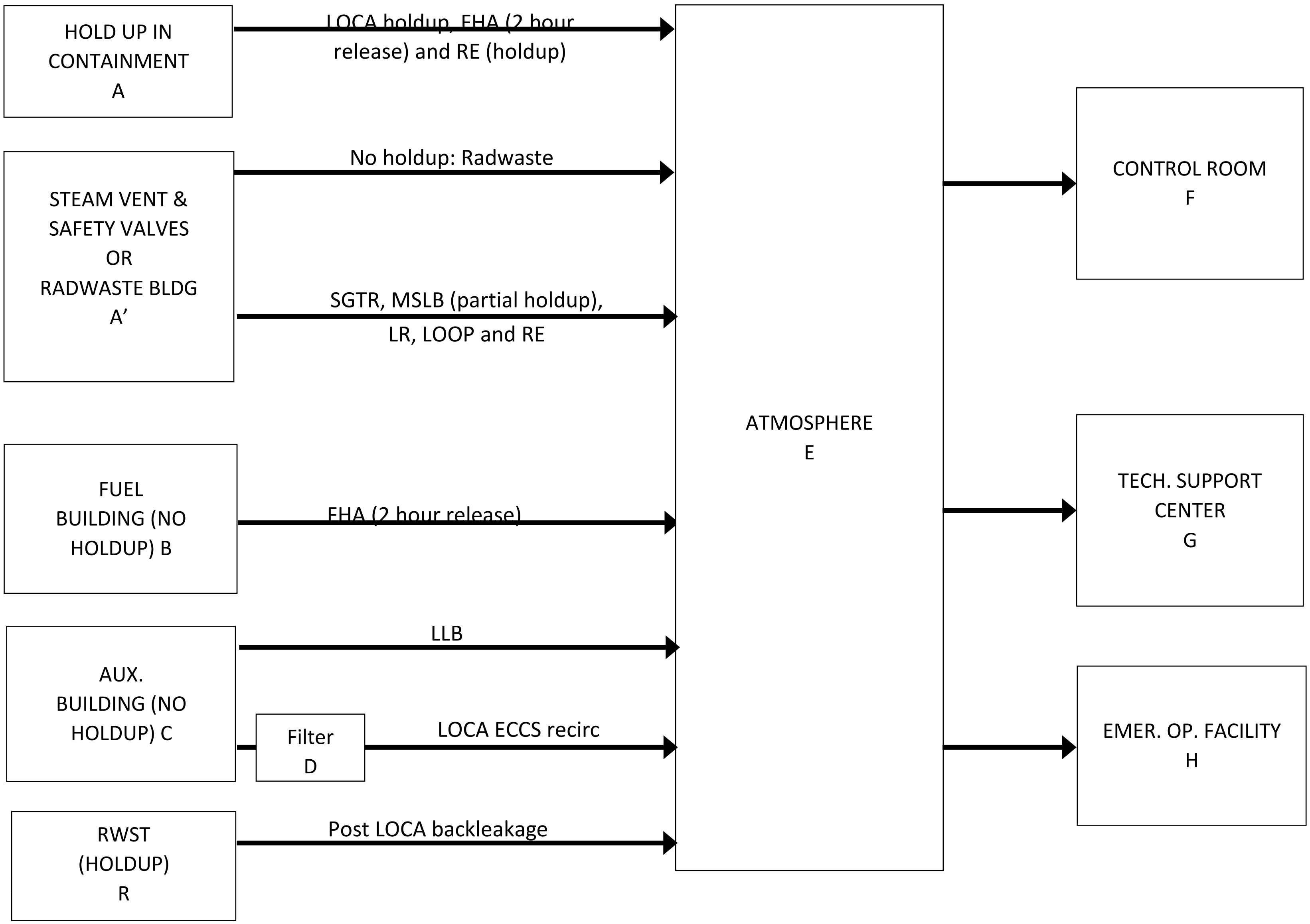
Figure 15.6-33k: Replaced with Tab K in spreadsheet “Overfill_FSAR_Plots.xlsx”
Figure 15.6-33l: Replaced with Tab L in spreadsheet “Overfill_FSAR_Plots.xlsx”
Figure 15.6-33m: Replaced with Tab M in spreadsheet “Overfill_FSAR_Plots.xlsx”
Figure 15.6-33n: Replaced with Tab N in spreadsheet “Overfill_FSAR_Plots.xlsx”

Figure 15A-1
Figure 15A-2
Figure 15A-3



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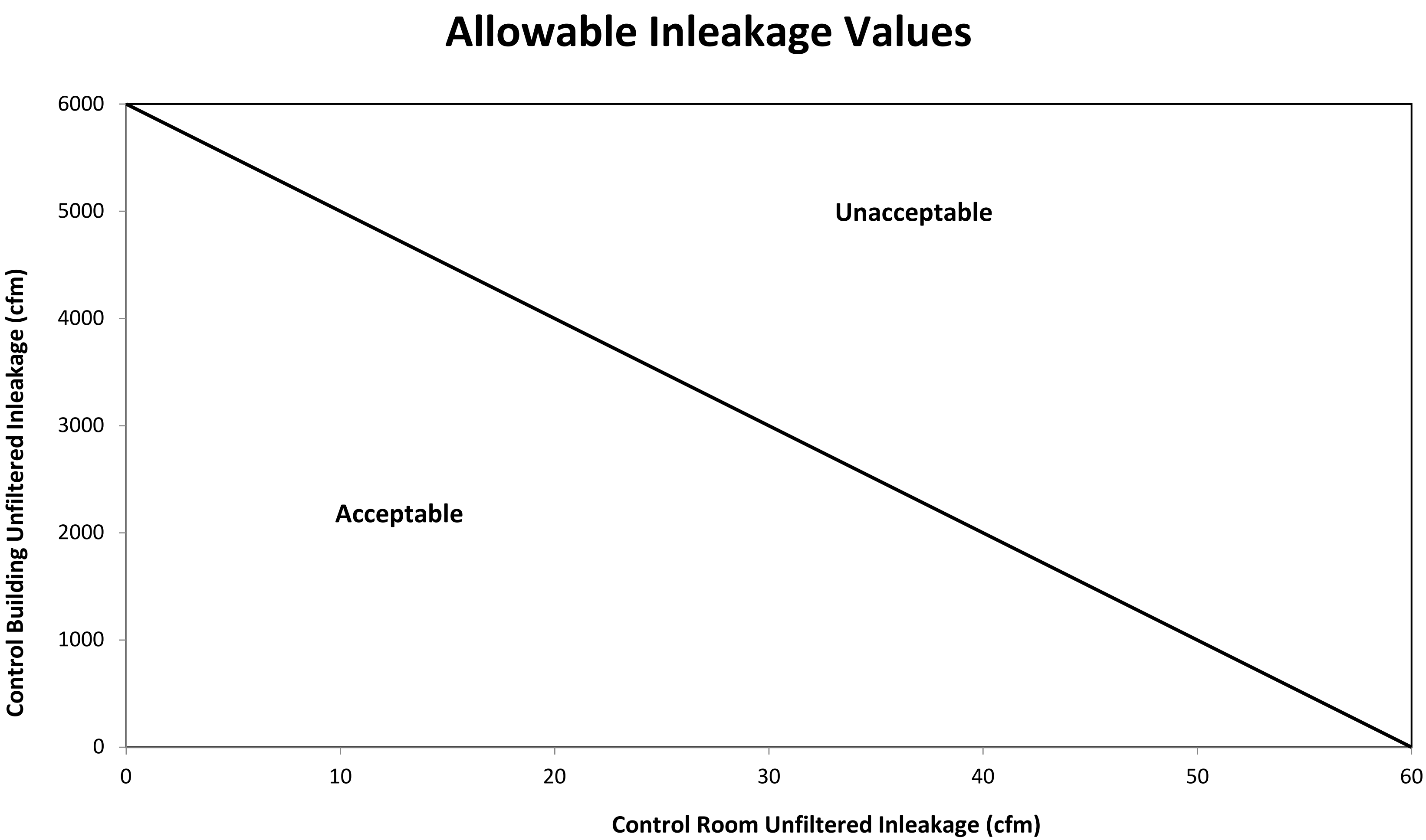
CALLAWAY PLANT

FIGURE 15A-1
RELEASE PATHWAYS



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CALLAWAY PLANT
FIGURE 15A-2 ALLOWABLE INLEAKAGE VALUES

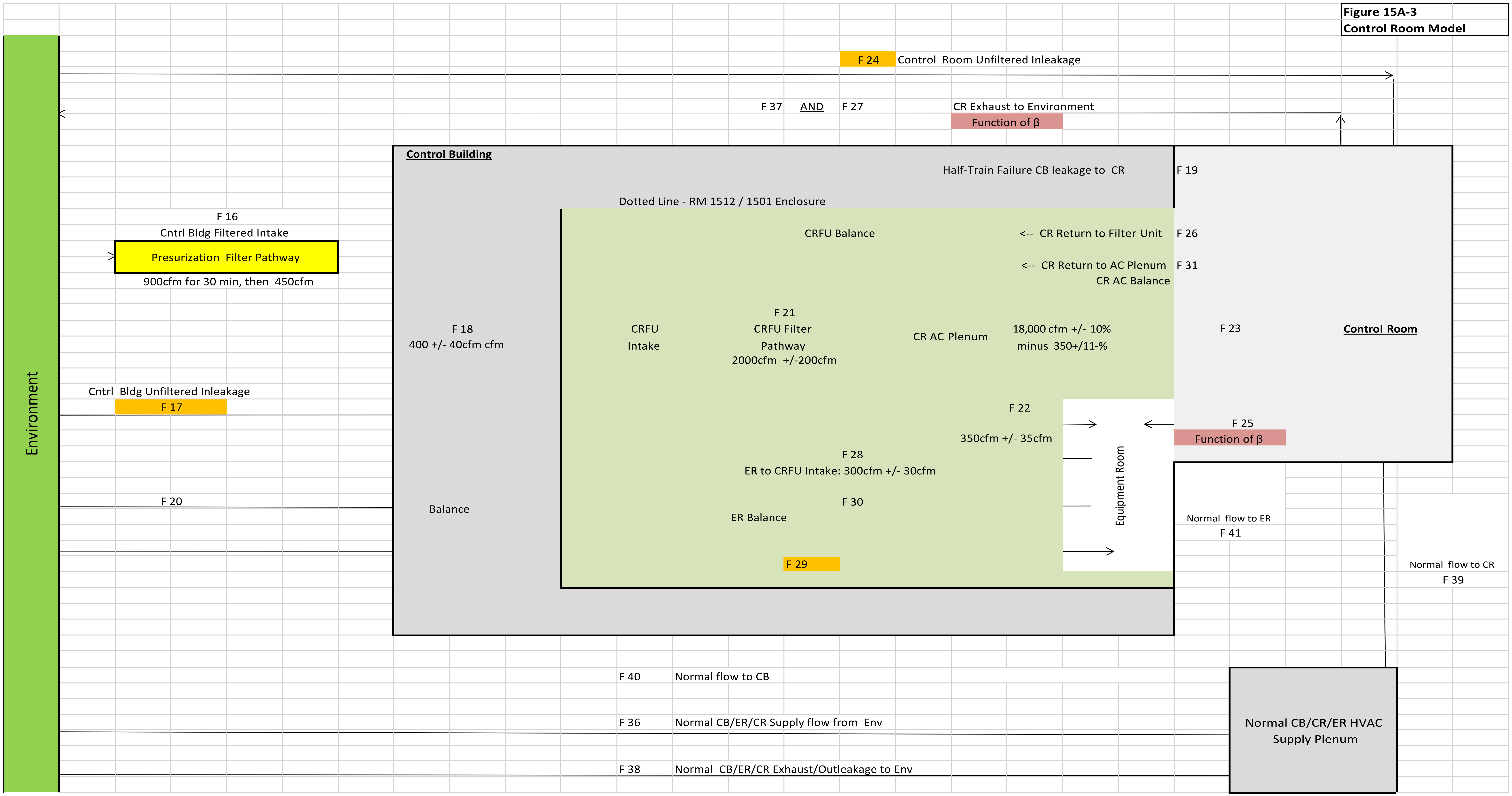
Figure 15A-3
Control Room Model
F 24
Control Room Unfiltered Inleakage
F 37
AND
F 27
CR Exhaust to Environment
Function of β
Control Building
Half-Train Failure CB leakage to CR
F 19
Dotted Line - RM 1512 / 1501 Enclosure
F 16
Cntrl Bldg Filtered Intake
Presurization Filter Pathway
900cfm for 30 min, then 450cfm
CRFU Balance
<-- CR Return to Filter Unit
F 26
<-- CR Return to AC Plenum
F 31
CR AC Balance
F 21
CRFU Filter
Pathway
2000cfm +/-200cfm
F 18
400 +/- 40cfm cfm
CRFU
Intake
CR AC Plenum
18,000 cfm +/- 10%
minus 350+/11-%
F 23
Control Room
Environment
Cntrl Bldg Unfiltered Inleakage
F 17
F 22
350cfm +/- 35cfm
F 25
Function of β
F 28
ER to CRFU Intake: 300cfm +/- 30cfm
Equipment Room
F 20
Balance
F 30
ER Balance
Normal flow to ER
F 41
F 29
Normal flow to CR
F 39
F 40
Normal flow to CB
F 36
Normal CB/ER/CR Supply flow from Env
Normal CB/CR/ER HVAC
Supply Plenum
F 38
Normal CB/ER/CR Exhaust/Outleakage to Env

Insert AddendumChapter2LOT

Insert Addendum2.3.4.4

2.3.4.4 Alternative Source Term Short-Term Diffusion Estimates

The alternative source term (AST) methodology implements the guidance of NRC Regulatory Guide (RG) 1.183 (2000).

2.3.4.4.1 Short-term (Accident) Atmospheric Dispersion Factors for the Exclusion Area Boundary and the Low Population Zone For AST

Conservative values of atmospheric dispersion factors at the exclusion area boundary (EAB) and the low population zone (LPZ) were calculated for appropriate time periods using meteorological data collected onsite. Four consecutive years of hourly measured site-specific meteorological data from January 1, 2013 to December 31st, 2016 were used in the evaluations. Meteorological data used are described in the joint frequency distributions of wind speed, wind direction, and of the atmospheric stability class presented Table 2.3-87 through 2.3-95.

2.3.4.4.1.1 Methodology

The methodology used for this calculation is consistent with Regulatory Guide (RG) 1.145 as implemented by the PAVAN computer code (Bander, 1982). Using joint frequency distributions of wind direction and wind speed by atmospheric stability, the PAVAN computer code provides relative air concentration (χ/Q) values as functions of direction for various time periods at the site boundary and LPZ. Three procedures for calculation of χ/Qs are utilized for the EAB and LPZ; a direction-dependent approach, a direction-independent approach, and an overall site χ/Q approach. The χ/Q calculations are based on the theory that material released to the atmosphere will be normally distributed (Gaussian) about the plume centerline. A straight-line trajectory is assumed between the point of release and all distances for which χ/Q values are calculated.

The theory and implementing equations employed by the PAVAN computer code are documented in Bander (1982).

2.3.4.4.1.2 Calculations/PAVAN Computer Code Input Data

The minimum EAB distance assumed for all directions is 1200 meters from the midpoint of the Unit 1 reactor building and the canceled Unit 2 reactor building. The LPZ distance is taken as 2.5 miles from the midpoint of the Unit 1 reactor building and the canceled Unit 2 reactor building in all directions.

Two PAVAN cases were executed for the offsite short term χ/Q determination. The first case (“RB”) simulated a release from the midpoint between the operating Unit 1 containment/reactor building and “disabled” Unit 2 containment/RB. The second case (“RWST”) simulated a release from the RWST. The cases differ based on the building wake effects (i.e. building area) and the release heights.

All of the releases were considered ground level releases because the highest possible release elevation is from the plant stack at 217.4 ft. From Section 1.3.2 of RG 1.145, a release is only considered a stack release if the release point is at a level higher than two and one-half times the height of adjacent solid structures. For the Callaway plant, the elevation of the top of the Unit 1 containment is 140.5 ft. Therefore, the highest possible release point is not 2.5 times higher than the adjacent containment buildings, and thus all releases were considered ground level releases. As such, the release height was set equal to 10.0 meters as required by Table 3.1 of Bander (1982). The building cross-sectional areas used for the building wake term were 1,526 m^2 for the RB release and 171 m^2 for the RWST release. The area of the containment was calculated to be conservatively small in that the height used in the area calculation was from the highest roof elevation of a nearby building to the elevation of the bottom of the containment dome. The area for the RWST release conservatively used the smaller RWST profile, since it is significantly distant from the containment structure and therefore the dominant wakes are those of the RWST.

The tower heights at which the wind speeds were measured are 10 m and 60 m above plant grade. The wind speed units are given in meters per second; therefore, the PAVAN variable UCOR was set equal to -1 to keep the wind speeds in meters per second. The maximum wind speed in each wind speed category was chosen to match the raw joint frequency distribution data, which conforms to the wind speed bins in Table 1 of RG 1.23. The maximum wind speed values are 0.5, 0.75, 1.0, 1.25, 1.5, 2.0, 3.0, 4.0, 5.0, 6.0, 8.0, and 10.0 mps. The maximum windspeed in each windspeed category was chosen to match the recommendation of RIS-2006-4 (USNRC, 2006).

2.3.4.4.1.3 Results

PAVAN computer runs for the EAB and LPZ boundary distances were performed using the data discussed previously. Per Section 4 of RG 1.145, the maximum χ/Q for each distance was determined and compared to the 5% overall site value for the boundary under consideration. The maximum EAB and LPZ χ/Qs that resulted from this comparison are provided in Table 2.3-96.

2.3.4.4.2 Short-term (Accident) Atmospheric Dispersion Factors For Onsite Receptors For AST

Conservative values of atmospheric dispersion factors to the emergency control room intake and the normal operation control room intake were calculated for appropriate time periods using meteorological data collected onsite. Four consecutive years of hourly measured site-specific meteorological data from January 1, 2013 to December 31st, 2016 were used in the evaluations. Meteorological data used are described in the joint frequency distributions of wind speed, wind direction, and of the atmospheric stability class presented in Table 2.3-87 through Table 2.3-95.

2.3.4.4.2.1 Methodology

The ARCON96 computer code is used by the USNRC staff to review licensee submittals relating to control room habitability (Ramsdell, 1995). Therefore, the ARCON96 computer code was used to determine the relative concentrations (χ/Qs) for the control room air intakes and inleakage locations.

The ARCON96 computer code uses hourly meteorological data for estimating dispersion in the vicinity of buildings to calculate relative concentrations at control room air intakes that would be exceeded no more than five percent of the time. These concentrations are calculated for averaging periods ranging from two hour to 30 days in duration.

The theory and implementing equations employed by the ACRCON96 computer code are documented in Ramsdell (1995).

2.3.4.4.2.2 Calculations/ARCON Computer Code Input Data

Four years of meteorological data (2013-2016) were used for the ARCON96 computer code runs.

A number of various release-receptor combinations were considered for the control room χ/Qs. These different cases were considered to determine the limiting release-receptor combinations for the various events. The case matrix for these combinations is provided in Table 2.3-98.

The distance and direction inputs for the ARCON96 runs may be found in Table 2.3-97. The distances were converted from feet to meters with a factor of 0.3048 m/ft. The distances in meters were then rounded down to the nearest tenth for conservatism. The elevation difference term was set to zero for each case since all elevation points are taken with respect to the same datum.

The lower and upper measurement heights for the meteorological data were entered as 10 m and 60.0 m, respectively, for each case. The mph option was selected for the wind speed units.

A ground level release was chosen for each scenario since none of the release points are 2.5 times taller than the closet solid structure as called out in Section 3.2.2 of RG 1.194 for stack releases. The top of the containment structures is at an elevation of 140.5 ft above grade. The highest release point is from the top of the plant stack at an elevation of 217.44 ft., which is not 2.5 times higher than the nearby containment structure. The vertical velocity, stack flow, and stack radius terms were all set equal to zero since each case is a ground level release. The vent release option was not selected for any of the scenarios.

The release heights and intake heights were determined as their respective elevations less the plant grade elevation of 1999.5 ft. No credit was taken for effective release height due to plume rise; therefore, for releases from the stacks, the release elevations were set equal to the stack top elevation.

For those cases that included a diffuse area source (i.e. RB wall), the release height was set at the vertical center of the projected plane (see RG 1.194, Section 3.2.4.5). The horizontal distance and direction between the release sources and receptor intakes were entered.

The only cases in this analysis that take credit for the building wake effect are the scenarios where the release is from the containment building. Some of the other scenarios have buildings between the release and receptor points, but for these cases the building wake was not credited for the sake of conservatism. Not crediting wakes was accomplished by setting the building area term equal to 0.01 m^2 as stated in Table A-2 of RG 1.194. The building area used is a conservatively determined containment cross sectional area. The width used is equal to the inside diameter of the containment building plus the thickness of the wall, while the height is taken as the distance between the top of the cylinder potion of the containment structure and the highest auxiliary building roof elevation. This building cross-sectional area is equal to 1,526 m^2.

All of the default values in the ARCON96 code were unchanged from the code default values with the following exceptions as recommended in Table A-2 of RG 1.194:

- A value of 0.2 is used for the surface roughness length, m, in lieu of the default value of 0.1, and
- A value of 4.3 is used for the averaging sector width constant, in lieu of the default value of 4.0.

The minimum wind speed was left at 0.5 m/s per the guidance instruction in Table A-2 of RG 1.194.

2.3.4.4.2.3 Results

ARCON96 computer runs for the various release points and control room intake locations were performed using the data discussed previously. Per RG 1.194, the 95th percentile χ/Q values were determined. The resulting χ/Qs are listed in Table 2.3-98.

For plants with dual CR air emergency intakes, RG 1.194, Section 3.3.2.3 states the χ/Q values may be reduced (by a factor of 2 or 4) to credit the dilution by the flow of dual intakes or by operator action to make the proper intake selection (i.e. air intake not in the direction of the wind). RG 1.194 goes onto say that this protocol should be used only if the dual intakes are in different wind direction windows. For Callaway, the two CR emergency air intakes (“A” and “B”) are two feet apart. Consequently, both emergency air intakes are within the same wind direction window. Therefore, no credit was taken for dual intake dilution.

Insert AddendumChapter2Tables

Table 2.3-87 Joint Frequency Distribution (in percent of total hours) for Stability Class A

Atmospheric Stability: Class A
Period of Record: January 1, 2013 to December 31, 2016
(based on lower wind speed instrument)

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.22																	0.00
0.50	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
0.75	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
1.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
1.25	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
1.50	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
2.00	0.00	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.01
3.00	0.00	0.01	0.01	0.01	0.00	0.00	0.04	0.07	0.05	0.04	0.02	0.03	0.02	0.03	0.01	0.00	0.34
4.00	0.01	0.01	0.02	0.01	0.02	0.01	0.13	0.08	0.03	0.06	0.11	0.07	0.08	0.08	0.09	0.03	0.84
5.00	0.01	0.03	0.01	0.01	0.01	0.01	0.07	0.06	0.06	0.06	0.10	0.04	0.08	0.13	0.11	0.05	0.85
6.00	0.02	0.01	0.01	0.00	0.00	0.02	0.03	0.05	0.03	0.08	0.07	0.04	0.07	0.08	0.07	0.02	0.61
8.00	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.01	0.04	0.03	0.01	0.01	0.03	0.01	0.03	0.03	0.22
10.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.01
26.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Total	0.04	0.07	0.05	0.03	0.03	0.04	0.28	0.27	0.21	0.27	0.32	0.19	0.28	0.33	0.31	0.13	2.88

Table 2.3-88 Joint Frequency Distribution (in percent of total hours) for Stability Class B

Atmospheric Stability: Class B
Period of Record: January 1, 2013 to December 31, 2016
(based on lower wind speed instrument)

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.22																	0.00
0.50	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
0.75	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
1.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
1.25	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
1.50	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.02
2.00	0.01	0.00	0.01	0.01	0.00	0.00	0.01	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.07
3.00	0.03	0.04	0.05	0.03	0.01	0.04	0.10	0.14	0.13	0.07	0.07	0.05	0.05	0.05	0.05	0.03	0.94
4.00	0.06	0.04	0.06	0.05	0.03	0.03	0.11	0.15	0.09	0.15	0.13	0.05	0.07	0.11	0.12	0.13	1.37
5.00	0.05	0.02	0.01	0.03	0.02	0.02	0.04	0.07	0.08	0.13	0.07	0.05	0.07	0.06	0.09	0.11	0.93
6.00	0.04	0.02	0.01	0.01	0.00	0.01	0.01	0.03	0.11	0.06	0.05	0.01	0.03	0.04	0.03	0.03	0.47
8.00	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.02	0.04	0.03	0.03	0.02	0.01	0.01	0.01	0.04	0.24
10.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.02
26.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Total	0.19	0.14	0.13	0.12	0.07	0.10	0.27	0.43	0.47	0.45	0.36	0.18	0.24	0.27	0.30	0.34	4.06

Table 2.3-89 Joint Frequency Distribution (in percent of total hours) for Stability Class C

Atmospheric Stability: Class C
Period of Record: January 1, 2013 to December 31, 2016
(based on lower wind speed instrument)

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.22																	0.00
0.50	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
0.75	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
1.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.01
1.25	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.01
1.50	0.00	0.01	0.00	0.00	0.00	0.00	0.01	0.01	0.02	0.00	0.01	0.00	0.01	0.00	0.01	0.00	0.08
2.00	0.02	0.01	0.03	0.02	0.02	0.01	0.02	0.07	0.03	0.05	0.03	0.03	0.04	0.05	0.03	0.01	0.47
3.00	0.09	0.09	0.10	0.08	0.07	0.07	0.28	0.27	0.22	0.22	0.12	0.08	0.11	0.13	0.10	0.10	2.14
4.00	0.07	0.11	0.05	0.07	0.05	0.06	0.19	0.21	0.20	0.16	0.13	0.07	0.10	0.12	0.13	0.12	1.83
5.00	0.10	0.05	0.02	0.03	0.02	0.05	0.05	0.08	0.10	0.15	0.09	0.04	0.05	0.06	0.05	0.11	1.05
6.00	0.05	0.03	0.01	0.01	0.01	0.02	0.01	0.03	0.09	0.07	0.04	0.03	0.03	0.05	0.05	0.07	0.60
8.00	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.03	0.08	0.03	0.01	0.01	0.01	0.03	0.03	0.02	0.28
10.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.02
26.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Total	0.34	0.29	0.21	0.21	0.17	0.22	0.57	0.72	0.75	0.70	0.43	0.27	0.36	0.44	0.40	0.43	6.51

Table 2.3-90 Joint Frequency Distribution (in percent of total hours) for Stability Class D

Atmospheric Stability: Class D
Period of Record: January 1, 2013 to December 31, 2016
(based on lower wind speed instrument)

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.22																	0.01
0.50	0.00	0.00	0.00	0.00	0.00	0.01	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.01	0.00	0.00	0.03
0.75	0.01	0.01	0.01	0.01	0.01	0.01	0.00	0.00	0.01	0.01	0.01	0.01	0.01	0.02	0.01	0.01	0.14
1.00	0.05	0.02	0.02	0.04	0.02	0.02	0.06	0.03	0.04	0.03	0.03	0.04	0.03	0.02	0.04	0.01	0.53
1.25	0.03	0.03	0.05	0.02	0.02	0.03	0.06	0.04	0.02	0.02	0.03	0.02	0.02	0.03	0.04	0.02	0.49
1.50	0.07	0.09	0.10	0.12	0.12	0.14	0.19	0.11	0.12	0.07	0.09	0.06	0.10	0.11	0.11	0.08	1.71
2.00	0.21	0.25	0.17	0.23	0.20	0.30	0.49	0.42	0.25	0.17	0.21	0.14	0.30	0.31	0.26	0.23	4.13
3.00	0.58	0.61	0.59	0.61	0.60	0.81	1.35	0.98	0.63	0.47	0.49	0.34	0.49	0.73	0.81	0.79	10.87
4.00	0.83	0.57	0.44	0.39	0.40	0.50	0.73	0.71	0.63	0.56	0.45	0.25	0.37	0.67	0.67	0.97	9.14
5.00	0.81	0.39	0.21	0.14	0.17	0.27	0.21	0.40	0.53	0.39	0.30	0.18	0.35	0.50	0.56	0.58	6.00
6.00	0.45	0.16	0.07	0.05	0.05	0.05	0.03	0.18	0.38	0.21	0.17	0.09	0.25	0.30	0.33	0.36	3.13
8.00	0.19	0.06	0.06	0.03	0.01	0.01	0.00	0.09	0.52	0.14	0.10	0.08	0.20	0.17	0.14	0.23	2.04
10.00	0.01	0.00	0.02	0.00	0.00	0.00	0.00	0.00	0.05	0.01	0.01	0.01	0.02	0.00	0.00	0.03	0.16
26.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.01
Total	3.25	2.19	1.76	1.63	1.61	2.15	3.12	2.96	3.18	2.07	1.90	1.23	2.14	2.88	2.99	3.32	38.37

Table 2.3-91 Joint Frequency Distribution (in percent of total hours) for Stability Class E

Atmospheric Stability: Class E
Period of Record: January 1, 2013 to December 31, 2016
(based on lower wind speed instrument)

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.22																	0.07
0.50	0.01	0.01	0.01	0.01	0.03	0.01	0.03	0.01	0.01	0.01	0.00	0.01	0.01	0.01	0.01	0.01	0.17
0.75	0.02	0.01	0.02	0.02	0.03	0.02	0.03	0.02	0.02	0.03	0.02	0.02	0.02	0.03	0.03	0.02	0.35
1.00	0.07	0.06	0.05	0.06	0.07	0.06	0.11	0.09	0.06	0.05	0.06	0.05	0.07	0.10	0.09	0.04	1.10
1.25	0.03	0.04	0.05	0.04	0.03	0.05	0.12	0.05	0.05	0.03	0.05	0.07	0.05	0.11	0.09	0.06	0.94
1.50	0.08	0.13	0.11	0.12	0.11	0.17	0.43	0.15	0.09	0.06	0.10	0.12	0.12	0.17	0.14	0.12	2.22
2.00	0.26	0.19	0.23	0.23	0.27	0.36	0.93	0.51	0.22	0.16	0.20	0.23	0.17	0.27	0.36	0.25	4.86
3.00	0.45	0.36	0.34	0.27	0.38	0.48	1.33	1.81	0.88	0.61	0.53	0.38	0.52	0.48	0.52	0.59	9.92
4.00	0.21	0.10	0.07	0.08	0.13	0.19	0.39	0.94	1.17	0.65	0.43	0.24	0.43	0.20	0.19	0.27	5.70
5.00	0.03	0.02	0.02	0.01	0.02	0.04	0.07	0.38	0.71	0.29	0.13	0.09	0.15	0.09	0.05	0.06	2.18
6.00	0.00	0.00	0.01	0.00	0.00	0.02	0.01	0.15	0.41	0.13	0.03	0.01	0.05	0.03	0.03	0.02	0.91
8.00	0.00	0.00	0.01	0.00	0.00	0.00	0.01	0.07	0.19	0.05	0.01	0.01	0.01	0.00	0.00	0.00	0.35
10.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.01	0.00	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.02
26.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Total	1.16	0.92	0.91	0.82	1.07	1.40	3.46	4.19	3.82	2.09	1.57	1.24	1.62	1.50	1.52	1.45	28.82

Table 2.3-92 Joint Frequency Distribution (in percent of total hours) for Stability Class F

Atmospheric Stability: Class F
Period of Record: January 1, 2013 to December 31, 2016
(based on lower wind speed instrument)

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.22																	0.11
0.50	0.03	0.01	0.01	0.02	0.03	0.03	0.03	0.02	0.01	0.01	0.01	0.01	0.02	0.01	0.02	0.01	0.27
0.75	0.02	0.02	0.03	0.04	0.03	0.05	0.07	0.05	0.04	0.02	0.02	0.01	0.04	0.02	0.02	0.03	0.48
1.00	0.06	0.08	0.09	0.07	0.08	0.11	0.19	0.11	0.06	0.03	0.05	0.06	0.07	0.11	0.07	0.08	1.33
1.25	0.07	0.05	0.06	0.06	0.07	0.07	0.21	0.10	0.04	0.04	0.05	0.05	0.04	0.07	0.06	0.01	1.04
1.50	0.07	0.09	0.14	0.14	0.11	0.15	0.40	0.25	0.10	0.11	0.08	0.08	0.05	0.17	0.19	0.06	2.20
2.00	0.17	0.11	0.11	0.11	0.08	0.13	0.42	0.65	0.27	0.19	0.21	0.09	0.10	0.19	0.25	0.14	3.20
3.00	0.09	0.07	0.05	0.05	0.03	0.04	0.27	1.57	0.67	0.34	0.35	0.11	0.10	0.13	0.11	0.23	4.22
4.00	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.27	0.35	0.18	0.08	0.02	0.02	0.00	0.00	0.01	0.95
5.00	0.00	0.00	0.00	0.00	0.00	0.00	0.01	0.01	0.04	0.02	0.00	0.01	0.00	0.00	0.00	0.00	0.08
6.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
8.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
10.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
26.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Total	0.51	0.43	0.48	0.49	0.43	0.58	1.60	3.04	1.60	0.93	0.84	0.43	0.44	0.69	0.71	0.58	13.90

Table 2.3--93 Joint Frequency Distribution (in percent of total hours) for Stability Class G

Atmospheric Stability: Class G
Period of Record: January 1, 2013 to December 31, 2016
(based on lower wind speed instrument)

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.22																	0.19
0.50	0.02	0.05	0.03	0.02	0.02	0.01	0.04	0.04	0.04	0.01	0.01	0.01	0.02	0.02	0.01	0.02	0.38
0.75	0.03	0.04	0.03	0.03	0.03	0.04	0.06	0.05	0.05	0.03	0.02	0.03	0.02	0.01	0.01	0.03	0.51
1.00	0.10	0.09	0.09	0.05	0.05	0.04	0.12	0.17	0.08	0.03	0.05	0.03	0.03	0.07	0.08	0.04	1.13
1.25	0.04	0.05	0.04	0.03	0.01	0.02	0.03	0.11	0.04	0.02	0.02	0.02	0.01	0.05	0.06	0.03	0.60
1.50	0.07	0.07	0.10	0.03	0.02	0.02	0.08	0.17	0.07	0.06	0.03	0.03	0.02	0.07	0.08	0.09	1.00
2.00	0.07	0.07	0.04	0.02	0.01	0.01	0.05	0.23	0.11	0.05	0.04	0.01	0.02	0.04	0.08	0.08	0.91
3.00	0.01	0.01	0.00	0.00	0.00	0.00	0.02	0.33	0.09	0.04	0.02	0.00	0.01	0.01	0.05	0.03	0.62
4.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.08	0.03	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.12
5.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
6.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
8.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
10.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
26.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Total	0.35	0.38	0.33	0.17	0.14	0.14	0.40	1.19	0.52	0.24	0.19	0.12	0.13	0.27	0.38	0.32	5.47

Table 2.3-94 Wind Direction Occurrence Frequency

Period of Record: January 1, 2013 to December 31, 2016
(based on lower wind speed instrument)

Wind Direction	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
Frequency	5.85	4.42	3.88	3.47	3.52	4.63	9.69	12.78	10.55	6.75	5.61	3.66	5.21	6.39	6.62	6.57

Table 2.3-95 Wind Speed Occurrence Frequency

Period of Record: January 1, 2013 to December 31, 2016
(based on lower wind speed instrument)

Maximum Wind Speed (m/s)	0.22	0.50	0.75	1.00	1.25	1.50	2.00	3.00	4.00	5.00	6.00	8.00	10.00	26.00
Frequency	0.39	0.85	1.48	4.1	3.09	7.23	13.66	29.06	19.95	11.09	5.72	3.13	0.24	0.02

Table 2.3-96 Exclusion Area Boundary and Low Population Zone χ/Qs

Exclusion Area Boundary	
RWST Vent	
0 – 720	2.05E-04
Reactor Building/Other Onsite Release Locations	
0 – 720	2.00E-04
Low Population Zone	
RWST Vent	
0 – 2	6.87E-05
2 – 8	3.57E-05
8 – 24	2.57E-05
24 – 96	1.26E-05
96 – 720	4.54E-06
Reactor Building/Other Onsite Release Locations	
0 – 2	6.87E-05
2 – 8	3.42E-05
8 – 24	2.42E-05
24 – 96	1.13E-05
96 – 720	3.83E-06

Table 2.3-97 Release/receptor pairs and inputs for χ/Q calculation

Release Point	Receptor Point	Horizontal Distance (m)	Release Height Above Plant Grade (m)	Intake Height Above Plant Grade (m)	Direction Looking at Source From Receptor (° from True North)
Stack/Plant Vent	'B' CB intake (Emergency)	70.9	66.3	5.5	338
Stack/Plant Vent	CB intake (Normal)	31.9	66.3	22.5	8
RWST	'B' CB intake (Emergency)	93.8	16.5	5.5	19
RWST	CB intake (Normal)	82.7	16.5	22.5	49
FHB Closest Point	'B' CB intake (Emergency)	73.6	5.5	5.5	2
FHB Closest Point	CB intake (Normal)	52.2	22.5	22.5	41
Closest ASD	Midpoint between Intakes	60.7	35.5	5.5	312
Closest ASD	CB intake (Normal)	15.9	35.5	22.5	292
Closest MSSV	CB intake (Normal)	14.5	34.8	22.5	316
Closest Main Steam Line Point	Midpoint between Intakes	60.5	12.0	5.5	316
Closest Main Steam Line Point	CB intake (Normal)	14.5	12.0	22.5	316
Emergency Personnel Access Hatch	'B' CB intake (Emergency)	88.4	4.3	5.5	328
Containment (Diffuse) (Note 1)	CB intake (Normal)	9.3	34.8	22.5	8
Containment (Diffuse)	CB Intake (Emergency) Midpoint	47.6	34.8	5.5	339

Notes:

1. As stated in Item 3.4 of RG 1.194, ARCON96 should not be used when the horizontal distance is less than 10 meters. A conservative estimate was used by running two additional cases with the horizontal distance set arbitrarily to 10 meters and 20 meters, and extrapolating using a $1/r^2$ shape.

Table 2.3-98 Calculated χ/Q values and associated events

Event(s) (Receptor Location) / Release Source / Time (hours)	χ/Q (sec/m³)
Control Room	
LOCA Containment Leakage, Rod Ejection (Diffuse Containment)	
0 – Isolation	7.12E-03
Isolation – 2	7.49E-04
2 – 8	5.32E-04
8 – 24	2.29E-04
24 – 96	1.50E-04
96 – 720	9.56E-05
LOCA Mini-Purge[1] & ECCS Leakage (Unit Vent Exhaust)	
0 – Isolation	1.90E-03
Isolation – 2	6.86E-04
2 – 8	5.72E-04
8 – 24	2.32E-04
24 – 96	1.42E-04
96 – 720	9.57E-05
Letdown Line Break[1] (Unit Vent Exhaust)	
0 – 2	1.90E-03
2 – 8	1.58E-03
8 – 24	6.67E-04
24 – 96	3.90E-04
96 – 720	2.29E-04
FHA in the Fuel Handling Building (Unit Vent Exhaust[2])	
0 – Isolation	2.23E-03
Isolation – 2	6.86E-04
2 – 8	5.72E-04
8 – 24	2.32E-04
24 – 96	1.42E-04
96 – 720	9.57E-05

[1]: In this accident, the control room never isolates, so the normal intake receptor location is used for the entire accident.

[2]: The Fuel Handling Buliding closest point is used before isolation instead, since it has a higher χ/Q value.

LOCA RWST Backleakage (RWST Vent)	
0 – Isolation	9.28E-04
Isolation – 2	7.47E-04
2 – 8	6.55E-04
8 – 24	2.71E-04
24 – 96	1.52E-04
96 – 720	9.17E-05
FHA in Containment (Emergency Personnel Access Hatch[3])	
0 – Isolation[3]	7.12E-03
Isolation – 2	8.61E-04
2 – 8	7.54E-04
8 – 24	3.22E-04
24 – 96	1.84E-04
96 – 720	1.43E-04
Locked Rotor, SGTR (Closest ASD[4])	
0 – Isolation[4]	1.76E-02
Isolation – 2	1.74E-03
2 – 8	1.33E-03
8 – 24	6.50E-04
24 – 96	3.62E-04
96 – 720	2.96E-04
LOOP[1] (MSSV)	
0 – 2	1.76E-02
2 – 8	1.46E-02
8 – 24	6.74E-03
24 – 96	3.81E-03
96 – 720	3.05E-03
MSLB (Closest MSL Point[5])	
0 – Isolation[5]	1.76E-02
Isolation – 2[5]	1.74E-03
2 – 8	1.56E-03
8 – 24	6.61E-04
24 – 96	3.83E-04
96 – 720	3.22E-04

[3]: Diffuse leakage through the containment wall is used before isolation instead, since it has a higher χ/Q value.

[4]: The closest MSSV is used before isolation instead, since it has a higher χ/Q value.

[5]: The closest MSSV is used before isolation instead, since it has a higher χ/Q value to the normal intake stack. The closest ASD is used for the first two hours after isolation, since it has a higher χ/Q value for that time frame.