15.0 ACCIDENT AND TRANSIENT ANALYSES
TABLE OF CONTENTS

			Page
15.0 ACCIDENT AND TRANSIENT ANALYSES			15.0-1
15.0.1	Frequency Classification		15.0-1
15.0.2	Transients and Accidents Analyzed		15.0-1
	15.0.2.1	Anticipated Operational Occurrences	15.0-1
	15.0.2.2	Design Basis Accidents	15.0-3
	15.0.2.3	Radiological Assessments of Design Basis Accidents	15.0-3
	15.0.2.4	References	15.0-4
15.1	INCREASE IN HEAT REMOVAL BY THE REACTOR COOLANT SYSTEM		15.1-1
15.1.1	Decrease in Feedwater Temperature		15.1-1
	15.1.1.1	Identification of Causes and Frequency Classification	15.1-1
	15.1.1.2	Sequence of Events and System Operation	15.1-2
	15.1.1.3	Barrier Performance	15.1-2
	15.1.1.4	Radiological Consequences	15.1-2
15.1.2	Increase in Feedwater Flow		15.1-2
	15.1.2.1	Identification of Causes and Frequency Classification	15.1-3
	15.1.2.2	Sequence of Events and System Operation	15.1-3
	15.1.2.3	Core and System Performance	15.1-3
	15.1.2.4	Barrier Performance	15.1-4
	15.1.2.5	Radiological Consequences	15.1-4
15.1.3	Increase in Steam Flow		15.1-4
	15.1.3.1	Identification of Causes	15.1-4
	15.1.3.2	Core and System Performance	15.1-5
	15.1.3.3	Barrier Performance	15.1-5
	15.1.3.4	Radiological Consequences	15.1-5
15.2	DECREASE IN HEAT REMOVAL BY THE REACTOR COOLANT SYSTEM		15.2-1
15.2.1	Steam Pressure Regulator Malfunction		15.2-1
	15.2.1.1	Identification of Causes	15.2-1
	15.2.1.2	Sequence of Events and System Operation	15.2-1
	15.2.1.3	Core and System Performance	15.2-1
	15.2.1.4	Barrier Performance	15.2-1
	15.2.1.5	Radiological Consequences	15.2-1
15.2.2	Load Rejection		15.2-2
	15.2.2.1	Load Rejection (Generator Trip) Without Bypass	15.2-2
	15.2.2.2	Load Rejection With Bypass	15.2-4
15.2.3	Turbine Trip		15.2-4
	15.2.3.1	Turbine Trip Without Bypass	15.2-5
	15.2.3.2	Turbine Trip with Bypass	15.2-6

	15.2.4	Inadvertent Closure of Main Steam Isolation Valves		15.2-7
		15.2.4.1	Identification of Causes	15.2-7
		15.2.4.2	Sequence of Events and System Operation	15.2-7
		15.2.4.3	Core and System Performance	15.2-7
		15.2.4.4	Barrier Performance	15.2-7
		15.2.4.5	Radiological Consequences	15.2-7
	15.2.5	Loss of Condenser Vacuum		15.2-8
		15.2.5.1	Identification of Causes	15.2-8
		15.2.5.2	Sequence of Events and System Operation	15.2-8
		15.2.5.3	Core and System Performance	15.2-8
		15.2.5.4	Barrier Performance	15.2-8
		15.2.5.5	Radiological Consequences	15.2-8
	15.2.6	Loss of Offsite AC Power		15.2-9
	15.2.7	Loss of Normal Feedwater Flow		15.2-9
		15.2.7.1	Identification of Causes	15.2-9
		15.2.7.2	Sequence of Events and System Operation	15.2-9
		15.2.7.3	Core and System Performance	15.2-9
		15.2.7.4	Barrier Performance	15.2-10
		15.2.7.5	Radiological Consequences	15.2-10
	15.2.8	References		15.2-11
15.3	DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE			15.3-1
	15.3.1	Single and Multiple Recirculation Pump Trips		15.3-1
		15.3.1.1	Trip of Both Drive Motors	15.3-2
		15.3.1.2	Trip of One Drive Motor	15.3-2
		15.3.1.3	Trip of One Recirculation Pump Motor	15.3-3
		15.3.1.4	Trip of Two Recirculation Pump Motors	15.3-3
	15.3.2	Recirculation Flow Controller Malfunctions		15.3-3
	15.3.3	Recirculation Pump Shaft Seizure		15.3-4
	15.3.4	Recirculation Pump Shaft Break		15.3-5
	15.3.5	Jet Pump Malfunction		15.3-5
	15.3.6	Transients During Single Loop Operation		15.3-5
	15.3.7	References		15.3-7
15.4	REACTIVITY AND POWER DISTRIBUTION ANOMALIES			15.4-1
	15.4.1	Control Rod Removal Error During Refueling		15.4-1
		15.4.1.1	Identification of Causes and Frequency Classification	15.4-1
		15.4.1.2	Sequence of Events and Systems Operation	15.4-1
		15.4.1.3	Core and System Performance	15.4-1a
		15.4.1.4	Barrier Performance	15.4-1a
		15.4.1.5	Radiological Consequences	15.4-1a
	15.4.2	Rod Withdrawal Error — At Power		15.4-1a
		15.4.2.1	Identification of Causes and Frequency Classification	15.4-1a
		15.4.2.2	Sequence of Events and System Operation	15.4-2
		15.4.2.3	Core and System Performance	15.4-2
		15.4.2.4	Barrier Performance	15.4-3
		15.4.2.5	Radiological Consequences	15.4-3
	15.4.3	Control Rod Maloperation		15.4-3

	15.4.4	Startup of Idle Recirculating Loop at Incorrect Temperature	15.4-3
		15.4.4.1 Identification of Causes	15.4-3
		15.4.4.2 Sequence of Events and System Operation	15.4-4
		15.4.4.3 Core and System Performance	15.4-5
		15.4.4.4 Barrier Performance	15.4-5
		15.4.4.5 Radiological Consequences	15.4-5
	15.4.5	Recirculation Loop Flow Controller Failure With Increasing Flow	15.4-5
		15.4.5.1 Identification of Causes	15.4-6
		15.4.5.2 Sequence of Events and System Operation	15.4-6
		15.4.5.3 Core and System Performance	15.4-6a
		15.4.5.4 Barrier Performance	15.4-7
		15.4.5.5 Radiological Consequences	15.4-7
	15.4.6	Chemical and Volume Control System Malfunction	15.4-7
	15.4.7	Mislocated Fuel Assembly Accident	15.4-7
	15.4.8	Misoriented Fuel Assembly Accident	15.4-7
		15.4.8.1 Identification of Causes and Frequency Classification	15.4-8
		15.4.8.2 Sequence of Events and System Operation	15.4-8
		15.4.8.3 Core and System Performance	15.4-8
		15.4.8.4 Barrier Performance	15.4-9
		15.4.8.5 Radiological Consequences	15.4-9
	15.4.9	Control Rod Ejection Accidents (PWR)	15.4-9
	15.4.10	Control Rod Drop Accident	15.4-9
		15.4.10.1 Identification of Causes and Frequency Classification	15.4-10
		15.4.10.2 Sequence of Events and System Operation	15.4-11
		15.4.10.3 Core and System Performance	15.4-12
		15.4.10.4 Barrier Performance	15.4-13
		15.4.10.5 Radiological Consequences	15.4-13
	15.4.11	Thermal Hydraulic Instability Transient	15.4-18
		15.4.11.1 Identification of Causes and Frequency Classification	15.4-18
		15.4.11.2 Sequence of Events and System Operation	15.4-18
		15.4.11.3 Core and System Performance	15.4-18a
		15.4.11.4 Barrier Performance	15.4-18a
		15.4.11.5 Radiological Consequences	15.4-18a
	15.4.12	References	15.4-19
15.5	INCREASE IN REACTOR COOLANT INVENTORY		15.5-1
	15.5.1	Inadvertent Initiation of High Pressure Coolant Injection During Power Operation	15.5-1
		15.5.1.1 Identification of Causes and Frequency Classification	15.5-1
		15.5.1.2 Sequence of Events and Systems Operation	15.5-1
		15.5.1.3 Core and System Performance	15.5-2
		15.5.1.4 Barrier Performance	15.5-2
		15.5.1.5 Radiological Consequences	15.5-2
	15.5.2	References	15.5-3

			Page
15.6	DECREASE IN REACTOR COOLANT INVENTORY		15.6-1
15.6.1	Inadvertent Opening of a Safety Valve, Relief Valve, or Safety Relief Valve		15.6-1
	15.6.1.1	Identification of Causes	15.6-1
	15.6.1.2	Sequence of Events and Systems Operation	15.6-2
	15.6.1.3	Core and System Performance	15.6-2
	15.6.1.4	Barrier Performance	15.6-2
	15.6.1.5	Radiological Consequences	15.6-3
15.6.2	Break in Reactor Coolant Pressure Boundary Instrument Line Outside Containment		15.6-3
	15.6.2.1	Identification of Causes	15.6-3
	15.6.2.2	Sequence of Events and Systems Operation	15.6-3
	15.6.2.3	Barrier Performance	15.6-4
	15.6.2.4	Radiological Consequences	15.6-4
15.6.3	Steam Generator Tube Failure		15.6-5
15.6.4	Steam System Line Break Outside Containment		15.6-5
	15.6.4.1	Identification of Causes and Frequency Classification	15.6-5
	15.6.4.2	Sequence of Events and Systems Operation	15.6-6
	15.6.4.3	Core and System Performance	15.6-6
	15.6.4.4	Barrier Performance	15.6-9
	15.6.4.5	Radiological Consequences	15.6-10
15.6.5	Loss-of-Coolant Accidents Resulting from Piping Breaks Inside Containment		15.6-12
	15.6.5.1	Identification of Causes and Frequency Classification	15.6-12
	15.6.5.2	Sequence of Events and Systems Operation	15.6-12
	15.6.5.3	Core and System Performance	15.6-13
	15.6.5.4	Barrier Performance	15.6-14
	15.6.5.5	Radiological Consequences	15.6-17
15.6.6	References		15.6-25
15.7	RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT		15.7-1
15.7.1	Postulated Liquid Releases Due to Liquid Tank Failures		15.7-1
15.7.2	Design Basis Fuel Handling Accidents Inside Containment and Spent Fuel Storage Buildings		15.7-1
	15.7.2.1	Identification	15.7-1
	15.7.2.2	Designed Safeguards	15.7-2
	15.7.2.3	Procedural Safeguards	15.7-2
	15.7.2.4	Accident Analysis	15.7-2
	15.7.2.5	Radiological Effects	15.7-6

			Page
15.7.3	Spent Fuel Cask Drop Accidents		15.7-8a
15.7.4	Radioactive Gas Waste System Leak or Failure		15.7-12
15.7.5	References		15.7-13
15.8	ANTICIPATED TRANSIENTS WITHOUT SCRAM		15.8-1
15.8.1	Closure of Main Steam Line Isolation Valves		15.8-1b
	15.8.1.1	Identification of Causes	15.8-1b
	15.8.1.2	Sequence of Events and Systems Operations	15.8-2
	15.8.1.3	Core and System Performance	15.8-2
	15.8.1.4	Barrier Response	15.8-4
	15.8.1.5	Radiological Consequences	15.8-4
15.8.2	Loss of Normal AC Power		15.8-4
	15.8.2.1	Identification of Causes	15.8-4
	15.8.2.2	Sequence of Events and Systems Operations	15.8-5
	15.8.2.3	Core and System Performance	15.8-5
	15.8.2.4	Barrier Performance	15.8-5
	15.8.2.5	Radiological Consequences	15.8-6
15.8.3	Loss of Normal Feedwater Flow		15.8-6
	15.8.3.1	Identification of Causes	15.8-6
	15.8.3.2	Sequence of Events and Systems Operations	15.8-6
	15.8.3.3	Core and System Performance	15.8-6
	15.8.3.4	Barrier Performance	15.8-7
	15.8.3.5	Radiological Consequences	15.8-7
15.8.4	Turbine-Generator Trip/Load Rejection		15.8-7
	15.8.4.1	Identification of Causes	15.8-7
	15.8.4.2	Sequence of Events and Systems Operations	15.8-7
	15.8.4.3	Core and System Performance	15.8-8
	15.8.4.4	Barrier Performance	15.8-8
	15.8.4.5	Radiological Consequences	15.8-8
15.8.5	Loss of Condenser Vacuum		15.8-8
	15.8.5.1	Identification of Causes	15.8-9
	15.8.5.2	Sequence of Events and Systems Operations	15.8-9
	15.8.5.3	Core and System Performance	15.8-9
	15.8.5.4	Barrier Performance	15.8-9
	15.8.5.5	Radiological Consequences	15.8-10
15.8.6	Pressure Regulator Failure – Open to Maximum Demand		15.8-10
	15.8.6.1	Identification of Causes	15.8-10
	15.8.6.2	Sequence of Events and Systems Operations	15.8-10
	15.8.6.3	Core and System Performance	15.8-10
	15.8.6.4	Barrier Response	15.8-10
	15.8.6.5	Radiological Consequences	15.8.10
15.8.7	References		15.8-11

15.0 ACCIDENT AND TRANSIENT ANALYSES
LIST OF TABLES

Table	
15.0-1	Summary of Maximum Offsite Doses From Postulated Accidents
15.4-1	Summary of Doses Calculated at LPZ, Control Room, and EAB for TID and Siemens Source Terms
15.4-1a	Control Rod Drop Accident EAB, LPZ, and Control Room Doses following EPU
15.4-1b	Control Rod Drop Accident EAB, LPZ, and Control Room Doses (Alternative Source Term)
15.4-2	Deleted
15.6-1	Effect of Main Steam Line Isolation Valve Closure Time
15.6-2	Radiological Effects of the Steam Line Break Accident (Initial Core Analysis)
15.6-2a	Radiological Effects of the Steam Line Break Accident (Alternative Source Term)
15.6-3	Loss-of-Coolant Accident Primary Containment Airborne Fission Product Inventory (Initial Core)
15.6-4	Loss-of-Coolant Accident Reactor Building Airborne Fission Product Inventory (Initial Core)
15.6-5	Loss-of-Coolant Accident Fission Product Release Rate From Chimney (Initial Core)
15.6-5a	Isotopic Core Inventory (Alternative Source Term)
15.6-5b	Reactor Building Isotopic Core Inventory (Alternative Source Term)
15.6-6	Offsite Radiological Effects of the Loss-of-Coolant Accident (Initial Core)
15.6-7	Loss-of-Coolant Accident Input Parameters (Alternative Source Term)
15.6-8	Loss-of-Coolant Accident Control Room Radiological Effects (rem)
15.6-8a	Loss-of-Coolant Accident EAB, LPZ and Control Room Doses following EPU (Alternative Source Term)
15.6-9	Atmospheric Dispersion Factors for LOCA (Alternative Source Term)
15.7-1	Reactor Building Airborne Fission Product Inventory
15.7-2	Refueling Accident Fission Product Release Rate from Chimney
15.7-3	Radiological Effects of the Refueling Accident
15.7-4	Radiological Effects of a Refueling Accident in the Spent Fuel Pool or Containment
15.7-4a	Fuel Handling Accident in the Spent Fuel Pool or Containment EAB, LPZ, and Control Room Doses following EPU
15.7-4b	Fuel Handling Accident in the Spent Fuel Pool or in Containment EAB, LPZ, and Control Room Doses (Alternative Source Term)
15.7-5	Radiological Effects of an Off-gas System component Failure

TABLE OF CONTENTS (Continued)

15.0 ACCIDENT AND TRANSIENT ANALYSES
LIST OF FIGURES

Figure

15.1-1	Results of Analysis — Feedwater Control to 110%
15.1-2	Results of Analysis — Feedwater Control
15.1-3	Results of Analysis — Pressure Regulator to 115%
15.1-4	Results of Analysis — Pressure Regulator to 115%
15.2-1	(Initial Core) Results of Analysis — Generator Trip
15.2-2	(Initial Core) Results of Analysis — Generator Trip
15.2-3	(Initial Core) Results of Analysis — Turbine Trip (40% Bypass)
15.2-4	(Initial Core) Results of Analysis — Turbine Trip (40% Bypass)
15.2-5	(Initial Core) Results of Analysis — MSIV Isolation in 3 Seconds (30% Relief)
15.2-6	(Initial Core) Results of Analysis — MSIV Isolation in 3 Seconds (30% Relief)
15.2-7	(Initial Core) Results of Analysis — Loss of All Feedwater
15.2-8	(Initial Core) Results of Analysis — Loss of All Feedwater
15.3-1	(Initial Core) Trip of Two Recirculation Drive Motor Breakers — Transient Analysis
15.3-2	(Initial Core) Trip of Two Recirculation Drive Motor Breakers — Transient Analysis
15.3-3	(Initial Core) Trip of One Recirculation Drive Motor Breaker — Transient Analysis
15.3-4	(Initial Core) Trip of One Recirculation Drive Motor Breaker — Transient Analysis
15.3-5	(Initial Core) Speed Control Failure, Zero Demand — Transient Analysis
15.3-6	(Initial Core) Speed Control Failure, Zero Demand — Transient Analysis
15.3-7	(Initial Core) Seizure of One Recirculation Pump Shaft - Transient Analysis
15.3-8	(Initial Core) Seizure of One Recirculation Pump Shaft - Transient Analysis
15.4-1	(Initial Core) Cold Loop Startup - Transient Analysis
15.4-2	(Initial Core) Cold Loop Startup - Transient Analysis
15.4-3	(Initial Core) Speed Control Failure, Full Demand - Transient Analysis
15.4-4	(Initial Core) Speed Control Failure, Full Demand - Transient Analysis
15.5-1	Deleted
15.5-2	Deleted
15.6-1	Steam Line Break — Coolant Loss
15.6-2	Steam Line Break — Reactor Vessel Pressure and Mass Changes
15.6-3	Steam Line Break — Core Inlet Coolant Flow
15.6-4	(Initial Core) Steam Line Break Analysis, Core Axial Power Distribution
15.6-5	Steam Line Break — MCHFR Response
15.6-6	Minimum Critical Heat Flux Ratio and Minimum Critical Power Ratio vs. Time (Outside Steam Line Break) Typical BWR/4
15.6-7	Control Room Dose Analysis Model
15.8-1	ATWS — MSIV Closure Transient: Reactor Pressure Response
15.8-2	ATWS — MSIV Closure Transient: Neutron Flux Response
15.8-3	ATWS — MSIV Closure Transient: Reactor Level Response

15.0 ACCIDENT AND TRANSIENT ANALYSES

The evaluation of the safety of a nuclear power plant includes analyses of the response of the plant to postulated disturbances in process variables and to postulated malfunctions or failures of equipment. These safety analyses provide a significant contribution to the design and operation of components and systems from the standpoint of public health and safety. [15.0-1]

In previous chapters, the important structures, systems, and components have been discussed. Chapter 4 describes the reactor and its' analyzed operational conditions, including provisions for Maximum Extended Load Line Limit (MELLLA), Increased Core Flow (ICF), and Equipment Out-of-Service (EOOS). The following EOOS conditions are analyzed or evaluated for impact on thermal limits and the results are reported for the applicable transient analyses: feedwater temperature reduction, TBV out of service, one SRV out of service, single loop operation, TCV slow closure, PLU out of service, pressure regulator out of service, one TCV stuck closed, one TSV stuck closed, and one MSIV out of service (at 75% power). Additional details related to the combination of equipment OOS options with ARTS, MELLLA, and ICF operation can be found in cycle specific reload documentation and Reference 4. The thermal limits associated with implementation of each EOOS options are provided in the Core Operating Limits Report.

The EOOS conditions have been analyzed for the impact on fuel thermal limits and the design basis of the fuel (the transient analysis does not evaluate the effect of the EOOS condition on the design basis of the system). EOOS Options, as provided in the cycle and unit specific Core Operating Limits Report, are only implemented for temporary conditions where equipment is operated in a degraded condition. In these instances, the operability process for degraded / non-conforming conditions is followed, with the EOOS fuel thermal limits being compensating actions. The EOOS options do not support permanent plant modifications, procedure revisions, or other permanent changes to the facility.

In this chapter, the effects of anticipated process disturbances and postulated component failures are examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such failures and situations (or to identify the limitations of expected performance).

The situations analyzed include anticipated operational occurrences (e.g., a loss of electrical load), unexpected operational occurrences, and postulated accidents of low probability (e.g., the sudden loss of integrity of a major component). The analyses include an assessment of the consequences of an assumed fission product release that would result in potential hazards not exceeded by those from any accident considered credible.

15.0.1 Frequency Classification

The effects of various postulated anticipated operational occurrences (AOOs) and accident events are investigated for a variety of plant conditions. Some of the events have been categorized into three groups according to frequency of occurrence. The frequency classifications are as following. [15.0-2]

A. Incidents of moderate frequency - these are incidents that may occur with a frequency greater than once per 20 years for a particular plant. This event is referred to as an "anticipated (expected) operational occurrence."

B. Infrequent incidents - these are incidents that may occur during the life of the particular plant (spanning once in 20 years to once in 100 years). This event is referred to as an "abnormal (unexpected) operational occurrence."

C. Limiting faults - these are incidents that are not expected to occur but are postulated because their consequences may result in the release of significant amounts of radioactive material. This event is referred to as a "design basis (postulated) accident."

15.0.2 Transients and Accidents Analyzed

The core-wide anticipated operational occurrences (AOOs) were analyzed to support the extended power uprate (EPU) conditions (including the MELLLA domain) and the incorporation of the APRM rod block monitor technical specifications (ARTS) power and flow dependent limits improvement program. This included re-evaluating a broad set of the most limiting transient events at EPU conditions. The basis of the selection of the transient events for re-analysis is documented in Reference 1. The transient events which are re-analyzed with power uprate conditions from 2511 MWt to 2957 MWt core thermal power are documented in Reference 2.

The existing licensing bases were reviewed by Westinghouse to determine the potentially limiting analyses that must be done on a cycle-specific basis or on a one-time basis to support the introduction of SVEA-96 Optima2 fuel. The basis for the selection of the limiting events is discussed in the Westinghouse reload licensing methodology basis document for Quad Cities (Reference 5). A summary of the results of the events that are re-analyzed is documented in the cycle-specific reload licensing report. The parameters used for the transient analysis are documented in the OPL-W. For the reload specific AOO safety analyses performed by Westinghouse, a reactor scram occurs on other RPS trip signals prior to the low reactor vessel water level (L3) being reached.

15.0.2.1 Anticipated Operational Occurrences

Transients typically occur as a consequence of a single equipment failure or malfunction or single operator error. Such transients are evaluated in the sections listed below: [15.0-3]

	Analysis	Section
A.	Startup of idle recirculation loop at incorrect temperature (Cold recirculation loop)	15.4.4
B.	Single and multiple recirculation pump trips	15.3.1
C.	Inadvertent opening of a safety valve, relief valve or safety Relief valve	15.6.1
D.	Recirculation flow controller failure (malfunction) - Increasing flow	15.4.5
E.	Recirculation flow controller failure (malfunction), zero speed demand – decreasing flow	15.3.2
F.	Inadvertent actuation of high pressure coolant injection (HPCI) during power operation	15.5.1
G.	Inadvertent closure of main steam line isolation valves (MSIVs)	15.2.4
H.	Loss of normal feedwater flow (Feedwater controller Malfunction) – zero flow	15.2.7
I.	Increase in feedwater flow (Feedwater controller malfunction) - max. flow	15.1.2
J.	Turbine trip with failure of bypass system	5.2.2.2.2, 15.2.3.1
K.	Turbine trip with partial bypass – max. power	15.2.3.2
L.	Turbine pressure regulator malfunction:	
	1. Increase in steam flow	15.1.3
	2. Decrease in steam flow	15.2.1
M.	Loss of main condenser vacuum	15.2.5
N.	Load rejection with bypass (Loss of electrical load)	15.2.2.2
O.	Loss of auxiliary power	8.3.1
P.	Instrument air failure	9.3.1.1
Q.	Failure of one diesel generator to start	8.3.1.6.4
R.	Power bus voltage	8.3.1

S.	Load rejection without bypass	15.2.2.1
T.	Decrease in feedwater temperature (loss of feedwater heating)	15.1.1
U.	Rod withdrawal error	15.4.2
V.	Thermal Hydraulic Instability	15.4.11

Some of the transients listed above are evaluated on a cycle specific basis. The nominal reactor operating pressure is approximately 1005 psig. Transient analyses typically use the nominal reactor operating pressure as an input to the analyses. Small deviations (5 to 10 psi) from the nominal pressure are not expected to change most of the transient analyses results. However, sensitivity studies for fast pressurization events (main turbine generator load rejection without bypass, turbine trip without bypass, and feedwater controller failure) indicate that the delta-CPR may increase for lower initial pressures. Therefore, the fast pressurization events have considered a bounding initial pressure based on a typical operating range to assure a conservative delta-CPR and operating limit. [15.0-4]

15.0.2.2 Design Basis Accidents

In order to evaluate the ability of the plant safety features to protect the public, a number of accidents are analyzed herein. These accidents are of very low probability; however they are considered in order to include the far end of the operating spectrum of challenges to the safeguards and the containment system. The accidents in this chapter are discussed in the following sections: [15.0-5]

	Analysis	Section
A.	Control rod drop	15.4.10
B.	Loss of coolant	15.6.2, 15.6.5
C.	Main steam line break	15.6.4
D.	One recirculation pump shaft seizure	15.3.3
E.	Fuel handling accidents	15.7.2
F.	Mislocated fuel assembly	15.4.7
G.	Misoriented fuel assembly	15.4.8
H.	Spent fuel cask drop	15.7.3

The analyses of design basis accidents provide expected maximum concentrations and discharge rates of radioactive effluents, and calculated offsite doses for certain postulated events. Table 15.0-1 tabulates this information. Further additional information is provided in the individual sections listed above relative to current analysis methods and results.

15.0.2.3 Radiological Assessments of Design Basis Accidents [15.0-6]

A chronology of different radiological assessments is given in UFSAR Section 15.6.5.5 for the loss-of-coolant accident.

The previous UFSAR licensing basis prior to extended power uprate utilized the TID-14844 methodology, which establishes the source term based on rated core thermal power. The power level used in the radiological assessment of design basis accidents is at 102% of the extended power uprate; i.e., 3016 MWt. Radiological doses following the power uprate were developed by applying scaling factors to the previous doses. These scaling factors accounted for higher fuel burnup levels and updated fission product inventories using the industry-accepted ORIGEN2 code, as discussed in Reference 2. Resultant impacts are discussed under the relevant sections.

In Reference 3, the NRC approved the use of Alternative Source Term (AST) for the evaluation of the onsite and offsite dose consequences for the following Design Basis Accidents: Loss of Coolant Accident (LOCA), Control Rod Drop Accident (CRDA), Fuel Handling Accident (FHA), and Main Steam Line Break (MSLB). The power level used in the radiological assessment of design basis accidents under AST is 102% of the extended power uprate thermal power limit (i.e., 2957MWt x 1.02 = 3016 MWt).

The design basis accidents assessed in the UFSAR which have a radiological release that is proportional to the core radionuclide inventory are the following:

A. Control Rod Drop (Section 15.4.10)

B. Loss-of-Coolant Accidents Resulting from Piping Breaks Inside Containment (Section 15.6.5)

C. Design Basis Fuel Handling Accidents Inside Containment and Spent Fuel Storage Buildings (Section 15.7.2)

The existing radiological assessments of the above design basis accidents as described in the referenced UFSAR sections utilize the core radionuclide inventory for GE fuel. Westinghouse SVEA-96 Optima2 fuel has different core radionuclide inventories than GE fuel, with some isotopes released in either higher or lower quantities when evaluated on an equivalent basis. The results of the SVEA-96 Optima2 fuel evaluation for these accidents have been added to the referenced UFSAR Sections.

The design basis accidents assessed in the UFSAR which do not have a radiological release that is proportional to the core radionuclide inventory are the following:

A. Main Steam Line Break (Section 15.6.4)

B. Instrument Line Break (Section 15.6.2)

C. Loss of Feedwater Flow (Section 15.2.7)

The specific activity of the primary coolant is limited by Technical Specification. In addition, there is no core uncovery and no perforations of the fuel during a main steam line break, instrument line break, or loss of feedwater flow. Therefore, since only the coolant activity is released, the radiological dose calculations are independent of fuel type or design.

15.0.2.4 References

1. "Licensing Topical Report Generic Guidelines for General Electric BWR Extended Power Uprate," NEDC-32424P-A, Appendix E, February 1999.

2. "Safety Analysis Report for Quad Cities 1 & 2 Extended Power Uprate," NEDC-32961P, Revision 2, August 2001.

3. Letter from M. Banerjee (U. S. NRC) to C. Crane (Exelon Corporation), Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2 - Issuance of Amendments Re: Adoption of Alternative Source Term Methodology," dated September 11, 2006 [SER correction letter: D. Collins (U. S. NRC) to C. Crane (Exelon Corporation), "Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2 – Correction of Safety Evaluation for Amendment Dated September 11, 2006," dated September 28, 2006].

4. "Dresden 2 and 3, Quad Cities 1 and 2 Equipment Out-of-Service and Legacy Fuel Transient Analysis," General Electric Company, GE-NE-J11-03912-00-01-R3, September 2005.

5. "Westinghouse BWR Reload Licensing Methodology Basis for Exelon Generation Company Quad Cities Nuclear Power Station Units 1 and 2," WCAP-16334-P, June 2005.

Table 15.0-1

SUMMARY OF MAXIMUM OFFSITE DOSES FROM POSTULATED ACCIDENTS
(Original analysis, retained for historical purpose)

Accident		Maximum Total Offsite Exposure - Rads	
		Whole Body	Thyroid
Rod drop	6.2×10^{4} curies noble gases 1.8 curies halogens released to condenser	1.2×10^{-2}	1.2×10^{-3}
Fuel loading	5.7×10^{3} curies noble gases 3.5×10^{3} curies halogens released to reactor water	5.9×10^{-3}	4.1×10^{-3}
Steamline rupture	5.4 curies noble gases 116 curies (principally) halogens released to reactor water	4.1×10^{-3}	5.2×10^{-1}
Loss-of-coolant	5.2×10^{5} curies noble gases 2.7×10^{4} curies halogens airborne in primary containment at 30 minutes	5.3×10^{-4}	1.3×10^{-4}

15.1 INCREASE IN HEAT REMOVAL BY THE REACTOR COOLANT SYSTEM

Events described in this section that result in decreased feedwater temperature may also result in a core thermal hydraulic instability transient. Refer to Section 15.4.11 for an overview of this event.

This section covers transients which involve an unplanned increase in heat removal from the reactor due to conditions or events in the reactor coolant system that are expected to occur with moderate frequency. Excessive heat removal, i.e., heat removal at a rate in excess of the heat generation rate in the core, causes a decrease in moderator temperature which increases core reactivity and can lead to an increase in power level and a decrease in shutdown margin. The power level increase, if sufficient, would be terminated by a reactor scram. Any unplanned power level increase, however, has the potential to cause fuel damage or excessive reactor coolant system pressure, and warrants analysis if expected with moderate frequency. [15.1-1]

The following design basis transients are covered in this section:

A. Feedwater system malfunctions that result in a decrease in final feedwater temperature;

B. Feedwater system malfunctions that result in an increase in feedwater flow; and

C. Steam pressure regulator malfunctions that result in increased steam flow.

These events, including the associated assumptions and conclusions, continue to be part of the plant's licensing basis. The conclusions of these analyses are still valid; however, specific details contained in the descriptions and associated figures should be used only to understand the analysis and its conclusions. Refer to the cycle reload licensing documents for cycle specific analyses performed.

For plant operation under extended power uprate (EPU) conditions, the limiting events (in terms of minimum critical power ratio (MCPR)) for an increase in heat removal by the reactor coolant system were found to be the loss of feedwater heating (LFWH) and feedwater controller failure (FWCF). The inadvertent HPCI event would also decrease the core coolant temperature similar to LFWH. Refer to Section 15.5 for discussion on analysis of the inadvertent HPCI event.

15.1.1 Decrease in Feedwater Temperature

Refer to the cycle reload licensing documents for the cycle-specific analysis performed.

Decrease in feedwater temperature due to loss of feedwater heating would result in core power increase due to the increase in core inlet subcooling and the reactivity effects of the corresponding increase in moderator density. [15.1-2]

15.1.1.1 Identification of Causes and Frequency Classification

Feedwater heating can be lost in at least two ways:

1. Steam extraction line to heater is closed, or

2. Feedwater is bypassed around heater.

The first case would produce a gradual cooling of the feedwater. In the second case the feedwater would bypass the heater and the reduction of heating would occur during the stroke time of the bypass valve. In either case the reactor vessel would receive feedwater that is cooler than normal. The maximum number of feedwater heaters which can be tripped or bypassed by a single event represents the most severe transient for analysis considerations. For Quad Cities, this would result in a final feedwater temperature decrease of 145°F. An instantaneous loss of feedwater heating capability would cause an increase in core inlet subcooling. This would increase core power due to the negative moderator temperature and moderator void reactivity coefficients. In automatic recirculation flow control mode some compensation of core power would be realized by modulation of core flow.

This incident is analyzed as having moderate frequency. [15.1-3]

15.1.1.2 Sequence of Events and System Operation

The following plant operating conditions and assumptions form the principal basis for which reactor behavior is analyzed during the loss of feedwater heating transient: [15.1-4]

A. The plant is operating at full power; and

B. The plant is operating in the manual recirculation flow control mode. (Automatic Flow Control is no longer used.)

For this event power would increase at a very moderate rate, and the operator would be expected to insert control rods as necessary to stay within the analyzed power-to-flow region. If this were not done the core power could exceed the scram setpoint and a scram would then occur.

15.1.1.3 Barrier Performance

The fuel-specific minimum critical power ratio (MCPR) limiting condition for operation (LCO) is determined for each reload core based on bounding events for the cycle. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit. Refer to the cycle-specific documentation or Core Operating Limits Report (COLR) for detailed results of the current cycle transient analyses.

15.1.1.4 Radiological Consequences

The fuel cladding integrity safety limit would not be violated; therefore, a radiological consequence analysis has not been performed.

15.1.2 Increase in Feedwater Flow

Refer to the cycle reload licensing documents for cycle-specific analyses performed.

15.1.2.1 Identification of Causes and Frequency Classification

This event is postulated on the basis of a single failure of a control device, specifically one which can directly cause an increase in coolant inventory by increasing feedwater flow. The most severe applicable event is a feedwater controller failure during maximum flow demand. The feedwater controller is forced to its upper limit at the beginning of the event. [15.1-5]

This is considered an incident of moderate frequency. [15.1-6]

15.1.2.2 Sequence of Events and System Operation

The operating conditions and assumptions considered in this analysis are as follows: [15.1-7]

A. Feedwater controller fails during maximum flow demand;

B. Maximum feedwater pump runout occurs; and

C. The reactor is operating in the manual recirculation flow control mode, which provides for the most severe transient.

A feedwater controller failure under these circumstances would produce the following sequence of events:

A. The reactor vessel receives an excess of feedwater flow;

B. This excess flow results in an increase in core subcooling, which results in a rise in core power, and an increase in reactor vessel water level; and

C. The rise in the reactor vessel water level eventually leads to a high water level turbine trip and a feedwater pump trip, and results in a reactor scram.

Under most conditions, no operator action would be required. The reactor would scram following the turbine trip on high water level and end the transient. The operator would verify that a feedwater pump trip had occurred to terminate the initiating condition. The analysis was initiated from a typical low power condition with reactor power and flow at various points along the APRM rod block and minimum pump speed lines, including ICF to 108% of rated flow. In the analysis of this event, operation in the manual recirculation flow control is considered. (Automatic recirculation flow control is no longer used.)

15.1.2.3 Core and System Performance

An excess feedwater flow transient due to a maximum feedwater demand by the feedwater controller was evaluated for the initial core and is shown in Figures 15.1-1 and 15.1-2. The low initial power level resulted in a more severe steam/feed flow mismatch and reactor water level transient.
[15.1-8]

The increase in feedwater flow due to a failure of the feedwater control system to maximum demand results in an increase in the water level and a decrease in the coolant temperature at the core inlet. The increase in core inlet subcooling causes an increase in core power. As the feedwater flow continues at the maximum demand, the water level will continue to rise and eventually will reach the high water level trip set point. The high water level trip causes the turbine stop valves to close to prevent damage to the turbine from excessive liquid inventory in the steam line. The turbine stop valve closure creates a compression wave that travels to the core causing a void collapse and subsequent rapid power excursion. The closure of the turbine stop valves initiates a reactor scram. As vessel pressure increases, the turbine bypass valves and relief valves provide pressure relief. The core power excursion is mitigated in part by the pressure relief, but the primary mechanisms for termination of the event are reactor scram and revoiding of the core. At power levels below 38.5% of rated, a reactor scram on turbine stop valve closure is bypassed. However, the reactor scram on high neutron flux and high pressure are still available. This transient is analyzed at numerous statepoints on a cycle-specific basis.

The FWCF event is also analyzed assuming turbine bypass valves are not available to provide a basis for this possible mode of operation. The thermal limits and limitations associated with implementation of this EOOS option are provided in the Core Operating Limits Report.

15.1.2.4 Barrier Performance

The fuel-specific minimum critical power ratio (MCPR) limiting condition for operation (LCO) is determined for each reload core based on bounding events for the cycle. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit. Cycle-specific results are described in the reload licensing documents or the Core Operating Limits Report.

15.1.2.5 Radiological Consequences

The fuel cladding integrity safety limit would not be violated; therefore a radiological consequence analysis has not been performed.

15.1.3 Increase in Steam Flow

See the introduction to Section 15.1 for information regarding use of details from this analysis description which may not be applicable to the current fuel cycle.

15.1.3.1 Identification of Causes

This event is postulated on the failure of the turbine pressure regulator in the valve-open direction. The maximum control-plus-bypass valve demand is limited by the control system to the EHC control system maximum combined flow limit (MCFL) setpoint (Section 7.7.4.2). [15.1-9]

15.1.3.2 Core and System Performance

Figures 15.1-3 and 15.1-4 show the results of an analysis of this malfunction at 2511 MWt. Vessel and steam line pressures drop 100 psi in the first 10 seconds. Core flux is decreased significantly as the pressure drop increases the moderator void fraction. When steam line pressure decreases, closure of the main steam isolation valves is initiated by a Group I isolation from steam line low pressure. A scram occurs when the isolation valves have reached 10% closed (analytical limit). The depressurization is stopped by the isolation and the reactor is shut down with pressure rising slowly. Pressure rise is limited by operation of the relief valves and the reactor core isolation cooling (RCIC) system, which is discussed in more detail in Section 5.4.

The sequence of events above continues to apply for operation under EPU conditions.

15.1.3.3 Barrier Performance

This transient is not analyzed for reload cores since the fuel-specific MCPR LCO is determined for each reload core based on bounding events for the cycle. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit.

15.1.3.4 Radiological Consequences

The fuel cladding integrity safety limit would not be violated; therefore a radiological consequence analysis has not been performed.

15.2 DECREASE IN HEAT REMOVAL BY THE REACTOR COOLANT SYSTEM

Some events described in this section have not been reanalyzed for the current fuel cycle, because these events continue to be bounded by other events which are analyzed for the current fuel cycle. These events, including the associated assumptions and conclusions, continue to be part of the plant's licensing basis. The conclusions of these analyses are still valid; however, specific details contained in the descriptions and associated figures should be used only to understand the analysis and its conclusions. These specific details should not be used as sources of current fuel cycle design information. Refer to the cycle-specific reload licensing documentation or the COLR, for detailed results of current cycle transient analyses.

For operation at EPU conditions, the events resulting in a decrease in heat removal by the reactor coolant system were analyzed. The events in this category are primarily represented in the EPU analysis guidelines by the turbine trip and load reject transient events with the assumed failure of the turbine steam bypass function. The feedwater controller failure (maximum demand) event also includes some aspects of this area, since it involves a turbine trip (from high water level). Other pressurization events analyzed include the MSIV closure with direct scram, load rejection with bypass, and a single MSIV closure. The loss of condenser vacuum is another type of turbine trip with bypass and is bound by events without bypass operation. The loss of offsite AC power and loss of normal feedwater are similar events. These events result in initial power decreases and are not limiting with respect to thermal limits.

15.2.1 Steam Pressure Regulator Malfunction

15.2.1.1 Identification of Causes

For this event, the turbine pressure regulator is assumed to fail low (i.e., zero output). [15.2-1]

15.2.1.2 Sequence of Events and System Operation

If one of the three processors in the pressure controller failed low, the pressure controller would maintain control of the turbine valves with no change in pressure. If either one or two of the three pressure transmitters providing input to the pressure controller failed low, the turbine control valves would adjust to the pressure sensed by the functioning transmitter and a small change in pressure could occur.

If a second processor or the third transmitter failed low, the turbine control valves will close resulting in an increase in reactor pressure.

15.2.1.3 Core and System Performance

If one of the three pressure transmitters and two of the three processors in the pressure controller remained functional, the transient would be similar to a pressure setpoint increase as shown in Section 4.3.2.3.4.4.

If a second processor or the third transmitter failed low, the turbine control valves will close resulting in an increase in reactor pressure leading to a reactor scram on high flux.

15.2.1.4 Barrier Performance

This transient is not analyzed for reload cores since the fuel-specific minimum critical power ratio (MCPR) limiting conditions for operation (LCO) is determined for each reload core based on other events which bound this event. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit. [15.2-2]

The failure of the pressure controller is bounded by the analysis performed for the turbine control valve slow closure in the equipment out of service report (Reference 2) or the Westinghouse Reload Licensing Report.

15.2.1.5 Radiological Consequences

The fuel cladding integrity safety limit would not be violated; therefore, a radiological consequence analysis was not performed.

15.2.2 Load Rejection

A power/load imbalance system is provided which senses the generator load and makes a comparison to the thermal power (as indicated by intermediate steam pressure). When a mismatch in excess of 40% occurs the power/load imbalance relay will energize the fast acting solenoid valves on the turbine control valves. This results in a turbine control valve fast closure and a subsequent reactor scram. [15.2-3]

15.2.2.1 Load Rejection (Generator Trip) Without Bypass (LRNB) (Evaluated Each Cycle in the Reload Licensing Documents)

15.2.2.1.1 Identification of Causes and Frequency Classification

The following plant operating conditions and assumptions form the principal bases for which the reactor transient is analyzed during a load rejection.

A. The reactor and turbine generator are initially operating at full power when the load rejection occurs.

B. All of the plant control systems continue normal operation.

C. Auxiliary power is continuously supplied at rated frequency.

D. The reactor is operating in the manual flow control mode when load rejection occurs. (Automatic Flow Control is no longer used.)

E. The turbine bypass valve system is failed in the closed position.

The LRNB transient is classified as a moderate frequency event. MCPR limits are defined such that the MCPR Fuel Cladding Integrity Safety Limit is not violated during the occurrence of this transient. [15.2-4]

This transient is a potentially limiting event requiring analysis on a cycle-specific basis to verify or establish operating limits. The results are contained in the cycle-specific reload licensing documents or the COLR.

The LRNB event is also analyzed or evaluated for three EOOS options: the power/load unbalance system out of service option, TCV slow closure option, and pressure regulator out of service option. The thermal limits associated with implementation of each EOOS option are provided in the Core Operating Limits Report.

15.2.2.1.2 Sequence of Events and System Operation

Complete loss of the generator load produces the following sequence of events: [15.2-5]

A. The power/load imbalance actuation steps the load reference signal to zero and closes the turbine control valves at the earliest possible time. The turbine accelerates at a maximum rate until the valves start to close. The turbine control valves will close at a rate of 0.150 seconds for the full valve stroke.

B. Reactor scram is initiated upon sensing control valve fast closure.

C. If the pressure rises to the pressure relief setpoint, some or all of the relief valves open, discharging steam to the suppression pool.

D. If the pressure rises to >1250 psig, the trip of recirculation pump drive motors occurs. For assessing consequences of this event on thermal margin, however, the recirculation pump trip is conservatively assumed not to occur. [15.2-6]

GE has identified in a 10 CFR Part 21 letter that at lower reactor power levels (above Pbypass), the Power Load Unbalance (PLU) device may not actuate and the turbine control system will initiate turbine control valve closure at normal speed, which would not generate a direct scram (Reference 3). This would occur if the PLU is calibrated to actuate at power levels above Pbypass.

15.2.2.1.3 Core and System Performance

Fast closure of the turbine control valves would be initiated whenever electrical grid disturbances occur which result in significant loss of load on the generator. The turbine control valves are required to close as rapidly as possible to prevent overspeed of the turbine generator rotor. The closing would cause a sudden reduction of steam flow which results in a nuclear system pressure increase. The reactor would be scrammed by the fast closure of the turbine control valves. [15.2-7]

The reactor core isolation cooling (RCIC) and shutdown cooling mode of the RHR system would be initiated to handle long-term decay heat removal.

The MCPR would not exceed the MCPR fuel cladding integrity safety limit as determined by cycle-specific reload analysis. [15.2-8]

For the situation that turbine control valve fast closure does not occur above Pbypass but below the power/load unbalance setpoint, the analysis accounting for how the plant actually behaves has been performed for the applicable fuel types. This analysis credits the generator protection logic, which would initiate a turbine trip within 0.625 seconds of load rejection resulting in a turbine stop valve position scram. The analysis concludes that the equipment-in-service thermal limits from the Core Operating Limits Report (COLR) are bounding for this event (Reference 4). See the Westinghouse Reload Licensing Report for a similar conclusion based on the Westinghouse methodology that is described in Reference 5.

15.2.2.1.4 Barrier Performance

The fuel-specific MCPR LCO is determined for each reload core based on bounding events for the cycle. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit. Refer to the cycle-specific documentation or the Core Operating Limits Report (COLR) for detailed results of current cycle transient analyses.

15.2.2.1.5 Radiological Consequences

The fuel cladding integrity safety limit would not be violated; therefore, a radiological consequence analysis was not performed.

15.2.2.2 Load Rejection With Bypass

15.2.2.2.1 Identification of Cause and Frequency Classification

The cause and frequency classification of this transient are the same as that for load rejection without bypass discussed in Section 15.2.2.1.

15.2.2.2.2 Sequence of Events and System Operation

A loss of generator load causes the turbine generator to overspeed. The turbine speed and acceleration protection systems quickly close the turbine control valves to avoid excessive turbine overspeed. The control valves are fully closed in about 0.15 second following the initiation of the event. Above 41% load, a scram is initiated by sensing the turbine generator load imbalance and sending an electrical signal to the fast acting solenoid. This results in control valve closure and reactor scram. [15.2-9]

15.2.2.2.3 Core and System Performance

The transient response of the unit to a generator trip from 2511 MWt is shown for the initial core in Figures 15.2-1 and 15.2-2. The pressure rise causes core voids to collapse and neutron flux reaches approximately 145% before the scram terminates the transient. The increased core pressure and saturation temperature momentarily stores heat in the fuel causing the dip in average surface heat flux. The average heat flux never exceeds the initial value before it decays following the reactor scram. Coupled with the slight increase in core flow, this produces no decrease in MCPR.

The transient response is similar for operation under EPU conditions.

15.2.2.2.4 Barrier Performance

This transient is not analyzed for reload cores since the fuel-specific MCPR LCO is determined for each reload core based on other events, e.g., LRNB, which bound this event. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit. [15.2-10]

15.2.2.2.5 Radiological Consequences

The fuel cladding integrity safety limit would not be violated; therefore, a radiological consequence analysis was not performed.

15.2.3 Turbine Trip

The analysis of a turbine trip, coincident with failure of the turbine bypass system which was used to evaluate the required capacity of the relief valves and safety valves is discussed in Sections 5.2.2.2.2 and 5.2.2.2.3, respectively. The turbine trip analyses without bypass and with bypass, presented in Sections 15.2.3.1 and 15.2.3.2, respectively, assume a reactor scram due to turbine trip (stop valve closure). [15.2-11]

15.2.3.1 Turbine Trip Without Bypass (TTNB) (Evaluated Each Cycle in the Reload Licensing Documents)

15.2.3.1.1 Identification of Causes and Frequency Classification

A variety of turbine or nuclear system malfunctions will initiate a turbine trip (see Section 10.2.2). [15.2-12]

This event is classified as a moderate frequency event.

15.2.3.1.2 Sequence of Events and Systems Operations

The sequence of events for a turbine trip would be similar to those for a generator load rejection. Position switches at the stop valves would sense the valve closure and provide a reactor scram signal. If the pressure were to rise to the pressure relief setpoints the relief valves would open and discharge steam to the suppression pool.

15.2.3.1.3 Core and System Performance

The turbine stop valves would close as rapidly as possible. The closing would cause a sudden reduction of steam flow which results in a nuclear system pressure increase. The reactor would be scrammed by the closure of the turbine stop valves. [15.2-13]

The reactor core isolation cooling (RCIC) and shutdown cooling mode of the RHR system would be initiated to handle long-term decay heat removal.

The MCPR limit would not exceed the fuel cladding integrity safety limit as determined by cycle-specific reload analysis.

15.2.3.1.4 Barrier Performance

The fuel-specific MCPR LCO is determined for each reload core based on bounding events for the cycle. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit. Refer to the cycle-specific documentation or the Core Operating Limits Report (COLR) for detailed results of current cycle transient analyses. [15.2-14]

15.2.3.1.5 Radiological Consequences

The fuel cladding integrity safety limit would not be violated; therefore, a radiological consequence analysis was not performed.

15.2.3.2 Turbine Trip with Bypass

15.2.3.2.1 Identification of Causes and Frequency Classification

A turbine stop valve closure can be initiated by a variety of turbine or reactor system malfunctions (see section 10.2). [15.2-15]

This event is classified as a moderate frequency event. [15.2-16]

15.2.3.2.2 Sequence of Events and System Operation

The sudden closure of the stop valves would cause a rapid pressurization of the steam line and reactor vessel with resultant void collapse and power increase. The reactor would scram immediately from position switches mounted on the stop valves (turbine trip scram). Closure of the stop valves would also immediately initiate bypass valve opening. [15.2-17]

15.2.3.2.3 Core and System Performance

The resulting transient from 2511 MWt is shown in Figures 15.2-3 and 15.2-4 (based on initial reload core). The bypass valves would limit the peak pressure rise at the relief valves to 1105 psig, 10 psi below the lowest relief valve setpoint of 1115 psig (analytical limit). Vessel pressure would peak at 1106 psig. A neutron flux peak of about 245% occurs about 0.5 second after the trip.

Results for EPU conditions are similar to non-EPU and are bound by the Turbine Trip Without Bypass Event.

15.2.3.2.4 Barrier Performance

This transient is not analyzed for reload cores since the fuel-specific MCPR LCO is determined for each reload core based on other events, e.g., TTNB, which bound this event. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit. [15.2-18]

15.2.3.2.5 Radiological Consequences

The fuel cladding integrity safety limit would not be violated; therefore, a radiological consequence analysis was not performed.

15.2.4 Inadvertent Closure of Main Steam Isolation Valves

A full closure of all main steam isolation valves (MSIVs) without direct (from position switch) scram and with no credit taken for the relief valves is used to evaluate the required capacity of the main steam safety valves. This analysis is included in Section 5.2.2.2.3. [15.2-19]

A full closure of all MSIVs with direct scram and relief valve operation is described in this section.

15.2.4.1 Identification of Causes

The inadvertent closure of the MSIVs may be caused by operator error. [15.2-20]

15.2.4.2 Sequence of Events and System Operation

A MSIV closure can occur in 3 seconds. Reactor scram is initiated when the valves reach 10% closed.

15.2.4.3 Core and System Performance

The transient response to inadvertent closure of these valves from 2511 MWt is shown in Figures 15.2-5 and 15.2-6 (initial core). No safety problems would be encountered. No significant neutron flux or surface heat flux peaks would be encountered since the first 10% of valve stroke would not reduce valve flow area, and therefore MCPR would not go below the MCPR Safety Limit. The relief valves would open to remove excess stored heat. The peak pressure at the safety valves would reach only 1144 psig, well below the lowest safety valve setpoint of 1240 psig. The reactor core isolation cooling (RCIC) and shutdown cooling mode of the RHR system would be initiated to handle long-term decay heat removal. [15.2-21]

15.2.4.4 Barrier Performance

This transient is not analyzed for reload cores since the fuel-specific MCPR LCO is determined for each reload core based on other events which bound this event. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit. [15.2-22]

15.2.4.5 Radiological Consequences

The fuel cladding integrity safety limit would not be violated; therefore, a radiological consequence analysis was not performed.

15.2.5 Loss of Condenser Vacuum

15.2.5.1 Identification of Causes

The main condenser vacuum is assumed to be suddenly lost while the unit is operating at rated thermal power. [15.2-23]

15.2.5.2 Sequence of Events and System Operation

The following would occur due to the loss of condenser vacuum:

Alarm at	24 in.Hg vacuum
Scram at	20 in.Hg vacuum
Turbine stop valve closure at	20 in.Hg vacuum
Turbine bypass valve closure at	7 in.Hg vacuum

The worst case would occur if the loss of vacuum were instantaneous. In this event the transient would become identical to the turbine trip with bypass failure discussed in Section 5.2.2.2. The relief valves would open to prevent safety valve operation.

15.2.5.3 Core and System Performance

The majority of the stored heat would be removed by the relief valves and the RCIC system using the suppression pool as a heat sink (see Section 5.4). Slower losses of condenser vacuum would produce less severe transients because the scram would precede the stop valve closure and some bypass flow to the main condenser would remove stored heat.

15.2.5.4 Barrier Performance

This transient is not analyzed for reload cores since the fuel-specific MCPR LCO is determined for each reload core based on other events which bound this event. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit. The loss of condenser vacuum is bounded by Turbine Trip Without Bypass Event. [15.2-24]

15.2.5.5 Radiological Consequences

The fuel cladding integrity safety limit would not be violated; therefore, a radiological consequence analysis was not performed.

15.2.6 Loss of Offsite AC Power

The onsite power systems provide power to vital loads in the event of a loss of auxiliary power from offsite sources. The consequences of a loss of offsite ac power are addressed in Section 8.3. This event is bounded by the load reject no bypass or turbine trip no bypass events.

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Identification of Causes

A loss of feedwater transient response is assumed to occur due to a feedwater controller malfunction demanding closure of the feedwater control valves. [15.2-25]

15.2.7.2 Sequence of Events and System Operation

With an initial power level of rated power, feedwater control valves are assumed to close at their maximum rate. The unit response to simultaneous tripping of all feedwater pumps would be very similar to the transient analyzed. The reactor water level would decrease rapidly due to the mismatch between the steam flow out of the vessel and the shut-off feedwater flow. Low water level scram would occur after about 7.4 seconds.

The recirculation flow controller would reduce to minimum speed demand when the feedwater flow dropped below 2 x 10^6 lbs/hr. This interlock would protect the recirculation drive pumps from steady-state NPSH problems.

15.2.7.3 Core and System Performance

The transient response to this event is shown in Figures 15.2-7 and 15.2-8 (initial core at 2511 Mwt).

Based on the initial reload core, the decrease in moderator subcooling would slightly decrease the neutron flux until a scram occurred and completely shut down the reactor. Vessel steam flow would closely follow the decay of fuel surface heat flux. Analysis of the transient was discontinued at 16 seconds because the model was not programmed to handle the situation when core inlet subcooling becomes negative, i.e. saturation at core inlet. Subsequent events would be a complete recirculation pump drive motor trip and main steam isolation valve closure, both occurring when the water level drops to the low-low level setpoint. The time when this would occur, predicted from the established rate of level decrease, is about 33.5 seconds. Pressure would rise following the isolation, and eventually actuation of the RCIC system and the RHR system (shutdown cooling mode) would handle the long-term shutdown heat removal. This again would be less severe than the turbine or generator trips.

Water inventory loss from 16 seconds until 36.5 seconds, the time the isolation valves would be closed, was conservatively estimated to be less than 550 ft^3 of saturated water. (At 16 seconds, vessel steam flow would be 45% of rated. For extreme conservatism, this rate was considered to exist until 36.5 seconds.)

Accounting for the conservative inventory loss after 16 seconds and assuming the recirculation pumps would trip, an estimate of the final water level was made. In this analysis, all steam existing as carry-under and as voids in the core, upper plenum, standpipes, and separators at 16 seconds was allowed to condense. The volume of water delivered to the scram discharge volume was considered to be removed from the vessel. Even neglecting the inventory makeup from the RCIC system, the calculations showed that more than 5 feet of water would remain above the core.

The loss of normal feedwater flow is bounded by more limiting events which are performed on a cycle-specific basis.

15.2.7.4 Barrier Performance

No thermal limits would be violated since the transient would be less severe than the turbine or generator trips. The fuel-specific MCPR LCO is determined for each reload core based on bounding events for the cycle. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit. As noted above and in the introduction to Section 15.2, this event is not reanalyzed for reload cores because its' results are bounded by turbine or generator trips. This transient is not analyzed for reload cores since the fuel-specific MCPR LCO is determined for each reload core based on other events which bound this event. [15.2-26]

15.2.7.5 Radiological Consequences

The fuel cladding integrity safety limit would not be violated; therefore a radiological consequence analysis was not performed.

15.2.8 References

1. “Dresden and Quad Cities Pressure Regulator OOS Analysis,” FRL02EX-014, dated October 22, 2002.

2. “Dresden 2 and 3, Quad Cities 1 and 2 Equipment Out-of-Service and Legacy Fuel Transient Analysis,” General Electric Company, GE-NE-J11-03912-00-01-R3, September 2005.

3. SC04-15, “Turbine Control System Impact in Transient Analyses,” 10 CFR Part 21 Communication, October 31, 2004.

4. “Dresden Units 2 and 3 and Quad Cities Units 1 and 2 Offrated Analyses Below the PLU Power Level,” GE-NE-0000-0040-2860-R0, July 2005.

5. “Westinghouse BWR Reload Licensing Methodology Basis for Exelon Generation Company Quad Cities Nuclear Power Station Units 1 and 2,” WCAP-16334-P, June 2005.

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

Events described in this section that result in reduced core flow rates may also result in a core thermal hydraulic instability transient. Refer to Section 15.4.11 for an overview of this event.

The AOO events in this category are not limiting for any GE BWR. These events are not reevaluated for reloads and are not required to be because they are not limiting (Reference 5). The decrease in core flow causes a decrease in reactor power and thermal limits are not challenged. However, the SLO pump seizure accident was analyzed under Reference 6 for the introduction of GE14 and EPU (2957 MWt) conditions.

This section describes events which cause a decrease in reactor coolant system flow rates, except for anticipated transients without scram (ATWS). The ATWS mitigation features, which include a double recirculation pump trip, are discussed in Section 15.8. The recirculation flow control system is described in Section 7.7.3.1.

Because they continue to be bounded by other events which are analyzed for the current fuel cycle, some events described in this section have not been reanalyzed for the current fuel cycle. These events, including the associated assumptions and conclusions, continue to be part of the plant's licensing basis. The conclusions of these analyses are still valid; however, specific details contained in the descriptions and associated figures should be used only to understand the analysis and its conclusions. These specific details should not be used as sources of current fuel cycle design information. Refer to the cycle reload licensing documents for cycle-specific analyses performed.

15.3.1 Single and Multiple Recirculation Pump Trips

The transient responses of the plant to the trip of one and of both recirculation pumps due to trip of the adjustable speed drives (ASDs) while operating at full power have been analyzed. No reactor scram is assumed due to these transients. However, a simultaneous trip of both drives implies a loss of auxiliary power, which would subsequently result in reactor scram. [15.3-1]

Extensive tests and analyses were conducted during the original design of the reactor coolant system to evaluate the performance characteristics of the jet pumps and the recirculation system, particularly with respect to pump design requirements and the effect of the pumping system on hydraulic and nuclear stability. These analyses included the evaluations which are in Section 5.4.1.3 and also included the evaluation of recirculation pump malfunctions which are discussed in the following subsections. [15.3-2]

If one of the recirculation pumps were to fail, flow through half of the jet pumps would decrease and the jet pumps would cease to function. Recirculation flow and then core flow would decay to a value lower than rated. In this case, flow would reverse through the 10 idle jet pump diffusers and the other 10 jet pumps would continue to function. The core flow reduction would result in less core pressure drop and the active jet pump flow ratio would increase. The driving flow in the active loop would remain essentially constant since the loop hydraulic characteristics would not change. [15.3-3]

Calculations for typical BWRs (see APED-5460[1]) show that the 10 jet pumps would provide nearly 150% of their normally rated flow at the lower core pressure drop. Therefore, the total flow injected by the jet pump system would be 75% of rated. About 22% of rated flow would bypass the core through the idle diffusers, hence the core flow would be about 53% of rated. This lower than normal core flow rate would result in more core coolant void formation. Core power would drop to and stabilize at about 70% of rated. If both drive pumps were tripped, natural circulation would provide approximately 30% of rated core flow.

The one-pump trip transient would be less severe than the complete loss of pumping power. Even in the two-pump trip case the MCPR would be greater than the MCPR Safety Limit. A gradual power decrease would be the only result. Therefore, there are no safety implications for either a one-pump trip or two-pump trip, provided that either the reactor is not operated in the region of potential thermal-hydraulic instability or the OPRM system is fully functional, as discussed in Section 4.4.3.1.10. After a pump trip, power could only be raised by a flow increase or by rod motion. Protection against excessive power generation at any given flow is provided by the rod block interlocks of the APRM and RBM (see Chapter 7). Loss of a driving pump would not result in unstable operation of the remaining pumps.

Subsequent to the original design evaluations, a series of tests performed at the General Electric Company (GE) Moss Landing Test Facility verified previous performance predictions. Throughout these tests, basic performance data were collected under conditions duplicating, in all important respects, the temperatures, pressures, and flow rates to be encountered by the recirculation system. Concurrent with performing the tests at Moss Landing, the loop in which the tests were being performed was analytically modeled. Agreement between the analytical model results and the actual test results was quite good. Recirculation system analyses were obtained and are presented in Sections 15.4.5, 15.3.2, 15.4.4, and in the following subsections. [15.3-4]

For all reloads, single and multiple recirculation pump trips are bounded by more limiting events which are performed on a cycle-specific basis.

15.3.1.1 Trip of Both Drive Motors (Historical for initial core with M-G sets, including Figures)

The two-drive-motor trip transient analysis provides an evaluation of the thermal margins associated with a core flow rate decrease commensurate with the rotating inertia of the recirculation drive equipment. The decrease in flow would cause additional void formation in the core which would decrease reactor power. The time constant of the fuel would cause the surface heat flux decay to lag behind the flow decay. The mismatch between reactor thermal power and recirculation flow would bring about a decrease in the minimum critical power ratio (MCPR). [15.3-5]

During a flow coastdown from a two-drive-motor trip, the MCPR was determined to be greater than the MCPR Safety Limit. This shows that adequate inertia has been provided in the recirculation drive equipment. The unit's response to the two-drive-motor trip is shown for initial core on Figures 15.3-1 and 15.3-2.

This transient is not analyzed for reload cores since the fuel-specific MCPR LCO is determined for each reload core based on other events which bound this event. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit. [15.3-6]

15.3.1.2 Trip of One Drive Motor (Historical for initial core with M-G sets, including Figures)

The results of this transient would be less severe than the trip of both drive motors or the stall of one pump. Therefore, the thermal margins during this transient would be greater than either of those cases. For Quad Cities, flow would increase through the active loop jet pump diffusers, and would finally provide about 72% of the original total jet pump diffuser flow. This flow would be split in the lower plenum with about 60% going through the core and the remainder providing reverse flow through the jet pump diffusers of the tripped loop. A small amount of forward flow would still be induced in the tripped drive loop due to the static pressure difference between the downcomer and the jet pump throat. The unit's response to the one-drive-motor trip is shown for initial core on Figures 15.3-3 and 15.3-4. [15.3-7]

This transient is not analyzed for reload cores since the fuel-specific MCPR LCO is determined for each reload core based on other events which bound this event. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit. [15.3-8]

15.3.1.3 Trip of One Recirculation Pump Motor

The transient responses of the plant to the trip of one or both recirculation pumps while operating at full power have been addressed in the plant licensing basis. No reactor scram is assumed due to these transients. The transient analysis provides an evaluation of the thermal margins associated with a core flow rate decrease commensurate with the rotating inertia of the recirculation pump. The decrease in flow causes additional void formation in the core which decreases reactor power. The time constant of the fuel causes the surface heat flux decay to lag behind the flow decay. The mismatch between reactor thermal power and recirculation flow brings about a decrease in the minimum critical power ratio (MCPR). The one pump transient is less severe than the complete loss of pumping power. The thermal margins would be greater than the seizure or stall of one pump, which was the most limiting transient analyzed at the time of the original license. Subsequent analyses (NEDO-10958-A[4]) showed that the pump seizure event was no longer a limiting transient. [15.3-9]

This transient is not analyzed for reload cores since the fuel-specific MCPR LCO is determined for each reload core based on other events which bound this event. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit. [15.3-10]

15.3.1.4 Trip of Two Recirculation Pump Motors

The trip of two recirculation pump motors is discussed in Section 15.8.

15.3.2 Recirculation Flow Controller Malfunctions

The equipment associated with the variable speed recirculation pump motors is designed with the basic objective that any failure should maintain the operating pump speed. However, the potential for failure in either direction (zero speed demand or full speed demand) does exist. These failures have been analyzed and are discussed here and in Section 15.4.5, respectively. [15.3-11]

[Start of HISTORICAL INFORMATION for initial core with M-G sets, including Figures]

A failure in a M-G set speed controller could cause the scoop tube positioner for the fluid coupler to move at its maximum speed in the direction of decreasing pump speed and flow. In the transient, the failed speed controller would move its positioner to near zero coupling at a maximum rate of about 10% per second. The resulting transient would be similar to a one-pump trip.

The recirculation flow and thermal power decay would be less severe than would result from tripping one of the recirculation drive motors. Therefore, the MCPR during this transient would be higher than for a one-pump trip, that is, the MCPR would always be greater than the MCPR Safety Limit. The unit's response to this transient is shown for initial core on Figures 15.3-5 and 15.3-6. [15.3-12]

[End of HISTORICAL INFORMATION]

A failure in an ASD speed signal would cause the ASD output to either go to zero frequency, thus shutting off the motor voltage the same as any other trip, or would attempt to increase frequency until the programmed limit is reached. Additionally, if the programmed limit function failed and the motor continued to accelerate, the over frequency protective relays would immediately trip the ASD feed and shut down the motor. The failure results in an increase in pump speed, which has been analyzed at 2.5% rated rpm/sec for a maximum run-up rate at normal operation (2 pump operation) and 42% rated rpm/sec for a maximum run-up rate for single pump operation.

The recirculation flow and thermal power decay resulting from the output going to zero frequency would be the same as analyzed in Section 15.3.1 for a trip of the recirculation pump ASD.

This transient is not analyzed for reload cores since the fuel-specific MCPR LCO is determined for each reload core based on other events which bound this event. Flow dependent MCPR limits are established to support operation at off-rated core flow conditions. The limits are based on the CPR changes experienced by the fuel during slow flow excursions. The slow flow excursion event assumes a failure of the recirculation flow control system such that the core flow increases slowly to the maximum flow physically attainable by the equipment. An uncontrolled increase in flow creates the potential for a significant increase in core power. The primary function of the flow-dependent MCPR limit is to protect against cladding overheat during transients initiated from less than rated flow conditions. Refer to the cycle reload licensing documents for cycle-specific analyses performed. [15.3-13] [15.3-14]

15.3.3 Recirculation Pump Shaft Seizure

This accident is assumed to occur as a consequence of an unspecified instantaneous stoppage of one recirculation pump shaft while the reactor is operating at full power. [15.3-15]

The pump seizure event would be a very mild accident in relation to other accidents such as a loss-of coolant accident (LOCA). In both a LOCA and a pump seizure, the recirculation driving loop flow would be lost extremely rapidly. Differences between the consequences of a LOCA and a pump seizure include:

A. In the case of the seizure, stoppage of the pump would occur; while for the LOCA, the severance of the line would have a similar, but more rapid and severe, influence.

B. Following a pump seizure event, flow would continue, water level would be maintained, and the core would remain submerged, which would provide a continuous core cooling mechanism. However, for the LOCA, complete flow stoppage would occur and the water level would decrease due to loss of coolant, eventually uncovering the reactor core and subsequently overheating the fuel rod cladding.

C. For the pump seizure accident, reactor pressure does not significantly decrease, whereas complete depressurization occurs for the LOCA.

The increased temperature of the cladding and reduced reactor pressure for the LOCA would combine to yield a much more severe stress and potential for cladding perforation than for the pump seizure. Therefore, the potential effects of the hypothetical pump seizure accident are bounded by the effects of a LOCA and specific analyses of the pump seizure accident are not required.

15.3.4 Recirculation Pump Shaft Break

The recirculation pump flow reduction events which have been analyzed are the trip of one drive motor (Section 15.3.1.2), trip of both drive motors (Section 15.3.1.1), trip of one recirculation pump motor (Section 15.3.1.3) and, recirculation pump shaft seizure (Section 15.3.3). The recirculation pump shaft break event is a limiting fault. The consequences of this event are bounded by other flow reduction events (Reference 6).

15.3.5 Jet Pump Malfunction

The effects of a malfunction of a single jet pump have been analyzed. For the purpose of analysis, one of the jet pump nozzles is assumed to be plugged while the plant is operating at full power. Two effects would be observed: flow through the blocked jet pump would reverse, bypassing some core flow and, the remaining 19 jet pumps would operate at a slightly higher flow ratio due to the altered hydraulic characteristics. The net effect would be a reduction in core flow to approximately 98% of rated (power approximately 99% of rated). Since the reactor power decrease would be essentially analogous to a flow control decrease, the margin from thermal limits would not be reduced. The flow and power reduction would be sensed by the monitoring equipment. Loss of a jet pump would not result in unstable operation of the remaining pumps. [15.3-16]

This transient is not analyzed for reload cores since the fuel-specific MCPR LCO is determined for each reload core based on other events which bound this event. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit. [15.3-17]

15.3.6 Transients During Single Loop Operation

[Start of HISTORICAL INFORMATION]

In Section 3.1 of NEDO-24807[2], GE has summarized their review of abnormal operating transients during single loop operation (SLO). In response to NRC questions raised during their review of Cooper Station's request for SLO Technical Specifications, GE completed specific analyses of numerous plant transients initiated during SLO. The results demonstrate the applicability of the analyses for Quad Cities Units 1 and 2. [15.3-18]

The three most important aspects of plant transients and SLO are:

A. They are initiated from less than rated power due to the lower core flow.

B. The safety limit MCPR and the limiting condition for operation (LCO) MCPR; i.e., operating limit MCPR, are increased. During SLO, the uncertainties of some of the core parameters increase (e.g, core flow, TIP readings). Therefore, the MCPR safety limit is increased to ensure that 99.9% of the fuel rods do not experience boiling transition during an AOO. Further discussion is provided in Section 4.4.4.2.5.

C. The average power range monitor (APRM) scram, APRM rod block, and rod block monitor (RBM) flow-biased setpoints are adjusted to preserve the relationship normally existent between the setpoints and operating points in the power/flow map during two-loop operation.

The highest power attainable during SLO is less than rated two-loop thermal power because core flow is reduced and rod patterns will be changed. Reductions in reactor power cause plant transients to become less severe. Abnormal plant transients initiated from SLO are conservatively bounded by two-loop analyses, provided that the above mentioned adjustments are made to the safety limit MCPR and operating limit MCPR, APRM scram, APRM rod block, and RBM setpoints.

A recirculation pump seizure is identified by GESTAR[3] as a design basis event. The consequence of this accident during single loop versus two loop operation has been addressed by GE in GESTAR. This accident is defined as the instantaneous stoppage of one recirculation pump shaft while the reactor is operating at full power. GE has considered this accident to be mild by comparison to a LOCA, but analyzed the event for SLO for EPU conditions (2957 MWt) at Quad Cities. As a result of this analysis it is required that a SLO OLMCPR minimum be implemented based on the value of the minimum DLO OLMCPR. These values can be found in Reference 6.

These events are not analyzed for reload cores since the fuel-specific MCPR LCO is determined for each reload core based on other events which bound this event. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit. [15.3-19]

[End of HISTORICAL INFORMATION]

For Westinghouse methodology, it has been demonstrated that the decrease in MCPR for the transients initiated from SLO are conservatively bounded by the DLO analysis. The safety limit for SLO is adjusted to include the increase in some of the core parameters uncertainties as specified in the Technical Specifications. See Reference 7 for more details. In addition, the maximum average linear heat generation rate (MAPLHGR) is adjusted for SLO as determined from the LOCA analysis.

For reactor power operation with SLO, both the fuel cladding integrity safety limit MCPR and the operating limit MCPR are increased by an amount specified in the Technical Specifications and Core Operating Limits Report, respectively. This increase is to account for increased uncertainties in core flow and traversing incore probe (TIP) instrumentation readings for single loop operations.

15.3.7 References

1. "Design and Performance of GE BWR Jet Pumps," General Electric Company, September 1968, APED 5460.

2. "Dresden Nuclear Power Station Units 2 and 3 and Quad Cities Nuclear Power Station Units 1 and 2, Single-Loop Operation," General Electric Company, December 1980, NEDO-24807.

3. "General Electric Standard Application for Reactor Fuel," (GESTAR II), General Electric Company, June 2000, NEDE-24011-P-A-14.

4. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," General Electric Company, January 1977, NEDO 10958-A.

5. "Safety Analysis Report for Quad Cities 1 & 2 Extended Power Uprate," NEDC-32961P Revision 2, August 2001.

6. "Dresden and Quad Cities Extended Power Uprate, Task T0900: Transient Analysis," GE-NE-A22-00103-10-01 Revision 0, October 2000.

7. "Westinghouse BWR Reload Licensing Methodology Basis for Exelon Generation Company Quad Cities Nuclear Power Station Units 1 and 2," WCAP-16334-P, June 2005.

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Because they continue to be bounded by generic analyses or analyses for previous fuel cycles, some events described in this section have not been reanalyzed for the current fuel cycle. These events, including the associated assumptions and conclusions, continue to be part of the plant's licensing basis. The conclusions of these analyses are still valid; however, specific details contained in the descriptions and associated figures should be used only to understand the analysis and its conclusions. These specific details should not be used as sources of current fuel cycle design information.

The events in this category that were analyzed for EPU conditions under Reference 23 are the Rod Withdrawal Error (RWE), Mislocated Fuel Assembly and the Recirculation Loop Flow Controller Failure. These events are analyzed by Westinghouse according to the reload licensing methodology basis in Reference 33.

15.4.1 Control Rod Removal Error During Refueling

15.4.1.1 Identification of Causes and Frequency Classification

This event considers an arbitrary full withdrawal of the most reactive control rod during refueling. Such an event is categorized by the frequency of a limiting fault. The probability of the initiating causes alone is considered low enough to warrant its being categorized as an infrequent incident because there is no practical set of circumstances which can result in an inadvertent RWE while in the REFUEL mode. [15.4-1]

15.4.1.2 Sequence of Events and System Operation

The refueling interlocks prevent any condition that could lead to a control rod withdrawal error during refueling, thus an inadvertent criticality is precluded.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the core. This requirement is backed up by refueling interlocks on rod withdrawal and on movement of the refueling platform. When the mode switch is in the “REFUEL” position, the interlocks prevent the platform from being moved over the core if a control rod is withdrawn and fuel is on the hoist. Likewise, if the refueling platform is over the core and fuel is on the hoist, control rod motion is blocked by the interlocks.

When the mode switch is in the REFUEL position, only one control rod can be withdrawn. Selection of a second rod initiates a rod block and thereby prevents the withdrawal of more than one rod at a time. Because the core is designed to meet shutdown requirements with the highest worth rod withdrawn, the core remains subcritical even with one rod withdrawn.

In addition, the design of the control rod, incorporating the velocity limiter, does not physically permit the upward removal of the control rod without the simultaneous or prior removal of the four adjacent fuel bundles. This precludes any hazardous condition.

No operator actions are required to preclude this event because the plant design prevents its occurrence. Even if the operator somehow withdraws one rod, the electrical interlocks prevent withdrawal of the second rod.

15.4.1.3 Core and System Performance

Subsection 4.3.2 contains the shutdown margin analysis.

No mathematical models were involved in this event. The need for input parameters or initial conditions was not required as there are no results to report. Consideration of uncertainties is not appropriate.

The probability of inadvertent criticality during refueling is precluded, hence the core and system performances were not analyzed. However, it is well known that withdrawal of the highest worth control rod during refueling results in a positive reactivity insertion but not enough to cause criticality. This is verified experimentally by performing shutdown margin tests during the startup series.

15.4.1.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since there is not a postulated set of circumstances for which this event could occur.

5.4.1.5 Radiological Consequences

An evaluation of the radiological consequences was not made for this event since no radioactive material is released from the fuel.

15.4.2 Rod Withdrawal Error — At Power

See the introduction to Section 15.4 for information regarding use of details from this analysis description which may not be applicable to the current fuel cycle.

15.4.2.1 Identification of Causes and Frequency Classification

This transient evaluation assumes the reactor is operating at a power level above 75% of rated power at the time control rod withdrawal error occurs. The reactor operator would have followed procedures, and up to the poi nt of the withdrawal error, the reactor would be in a normal mode of operation (i.e., the control rod pattern, flow setpoints, etc., would all be within normal operating limits). For these conditions, it is assumed that the withdrawal error occurs with the maximum worth control rod. Therefore, the maximum positive reactivity insertion would occur. [15.4-1a]

While operating in the power range in a normal mode of operation, the reactor operator is assumed to make a procedural error and withdraws the maximum worth control rod to its rod block position. Due to the positive reactivity insertion, the core average power would increase. More importantly, the local power in the vicinity of the withdrawn control rod would increase and could potentially cause cladding damage due to overheating. This may be accompanied by boiling transition, which is the assumed transient failure threshold. See Westinghouse reload licensing methodology basis for details specific to this transient (Reference 33).

The rod withdrawal error is considered a moderate frequency event. The control rod withdrawal error transient is a potentially limiting event. This event is analyzed on a cycle-specific basis to verify or establish operating limits.

15.4.2.2 Sequence of Events and System Operation

The following list depicts the sequence of events for this transient.

A. Event begins, operator selects the maximum worth control rod, acknowledges any alarms and withdraws the rod at the maximum rod speed. 0 seconds.

B. Core average power and local power increase causing local power range monitor alarm. 5 seconds.

C. Event ends — rod block by rod block monitor (RBM). 30 seconds (cycle-specific analysis may not credit RBM, in which case the rod withdrawal error event terminates at the full out position).

The worst case situation is established for the most reactive reactor state and assumes that no xenon is present. This ensures that the maximum amount of excess reactivity which must be controlled with the movable control rods is present. During a normal startup, sufficient time would be available to achieve some xenon and samarium buildup, and after some short period of operation, samarium will always be present. This assumption makes it possible to obtain a worst case situation in which the maximum worth control rod is fully inserted, and the remaining control rod pattern is selected in such a way as to achieve design thermal limits in the fuel assemblies directly or diagonally adjacent to the inserted maximum worth control rod which is to be withdrawn. [15.4-2]

15.4.2.3 Core and System Performance

The cycle-specific analysis considers the continuous withdrawal of the maximum worth control rod at its maximum drive speed from the reactor, which is assumed to be operating at rated power with a control rod pattern which results in the core being placed on thermal design limits. A worst case condition is analyzed to ensure that the results obtained are conservative; this approach also serves to demonstrate the function of the RBM system. A description of how the RBM system setpoints are determined appears in Section 7.6.1.5.3.3.

For Westinghouse reload cores, the RWE event is analyzed for potentially limiting cycle reactivity conditions, including the most reactive statepoint during the cycle. All control rods with a potential for being limiting are evaluated. The analysis is performed to select control rod patterns that would approach the thermal limits (e.g., MCPR and LHGR limits) in the region of the core where the rod is being erroneously withdrawn. The control rod patterns analyzed need not be consistent with normal control rod patterns. As a result, the control rod patterns analyzed are highly unlikely to occur, are very conservative, and establish a limiting analysis.

Results for this worst case condition for each reload will be provided in the cycle-specific licensing documents.

The control rod withdrawal error transient is a potentially limiting event. This event is analyzed on a cycle-specific basis to verify or establish operating limits. Refer to the cycle-specific documentation or Core Operating Limits Report (COLR) for detailed results of current cycle transient analyses. The maximum control rod drive withdrawal speed is 5.14 inches/second when the Operating Limit MCPR established in the Core Operating Limits Report (COLR) is set greater than or equal to the value corresponding to a RWE – at Power analysis for an "unblocked" condition (Reference 24).

15.4.2.4 Barrier Performance

The fuel-specific minimum critical power ratio (MCPR) limiting condition for operation (LCO) is determined for each reload core based on bounding events for the cycle. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit. Refer to the cycle-specific documentation or Core Operating Limits Report (COLR) for detailed results of current cycle transient analyses.

15.4.2.5 Radiological Consequences

The fuel cladding integrity safety limit would not be violated; therefore a radiological consequence analysis has not been performed.

15.4.3 Control Rod Maloperation

This event has not been analyzed for the Quad Cities Station. This transient is not analyzed for reload cores since the fuel-specific MCPR LCO is determined for each reload core based on bounding events for the cycle. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit.

15.4.4 Startup of Idle Recirculating Loop at Incorrect Temperature

See the introduction to Section 15.4 for information regarding use of details from this analysis description which may not be applicable to the current fuel cycle.

This section contains historical information applicable to Quad Cities initial core analysis with M-G sets in place. For a transition core with SVEA-96 Optima2 fuel, Westinghouse has evaluated the idle loop startup with ASD implemented for which details are documented in Reference 49.

15.4.4.1 Identification of Causes

The plant is operating with one recirculation loop in service and the other loop out of service. The idle loop is inadvertently started without warming the loop water. [15.4-4]

15.4.4.2 Sequence of Events and System Operation (Historical for initial core with M-G sets, including Figures)

The initial conditions, representing the worst case for this incident (Quad Cities initial core), are: [15.4-5]

A. One drive loop is shutdown and filled with cold water (110°F).

B. The active recirculation pump is operating at a speed producing about 61% of rated drive flow and 87% of normal rated diffuser flow in the 10 active jet pumps.

C. The core is receiving 40% of its normal flow, while the remainder of the flow is reversed up the 10 inactive jet pumps.

D. Reactor power is 60% of 2511 MWt, a rather severe condition for startup of the dead loop.

E. The drive pump suction valve is open but the equalizer line and pump discharge valves are closed.

F. The fluid coupler scoop tube in the idle recirculation loop is at a position approximately 50% (normal) generator speed demand.

The startup transient sequence for an M-G set driven pump is:

A. The drive motor breaker is closed at t = 0.

B. The drive motor reaches near synchronous speed quickly, while the generator approaches full speed in about 5 seconds.

C. At 5 seconds, the generator field breaker is automatically closed, loading the generator and starting the pump. Pump speed is shown in Figure 15.4-2. The coupler demand is automatically programmed back to minimum speed. [15.4-6]

D. The pump discharge valve is manually opened as soon as its interlock clears when the drive motor breaker is closed. (Normal procedure would delay valve opening to separate the two portions of the flow transient and make sure the drive loop contains properly mixed vessel-temperature water.) A nonlinear 30-second valve opening characteristic was used. [15.4-7]

15.4.4.3 Core and System Performance

The initial core transient response of the plant to the starting of an idle recirculation loop without warming the drive loop water is shown in Figures 15.4-1 and 15.4-2. The neutron flux would show a fairly sharp peak (to 92%) shortly after the actual pump excitation due to the slight peak in core inlet flow (to about 44%). The core flow would subsequently increase slowly to its final value of about 45%. A peak fuel surface heat flux of 82% would occur about 1 minute after the start of the transient. No thermal limits would be violated.

The startup of an idle recirculation loop is not analyzed for the introduction of Westinghouse's reload fuel. This event is considered to be bounded by other events that are analyzed on a cycle-specific basis. Verification that the startup of an idle recirculation loop is bounded by other events was performed for the SVEA-96 Optima2 first reload considering a 50°F temperature difference between loops.

15.4.4.4 Barrier Performance

This transient is not analyzed for reload cores since the fuel-specific MCPR LCO is determined for each reload core based on other events which bound this event. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit.

15.4.4.5 Radiological Consequences

The fuel cladding integrity safety limit would not be violated; therefore a radiological consequence analysis has not been performed.

15.4.5 Recirculation Loop Flow Controller Failure With Increasing Flow

See the introduction to Section 15.4 for information regarding use of details from this analysis description which may not be applicable to the current fuel cycle.

This event (the fast recirculation flow increase transient) is not considered a limiting event and as such is not normally included as an AOO to be performed during reload analyses. The analysis results indicated in Reference 23 show that this event is not limiting for EPU conditions. The CPR result of Reference 23 is bounded by the ARTS flow dependent MCPR limit, $MCPR_F$. The design basis flow increase event is a slow recirculation flow increase which is not terminated by scram. This event is evaluated on a cycle specific basis.

A slow increase in recirculation flow resulting from the flow control failure is analyzed as a part of the Westinghouse reload safety analysis using methodology described in Section 7.5.3 of Reference 32. The event may also be analyzed as a pressurization transient if the maximum combined steam flow limit as specified in the reload transient analysis input document is exceeded during simulation as a slow event (e.g., for certain EOOS conditions).

The slow recirculation flow increase event is also analyzed for the one TCV stuck closed EOOS option. The thermal limits associated with implementation of this EOOS option are provided in the Core Operating Limits Report. The thermal limits for one TCV stuck closed is also applicable to the one TSV stuck closed EOOS option. The plant operation with one TCV stuck closed or one TSV stuck closed is subject to limitation specified in the Core Operating Limits Report.

For a transition core with SVEA-96 Optima2 fuel, Westinghouse has evaluated the fast recirculation flow controller failure for a single pump with ASD implemented for which the details are documented in Reference 49.

15.4.5.1 Identification of Causes

The most severe case of increasing recirculation flow results from the failure of one of the adjustable speed drive (ASD) speed controllers. The rate limits associated with Master Control are low rates in the Control System that do not exceed the individual speed control rate. [15.4-8]

The recirculation loop flow controller failure with increasing flow is classified as an event of moderate frequency.

The recirculation loop flow controller failure with increasing flow (fast transient) is bounded by the event used to define flow-dependent MCPR limits. Refer to the cycle-specific documentation or the Core Operating Limits Report (COLR) for detailed results of current cycle transient analyses.

15.4.5.2 Sequence of Events and System Operation

This section contains historical information applicable to the Quad Cities initial core analysis with Recirculation System M-G sets in place.

The initial core analyses with M-G sets simulated a worst case condition where the failed speed controller moved its positioner to full coupling at about 10% per second. The initial conditions were 65% power and 50% flow to fully bracket the band of expected operating conditions.

For the M-G set control, prior to the failure, both speed control positioners are assumed to be at about the 20% (minimum) position. The functioning positioner remains near this position, while the failed positioner reaches full stroke by 9.0 seconds. Diffuser flow for the jet pumps in the failed loop quickly would increase. It would reach 125% by 5 seconds and would level out near 140%. Diffuser flow for jet pumps in the opposite loop would decrease by the greater core ΔP. Core inlet flow would increase rapidly to about 75% causing neutron flux to increase and to scram the reactor at 2.15 seconds. Refer to the cycle-specific documentation or the Core Operating Limits Report (COLR) for detailed results of current cycle transient analyses. [15.4-9]

For the ASD, the speed control increase can be instantaneous to the over frequency limit, with the motor using full torque capability for acceleration of up to 42% per second from 20% speed to 95.8% speed (57.5 Hz), at which time the pump would maintain this speed or trip if this speed is exceeded. Diffuser flow for the jet pumps in the failed loop quickly would increase. Diffuser flow for jet pumps in the opposite loop would decrease by the greater core ΔP. Core inlet flow would increase rapidly causing neutron flux to increase and to scram the reactor.

15.4.5.3 Core and System Performance

This paragraph and the associated Figures are historical for the M-G sets only. Initial core response of the plant to this transient with M-G set drives is shown in Figures 15.4-3 and 15.4-4. The neutron flux would reach 125%, however, peak fuel surface heat flux would be only 70%, maintaining adequate thermal margins.

For core response with ASD failure to this transient, the results show that the neutron flux reaches 228% and the corresponding clad heat flux reaches 107%. These results remain considerably less limiting than those of other transients analyzed on a cycle-specific basis (see Reference 49).

15.4.5.4 Barrier Performance

The fast recirculation flow increase transient is not analyzed for reload cores since the fuel-specific MCPR LCO is determined for each reload core based on other events which bound this event. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit.

The fast recirculation flow controller failure for a single pump with ASD was analyzed by Westinghouse in a conservative manner as documented in Reference 49. It was determined that relatively large thermal margins for MCPR and LHGR exist when compared to those of other transient events analyzed as part of each cycle's specific reload. Therefore, this event does not need to be analyzed on a cycle-specific basis.

A slow recirculation flow increase is analyzed on a cycle-specific basis, which may establish flow dependent operating limits.

Refer to the cycle-specific documentation or the Core Operating Limits Report (COLR) for detailed results of current cycle transient analyses.

15.4.5.5 Radiological Consequences

The fuel cladding integrity safety limit would not be violated; therefore a radiological consequence analysis has not been performed.

15.4.6 Chemical and Volume Control System Malfunction

This event is not applicable to Quad Cities Station.

15.4.7 Mislocated Fuel Assembly Accident

A mislocated fuel assembly is an assembly which is loaded into an incorrect core position and not subsequently identified and corrected prior to core operation. The fuel assembly could then be monitored incorrectly, possibly resulting in a high reactivity, or limiting assembly being modeled during the cycle with a different reactivity or as a non-limiting assembly. [15.4-10]

The mislocated fuel assembly has a lower frequency of occurrence than moderate frequency, but is evaluated as a moderate frequency event. This event is evaluated on a cycle-specific basis to determine the need for further analysis to establish operating limits.

The mislocated fuel assembly accident has been analyzed in conjunction with the misoriented fuel assembly accident described below. The barrier performance and radiological consequences are presented in Sections 15.4.8.4 and 15.4.8.5. [23]

15.4.8 Misoriented Fuel Assembly Accident

See the introduction to Section 15.4 for information regarding use of details from this analysis description which may not be applicable to the current fuel cycle.

15.4.8.1 Identification of Causes and Frequency Classification

A fuel assembly is misoriented if it is loaded and operated in a position that is rotated from its proper orientation. [15.4-11]

Only the 90° and 180° rotations are investigated; the 270° rotation is equivalent to the 90° rotation due to the symmetry of Westinghouse bundles.

The misoriented assembly has a lower frequency of occurrence than moderate frequency but is evaluated as a moderate frequency event. [15.4-12]

15.4.8.2 Sequence of Events and System Operation

Quad Cities Station is a D-Lattice plant, which has non-uniform water gaps. An undetected and uncorrected misorientation of the fuel assembly may result in larger than anticipated local peaking on the wide-wide side of the fuel assembly, since the wide-wide side has the larger water gap, and hence, greater neutron thermalization. This may lead to a degradation of MCPR margin. [15.4-13]

Verification of the proper orientation of the fuel assemblies is one of the checks of Quad Cities' BWR core loading procedures. All of the following items provide indication of proper orientation:

A. The channel fastener must be located at the corner of the assembly which is placed next to the center of the control blade.

B. For fuels with an orientation boss, this feature points toward the adjacent control rod. In all fuel types, the assembly serial number is engraved on the top of the handle in a standard orientation; specifically, it is readable looking from the center of the control cell.

C. The channel spacer buttons are adjacent to the control rod passage area.

D. There is cell-to-cell replication, meaning the preceding elements form a repeating pattern as a whole, and the handles form a square in each cell.

15.4.8.3 Core and System Performance

For Westinghouse reload cores, the internal power distribution is the key parameter in establishing the impact of the rotation on MCPR. Consequently, the two-dimensional lattice physics code is utilized to determine the impact on CPR; specifically, the internal power distribution factor required for the MCPR evaluation (e.g., "R-factor") is calculated. Calculations are performed to compute R-factors for the case of 4 fresh assemblies, 4 depleted assemblies, and 3 depleted assemblies with one fresh assembly at the nominal orientation and misorientation. Based on the change in power distribution factor associated with the rotation, the change in CPR is calculated at rated power and flow conditions as a function of burnup. A complete description of the Westinghouse analysis methodology and assumptions for the misoriented fuel assembly accident is discussed in Section 8.5.2 of Reference 32.

15.4.8.4 Barrier Performance

The fuel-specific minimum critical power ratio (MCPR) limiting condition for operation (LCO) is determined for each reload core based on bounding events for the cycle, including misoriented fuel assembly event and mislocated assembly event. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit. Refer to the cycle-specific documentation or the Core Operating Limits Report (COLR) for detailed results of current cycle transient analyses.

15.4.8.5 Radiological Consequences

The fuel cladding integrity safety limit would not be violated; therefore a radiological consequence analysis has not been performed.

15.4.9 Control Rod Ejection Accidents (PWR)

These events are not applicable to Quad Cities Station.

15.4.10 Control Rod Drop Accident

See the introduction to Section 15.4 for information regarding use of details from this analysis description which may not be applicable to the current fuel cycle.

15.4.10.1 Identification of Causes and Frequency Classification

The control rod drop accident (CRDA) is defined as a power excursion caused by the accidental removal of a control rod from the core at a more rapid rate than can be achieved by the use of the control rod drive mechanism. In the control rod drop accident, a fully or partially inserted control rod is assumed to fall out of the core after becoming disconnected from its drive, and after the drive has been removed to an intermediate or fully withdrawn position. [15.4-14]

The design of the drive and its coupling uses high quality materials and it receives stringent quality control and testing procedures appropriate to other equipment typically listed in the critical component list for a plant. Additionally, tests conducted under both simulated reactor conditions and conditions more extreme than those expected in reactor service have shown that the drive (or coupling) retains its integrity even after thousands of scram cycles. Tests also show that the drive and couplings do not fail when subjected to forces 20 times greater than that which can be achieved in a reactor. [15.4-15]

Sticking of the control blade in its fully inserted position is highly unlikely because each blade is equipped with rollers or low-friction pads that make contact with the nearly flat fuel channel walls, travelling in a gap of approximately 1/2-inch clearance. Since a control blade weighs approximately 186 pounds, even if it separates from its drive, gravity forces would tend to make the blade follow its drive movement as if it were connected. Control blades of the current design now in use in operating reactors have exhibited limited tendency to stick.

The CRDA is considered a limiting fault. [15.4-16]

The assumed control rod drive/control rod blade separation does not, of itself, produce any unplanned or uncontrolled perturbation on normal plant operation that requires immediate operator action. This event, therefore, is not of immediate reactor safety consequence as is the loss of coolant accident (LOCA) or steam line break accident (SLBA). In most cases, if such a separation occurred, it is expected that the blade would not stick, but rather follow its drive movement. The separation would be detected at the next fully withdrawn stroke where the ability to withdraw to the overtravel position would signal separation, since the blade bottoms on a seat and prevents withdrawal to the overtravel position if connected. Thus, the drive could be inserted and declared inoperable in accordance with the technical specifications until the next refueling outage when it could be repaired. [15.4-17]

The RWM, required to be operable from 0% power to the low power setpoint (LPSP), provides backup to operating procedures and offers additional assurance that, if a CRDA occurs, it will occur for a normal or in-sequence rod worth. If the RWM is not operable in this power range, proper rod movement is verified by an independent observer. Above the LPSP, worths are not high enough to result in a CRDA with unacceptable consequences and therefore the RWM is not required in this range. [15.4-18]

Section 7.7.2 provides a detailed description of the RWM. Section 4.6 provides a description of the control rods and associated equipment.

Operating procedures require rod following verification checks during startup and during major rod movements and weekly verification checks on all rods not fully inserted to ensure that any uncoupling separation would be detected. Procedures require full insertion of rods when rod following cannot be verified below the LPSP. [15.4-19]

Operating procedures require that control rod movements follow preplanned patterns designed to flatten the power distribution. This tends to minimize the reactivity worth of individual rods, so that extensive fuel damage would not be expected if a control rod drop were to occur. The control rod sequences are designed to limit rod drop power excursions to less than 280 cal/gm.

15.4.10.2 Sequence of Events and System Operation

The design basis CRDA would proceed as follows:

A. The reactor has a control rod pattern corresponding to maximum incremental rod worth. [15.4-20]

B. Rod withdrawal sequence control measures (Rod Worth Minimizer or operators) are functioning within constraints of the BPWS (Reference 5) or the Analyzed Rod Position Sequence (Reference 14). The control rod that will result in the maximum incremental reactivity worth addition at any time in core life under any operating condition while employing the BPWS (Reference 5) or the Analyzed Rod Position Sequence (Reference 14) becomes decoupled from the control rod drive.

C. The operator selects and withdraws the drive of the decoupled rod along with the other required control rods assigned to the banked-position group such that the proper core geometry for the maximum incremental rod worth exists.

D. The decoupled control rod sticks in the fully inserted position.

E. The control rod becomes unstuck and drops at the maximum velocity determined from experimental data (3.11 ft/s).

F. The reactor goes on a positive period, with the initial power burst terminated by the Doppler reactivity feedback.

G. An APRM high neutron flux signal scrams the reactor (conservative; in startup mode, IRM would also be operative).

H. The scram terminates the accident.

15.4.10.3 Core and System Performance

The CRDA is initially mitigated by negative Doppler reactivity resulting from the temperature increase of the fuel, and is terminated by scram reactivity. The analysis assumes no void feedback to mitigate the effects of the accident. [15.4-21]

Historically, ComEd utilized the generic General Electric Banked Position Withdrawal Sequence methodology (Reference 5) to protect the 280 cal/gm fuel design limit. This analysis was a bounding and conservative generic calculation. As with most generic analyses, it can also be unnecessarily restrictive. Using in-house ability, ComEd began to perform cycle specific CRDA analyses (References 19, 20, 25 and 26). Using cycle specific calculations, ComEd is able to modify the original BPWS sequence to remove some of the unnecessary conservation (typically elimination of some of the banked positions) (Reference 14). These sequences are referred to as the Analyzed Rod Position Sequence (ARPS).

Control rod patterns analyzed in the cycle specific CRDA analyses follow predetermined sequencing rules. This sequence (ARPS) applies to all control rod movement from the all rods in condition to the Low Power Setpoint (LPSP). These rules include the designation of control rod groups. All control rods within a group must be maintained within the specific bank limits (e.g., between notches 08 and 12). The banked positions are established to limit the maximum incremental control rod worth such that the 280 cal/gm design limit is not exceeded. Cycle specific analyses ensure that the 280 cal/gm fuel design limit is not exceeded during worst case scenario. These worst case scenarios account for a limited number of inoperable control rods (8) with a specified separation criteria. Specific evaluations or analyses can be performed for atypical operating conditions (e.g., fuel leaker suppression).

15.4.10.3.1 Generic Analysis

The Westinghouse methodology (Reference 34) for the CRDA event utilizes the following systematic approach for a cycle-specific evaluation: [15.4-22]

1. Existing results for the plant will be evaluated to determine if they demonstrate that the design criteria will not be exceeded during a postulated CRDA during the cycle under consideration. If existing results are available to demonstrate that design criteria will be satisfied for the most limiting CRDA during the cycle, no further analysis will be performed.

2. Should the CRDA analyses be insufficient to demonstrate that design criteria will be satisfied for the most limiting CRDA during the cycle, a cycle-specific evaluation will be performed.

The control rod worth and the post-drop nodal power peaking conditions were analyzed to establish limiting cases for which transient calculations were performed to ensure that the peak fuel enthalpy, including analysis uncertainties, will be less than the accepted threshold for failed fuel of 170 cal/g.

Simplified shutdown sequences that eliminate the group banking requirements have been generically analyzed in Reference 27.

15.4.10.3.2 Cycle-Specific Analysis

Simplified ARPS patterns may be analyzed on a cycle-specific basis to eliminate overly restrictive bank positions which have been introduced due to the nature of the generic BPWS methodology. These CRDA analyses are based on methodology developed by SPC (with banking of rod groups as discussed in References 5, 13, and 14). Analyses are performed on a cycle-specific basis to determine four different parameters associated with the Control Rod Drop Accident:

1. The maximum dropped rod worth
2. The core average Doppler coefficient
3. The core average delayed neutron fraction (beta)
4. The relative power peaking of the fuel assemblies surrounding the control rod that is postulated to drop.

These cycle-specific parameters are then used with a generic set of parametric curves to determine the energy deposition associated with the CRDA for the cycle of interest. These analyses provide assurance that the 280 cal/gm enthalpy deposition limit will not be violated. This enthalpy deposition is calculated each cycle. (Reference 12)

The analyzed control rod sequence and the CRDA analysis provide confidence that the design limit will not be violated in the unlikely event of the postulated Design Basis CRDA. Even if the 280 cal/gm design limit is not exceeded, fuel failure is still assumed to occur for those fuel rods which exceed 170 cal/gm. The methodology evaluates the number of postulated fuel failures relative to the value previously used to assess radiological consequences (e.g., 850 failed rods for 8x8 and 9x9 fuel).

The CRDA event is analyzed using the Westinghouse methodology with SVEA-96 Optima2 fuel bundles for the entire cycle and control rod sequences. Based on the static and dynamic analyses of control rod worths and nodal power peaking conditions, the peak fuel enthalpy, including analysis uncertainties was evaluated to be less than the accepted cladding failure threshold of 170 calories/gram. The cycle-specific CRDA event results are documented in the reload licensing report.

Input Parameters and Initial Conditions

At the time of the control rod drop accident, the core is assumed to be at an operating cycle point which results in the highest worth of the dropped control rod. The core is also assumed to contain no xenon, to be in a hot-standby condition, and to have the control rods in sequence A and be near critical. The assumption to remove xenon, which competes well for neutron absorptions, increases the fractional absorptions, or worth, of the control rods.

15.4.10.4 Barrier Performance

Since peak fuel enthalpy is the most important single parameter for determining the severity of a transient and the onset of fuel pin failure, results are presented as a function of the resultant peak fuel enthalpy. As reference points, the following design and fuel failure criteria have been accepted by the NRC (NUREG-0800, Section 15.4-9): [15.4-24]

A. Enthalpy = 170 cal/g, cladding failure threshold, [15.4-25]

B. Enthalpy = 280 cal/g, specific energy design limit, and

C. Enthalpy = 425 cal/g, prompt fuel dispersal threshold.

The number of failed fuel rods is calculated to be 660. This is the maximum number of failed fuel rods resulting from the analysis of the rod drop accident for the initial core (7x7) fuel arrays. For a reload core of 8x8 fuel and/or 9x9 fuel, the number of fuel rods conservatively estimated to reach a fuel enthalpy of 170 cal/g, the enthalpy limit for eventual cladding perforation, is 850. For the bounding worst case, 850 fuel rods are assumed to fail as a result of the CRDA. At the uprated power and with GE14 fuel (10x10), the number of failed fuel rods was estimated to be 1153 rods. [15.4-26]

For reloads containing Westinghouse fuel it has been verified that no failed rods can be expected for SVEA-96 Optima2. The fuel enthalpy is below the threshold of 170 cal/g.

15.4.10.5 Radiological Consequences

15.4.10.5.1 Application of Alternative Source Term Methodology

Regulation 10 CFR 50.67, "Accident Source Term," provides a regulatory mechanism for power reactor licensees to voluntarily replace the traditional accident source term used in design basis accident analyses (i.e., TID 14844, Reference 35) with an "Alternative Source Term" (AST). The methodology for this approach is provided in NRC Regulatory Guide 1.183 (Reference 36).

Accordingly, Quad Cities applied for the AST methodology for key Design Basis Accidents (DBAs). In support of a full-scope implementation of AST as described in Reference 36, AST radiological consequence are performed for the four DBAs that result in offsite exposure consequences: Loss of Coolant Accident (LOCA), Main Steam Line Break (MSLB), Fuel Handling Accident (FHA), and Control Rod Drop Accident (CRDA).

The NRC approved AST for Quad Cities in Reference 37.

Fission Product Inventory
The inventory of reactor core fission products is based on maximum full power operation at a power level of 3016 MWth, the Extended Power Uprate (EPU) thermal power of 2957 MWth plus 2% to account for uncertainties in accordance with NRC Regulatory Guide 1.49 (Reference 38). The AST source term values for this analysis were derived using the ORIGEN2 computer code (Reference 39) and the guidance outlined in NRC Regulatory Guide 1.183. Core isotopic nuclides and their curie per megawatt activities were utilized from Appendix D of GE task report GE-NE-A22-00103-64-01 (Reference 40) for input into the RADTRAD computer code, which was used to determine dose consequences (note that the CO-58 and CO-60 activities were obtained from the RADTRAD User's Manual). The SVEA-96 Optima2 fuel was also evaluated and determined to be bounded by the GE results.

Core Inventory Release Fractions
The core inventory release fractions were determined using the guidance in NRC Regulatory Guide 1.183, Section 3.2 (Release Fractions).

Atmospheric Dispersion Factors
The main control room and offsite dose χ/Q values were calculated using the methodology described in UFSAR Section 2.3.6. The atmospheric relative concentrations used are as follows:

Main Control Room
χ/Q = 1.02E-03 sec/m^3 (0-2 hours)
χ/Q = 8.23E-04 sec/m^3 (2-8 hours)
χ/Q = 3.55E-04 sec/m^3 (8-24 hours)

Exclusion Area Boundary (EAB)
χ/Q = 1.36E-03 sec/m^3 (0-2 hours)

Low Population Zone (LPZ)
χ/Q = 1.04E-04 sec/m^3 (0-2 hours)
χ/Q = 4.14E-05 sec/m^3 (2-8 hours)
χ/Q = 2.62E-05 sec/m^3 (8-24 hours)

Release Paths
Following a postulated CRDA, radioisotopes will be transported through the main steam lines directly to the main steam condenser. The main condenser mechanical vacuum pump will isolate on main steam line high radiation; hence the release pathways will include the condenser (1% per day) and gland seal steam system. The activity is postulated to leak from the turbine building to the environment as an unfiltered ground release. An additional leakpath evaluated is through the steam jet air ejectors (SJAE). In this scenario, activity is released to a system of charcoal beds, where iodine nuclides and particulates are effectively removed and only a delayed release of noble gas nuclides occurs. Although this release path would occur through the station chimney, for conservatism, the release path is treated as ground level release.

Dose Consequences
RADTRAD (Reference 41) is used to determine the dose consequences for the CRDA. The CRDA assessment takes no credit for control room isolation, emergency ventilation or filtration of intake air for the duration of the accident event. Dose conversion factors were obtained from Federal Guidance Reports 11 and 12 (References 42 and 43).

15.4.10.5.2 Acceptance Criteria

The AST acceptance criteria for postulated major credible accident scenarios are provided by 10 CFR 50.67 and Regulatory Guide 1.183. For the main control room, adequate radiation protection is provided to permit access to and occupancy under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident. The AST acceptance criteria for an individual located at any point on the boundary of the exclusion area (the Exclusion Area Boundary or EAB) are provided by 10 CFR 50.67 as 25 rem TEDE for any 2-hour period following the onset of the postulated fission product release. The AST acceptance criteria for an individual located at any point on the outer boundary of the low population zone (LPZ) are provided by 10 CFR 50.67 as 25 rem TEDE during the entire period of passage of the radioactive cloud resulting from the postulated fission product release. Regulatory Guide 1.183 applies additional limits to events with a higher probability of occurrence (including MSLB, CRDA, and FHA). For the CRDA, doses at the EAB and LPZ should not exceed 6.3 rem TEDE.

15.4.10.5.3 Computer Models

AST calculations for the CRDA were prepared to simulate the radionuclide release, transport, removal, and dose estimates associated with the postulated accident scenario.

Source term calculations were determined using the ORIGEN2 computer code. The ORIGEN2 computer code was designed for reactor fuel cycle mass and radioactivity inventory calculations. The ORIGEN2 computer code is widely recognized for calculating fission product inventories.

Offsite χ/Qs were calculated with the PAVAN computer code (Reference 44) using the guidance of NRC Regulatory Guide 1.145 (Reference 45); control room χ/Qs were calculated with the ARCON96 (Reference 46) and PAVAN computer codes consistent with the guidance of Regulatory Guide 1.194 (Reference 47). The PAVAN and ARCON96 codes are generally used to calculate relative concentrations in plumes from nuclear power plants at offsite locations and control room air intakes, respectively.

The RADTRAD computer code was used for determining dose consequences for the CRDA. The RADTRAD program is a radiological consequence analysis code used to estimate post-accident doses at plant offsite locations and in the control room.

15.4.10.5.4 Key Plant Assumptions

Power Level:	3016 MWt
Radial Peaking Factor:	1.7
Number of Failed Fuel Rods:	850 (bounding 7X7 array)
Isotopic Release Fraction:	From Regulatory Guide 1.183
Primary System Volume (MSL and Dome):	5000 ft^3
Condenser Vapor Space Volume:	92,000 ft^3
Gland Seal Condenser Release Fraction:	0.0015
Control Room Emergency Zone Volume:	184,000 ft^3
Control Room Proper Volume:	58,300 ft^3
Control Room Intake Flow:	58,300 scfm

15.4.10.5.5 Results

The reanalysis of the CRDA event using AST methodology is documented in calculation QDC-0000-N-1268 (Reference 48). Dose results for the main control room, EAB and LPZ are tabulated in Table 15.4-1b.

UFSAR Section 15.4.10.5.6 describes the radiological assessment performed prior to the adoption of an updated accident source term in accordance with 10 CFR 50.67, "Accident Source Term."

15.4.10.5.6 Historical Information

The following is the CRDA radiological analysis before the extended power uprate. The impact of EPU and the utilization of Optima2 fuel on the calculated doses is discussed at the end of this section.

The pre-uprate CRDA radiological analysis is based on GE Topical Report NEDO-31400A (Reference 21) which justified elimination of the Main Steam Line Rad Monitor (MSLRM) isolation and scram functions for the CRDA. The radiological consequences as determined per the Reference 21 analysis is limiting and forms the basis for the consequences of a postulated CRDA. [15.4-27]

The radiological consequences for pre-EPU conditions are based on methods established in Section 15.4.9 of the Standard Review Plan (NUREG 0800). The methodology used is consistent with that used for the Dresden Nuclear Plant analysis which is similar in design. The radiological consequences of a CRDA to the Control Room operators and the offsite population were assessed with the "Axident" transient control room and site boundary dose analysis program.

Fission product release estimates for the CRDA model analysis performed are based on the following assumptions: [15.4-28]

A. The steam jet air ejectors are in operation and all leakage from the main condenser passes through the augmented off gas system for ultimate release from the main stack.

B. The augmented off gas (AOG) system removes all iodides from the flow passing through the system from the condenser. This is based on the deep bed charcoal filters, which are very efficient at iodine removal.

C. Coincident Loss of Offsite Power occurs at the time of the CRDA. (This is part of SRP 15.4.9 method. This analysis conservatively assumes power is available to allow release to the condenser).

D. A melting of 0.0077 fraction of fuel that is damaged was modeled.

E. 10% of the noble gas inventory and 10% of the radio-iodine inventory of fuel that is damaged but did not undergo melting will be released from the fuel to the coolant during the gap release. 100% of the noble gases and 50% of the iodines contained in damaged rods that did undergo melting as assumed released to the reactor coolant.

F. Per NEDO 31400A, the CRDA is assumed to result in 850 fuel rod failures. The peaking factor is 1.5.

G. Core 102% power was used to account for uncertainty in the core power level.

H. Gap activity from failed fuel is assumed to mix instantaneously with the reactor coolant.

I. 10% of all iodides and 100% of all noble gases are transported to the turbine condensers. 100% of the noble gases are available for release from the turbine and condensers.

J. 90% of the iodides plate out leaving 10% for release. 100% of all iodine isotopes that are released to the gland seal steam condenser is conservatively released to the atmosphere.

K. The turbine and condenser leak to the atmosphere for a period of 24 hours, at which time the leakage terminates.

L. No credit is taken for decay due to the hold up in the gland steam condenser.

M. Atmosphere dispersion factors, breathing rates and dose conversion factors are the same as those used in the dose calculation for the loss of coolant accident.

N. The reactor scrams in six seconds with reduction in steam flow. The mechanical vacuum pump (MVP) trips at the same time. Six seconds of release through the MVP release path is assumed.

O. Steam flow rate after initiation of CRDA is 5% of its full power value.

P. Before the MVP is isolated all exhaust from the main condenser was assumed to release bypassing the offgas system. After the MVP trip all exhaust from the main condenser releases through the AOG.

Q. The exhaust rate from the main condenser was assumed to be 100% per day. This is conservative because we do not credit the Steam Jet Air Ejector radiation isolation.

A source term for the activity to be released is based on the NEDO-31400A guidelines and plant specific data. A model to evaluate the CRDA was developed to include source term contributions from the main steam line and the gland seal steam line, their leakage paths and release points. The plant specific design also includes the untreated release path flow provided by MVP. The total radioactive inventory of the core was determined from TID 14844 methodology (low burnup fuels). The impact of high burnup fuels was also evaluated and determined to be non-limiting with respect to using TID 14844 assumptions.

The licensing basis uses dose conversion factors (DCF's) derived from Regulatory Guide 1.3 and TID-14844. The basis for these factors is ICRP-2, published in the early 1960's. R. G. 1.109 and ICRP-30 recommend use of significantly lower DCF's. This dose assessment uses the higher DCF's from the ICRP-2 to calculate thyroid doses from inhalation of iodine isotopes.

The plant specific NEDO 31400A scenario 2 is based on an accident analysis model with no closure of the main steam line isolation valves on high radiation. Two release paths, 1) main steam to the turbine condenser and 2) gland seal steam to the gland steam condenser. The plant specific design also assumes the MVP operates until it is tripped by the main steam high-high radiation trip. This model assumes no ground level release paths due to the continued operation of the Steam Jet Air Ejectors (SJAE) which provides the offgas system flow.

For Control Room dose the Control Room was modeled as one volume with instantaneous air mixing of the air intake. The model is based on 110% of design flow (i.e. 2200 cfm) with an in-leakage flow of 260 cfm, no filtered intake or recirculation, and an exhaust flow equivalent to the total.

The key plant data used for the analysis is as follows:

Flow rate of Main Steam	9.700 E6 lb/hr
Flow rate of Gland Steam	1.50 E4 lb/hr
Specific volume of steam	0.445 ft^3 /lb

Volume of primary system (MSL and dome) (conservatively reduced from original 8,431 ft^3)	5,000 ft^3
Design flow rate of mechanical vacuum pump	5,000 ft^3 /min
Free Volume of the condenser	92,000 ft^3
Volume of water in the condenser	11,000 ft^3
Power Level MWt	2561 Mwt
Volume of Control Room Emergency zone	184,000 ft^3
Volume of Control Room Proper	58,300 ft^3
Normal Control Room air supply Flow Rate (unfiltered)	2000 scfm

The analysis calculated the release on the basis of the total flow going to each of the two release paths. The releases were weighted by the ratio of the flow per path to the total flow and then summed to obtain doses at the Control Room (CR), Low Population Zone (LPZ), and the Exclusion Area Boundary (EAB). Table 15.4-1 shows the individual contribution from each path and the total dose released.

NEDO-31400A analyzes two CRDA scenarios. Scenario 2 is the limiting case and defines the Quad Cities consequences. Scenario 1 involves an automatic MSIV isolation and therefore the SJAE would not be in operation. The Standard Review Plan (SRP) assumes the volume contained in the condenser would leak to the environment at the rate of 1% per day. If the SJAE under scenario 2 were to isolate or not operate, then the calculated dose would be equivalent to NEDO 31400A scenario 1. Per the offsite dose criterion in SRP Section 15.4.9 the doses for scenario 1 were 5.7% and 5.2% of the limits for thyroid and whole body dose respectively.

The MSL radiation monitor trip setpoint is at 15 times full power background radiation without hydrogen addition on to accommodate the radiation levels expected from hydrogen injection into the coolant system (increased N-16 activity levels). The normal does rates at the MSL Rad Monitors without hydrogen addition at full power are higher than 100 units/hr. The MSL Rad Monitors are Numac Log-Rad monitors, which provide indication in units. Therefore, 100 units/hr is used as a conservative value for determining the setpoint. The expected dose rate at the MSL radiation monitors following a CRDA is 8 R/hr. The setpoint is set as low as possible for rapid detection that a CRDA event has occurred to provide a trip of the mechanical vacuum pump, but avoiding unnecessary trips. It is not expected that the mechanical vacuum pump would be running at or near full power operations, however the vacuum pump is conservatively tripped due to the potential untreated release path to the main stack.

Radiological consequences of this event for reloads containing Siemens ATRIUM-9B fuel are based on 850 failed rods. [15.4-29]

Fuel Type	GE14	ATRIUM-9B	GE8	GE9 & GE10	7x7
Fuel Rods Per Assembly	87.33	72	62	60	49

Given the number of fuel rods per assembly as shown above, the activity of a Siemens ATRIUM-9B fuel rod is either 62/72 = 0.86 or 60/72 = 0.83 times the activity of a GE 8x8 fuel rod and 49/72 = 0.68 times the activity of a GE 7x7 fuel rod. Therefore, the radiological consequences due to a failure of 850 Siemens ATRIUM-9B fuel rods are less than the radiological consequences resulting from the failure of 850 GE 8x8 fuel rods or 660 GE 7x7 fuel rods.

The resulting CRDA radiological exposures as evaluated per GE NEDO-31400A methodologies with plant specific release paths well below the SRP and 10 CFR 100 limits. Table 15.4-1 provides a calculated value for doses at the CR, LPZ, and EAB locations. Table 15.4-1 shows the Control Room doses are within the requirements of GDC19 of 10 CFR 50 App. A.

At the uprated power of 3016 MWt (including 2% instrument error) and with a GE14 fueled core, the amount of failed fuel was estimated to be 1153 rods. The percentage of fuel failure for EPU is bounded by the conservative 850 failed rods for a core with 63 rods per assembly as described above for the CRDA before EPU was implemented.

The extended power uprate will increase the halogen and noble gas inventory in the fuel. The increased steam flow rate of EPU will also increase the activity released via the mechanical vacuum pump exhaust path. The thyroid, whole body, and beta dose at the site boundary and in the control room contributed by the augmented off-gas system release path and gland steam release path, listed in Table 15.4-1 (based on TID-14844 fuel inventory methodology) will be increased by 27%, 18% and 18%, respectively. The doses contributed by the mechanical vacuum pump release path will increase by an additional 8% due to increased transfer of activity to the main condenser before the mechanical vacuum pump is isolated. The increased doses are noted in Table 15.4-1a. The table shows that all doses are well below the 10CFR100 thresholds and within GDC19 limits.

The CRDA event was analyzed using the Westinghouse methodology for both GE14 and SVEA-96 Optima2 fuel bundles for the entire cycle and control rod sequences. Based on the results of the analyses, the peak fuel enthalpy, including analysis uncertainties, was evaluated to be less than the accepted cladding failure threshold of 170 calories/gram. The radiological consequences calculated above remain valid after the introduction of the SVEA-96 Optima2 fuel. An analysis was done which determined that 980 Optima2 rods would be required to fail to produce the equivalent radiological consequences of the existing CRDA analysis [31].

[END OF HISTORICAL INFORMATION]

15.4.11 Thermal Hydraulic Instability Transient

This section covers events that result in a thermal hydraulic instability transient. Additional information regarding the transient and the system designed to respond to it, namely the Oscillation Power Range Monitor (OPRM) system, is contained in Chapters 4 and 7.

15.4.11.1 Identification of Causes and Frequency Classification

Events such as Reactor Recirculation (RR) pump trips and runbacks, turbine/generator runbacks, loss of feedwater heating, and RR flow controller failures can result in unplanned entry into the high power and low flow region of the power to flow map. Under these conditions, axially varying moderator density in the fuel channels can cause flux oscillations that increase in amplitude. Without manual or automatic suppression, such oscillations can cause the MCPR Safety Limit to be exceeded (Reference 28).

This event is controlled by a system designed for detection and suppression of oscillations in accordance with GDC 10 and 12. The system is the Oscillation Power Range Monitor (OPRM) system, which provides automatic protection for this event when it is installed and fully functional. For operation prior to the installation of OPRM, or when OPRM is not fully functional, the operator provides protection from this event by scramming the reactor upon recognition of an instability or upon entry into the scram region of the power to recirculation flow map.

Anticipated stability-related neutron flux oscillations are those instabilities that result from normal operating conditions, including conditions resulting from anticipated operational occurrences. This category of events is equivalent to the standard terminology for the analysis of events of moderate frequency (Reference 29).

15.4.11.2 Sequence of Events and System Operation

For this event, the plant must be operating in mode 1.

A. As a result of some manual actions or equipment problems (e.g., RR pump runback, loss of feedwater heating), the core power and flow combination may be such that oscillations of neutron flux may be possible.

B. Due to forced flow being inadequate to control density wave transit time up the fuel channels, flux oscillations start and begin to increase in amplitude.

C.1 Without OPRM being installed, armed, and operational, the operator manually scrams the reactor upon recognition of the instability.

C.2 With the OPRM installed, armed, and operational, the operator may be able to take action based on pre-trip alarms to insert control rods or increase flow. If the operator is not able to take action because of the rate of increasing oscillations, the OPRM automatically scrams the reactor before the Safety Limit MCPR is violated.

15.4.11.3 Core and System Performance

The OPRM system contains 4 LPRMs per OPRM cell (using the Bockstanz-Lehmann LPRM assignment methodology described in Reference 30) and requires 1 LPRM input for the cell to be operable. The amplitude setpoint for oscillation magnitude and the number of confirmation counts are specified for the analysis. Since core thermal hydraulic instability is characterized by a consistent period for the oscillations, the OPRM logic includes a check for a set number of consecutive counts as well as a magnitude.

The specified system setpoints are used to determine the hot bundle oscillation magnitude. This information is used, along with empirical data applicable to the fuel in the core, to determine the fractional change of CPR (ΔCPR/IMCPR, where IMCPR is initial MCPR).

The Initial (pre-oscillation) MCPR (IMCPR) is determined as the lower of the following:

1. The MCPR following a dual RR pump trip from rated power on the highest allowed flow control line, after the coastdown to natural circulation and after feedwater temperature reaches equilibrium. The assumption is that the core was operating at the Operating Limit MCPR prior to the dual pump trip.

2. The MCPR Operating Limit with the reactor at steady state conditions at 45% core flow on the highest allowed flow control line.

The Final MCPR (FMCPR) is determined using the IMCPR and ΔCPR/IMCPR data (Reference 30).

The FMCPR is then verified to be greater than the Safety Limit MCPR.

Alternatively, by using the same equation and selecting a FMCPR equal to a given Safety Limit, a minimum IMCPR for that Safety Limit can be determined and verified to be less than the cycle specific Operating Limit.

If the FMCPR is less than the Safety Limit MCPR, or the minimum IMCPR is greater than the Operating Limit determined from other cycle analyses, the system setpoint may be changed and the reload confirmation performed again. Alternatively, the Operating Limit MCPR may be changed, or the LPRM assignment scheme may be modified.

The above is confirmed for each cycle as part of the reload analysis.

15.4.11.4 Barrier Performance

Since the successful completion of this analysis demonstrates that the MCPR Safety Limit is not exceeded, fuel-cladding integrity is not challenged.

15.4.11.5 Radiological Consequences

Since fuel-cladding integrity is not challenged, there are no radiological consequences warranting evaluation for this event.

15.4.12 References

1. Letter from R.E. Engel (GE) to T.A. Ippolito (NRC), Control Rod Withdrawal Error, August 28, 1981.

2. Letter from L.S. Rubenstein (NRC) to R.E. Engel (GE), Change in General Electric Analysis of Rod Withdrawal Error, November 25, 1981.

3. "General Electric Fuel Bundle Designs," General Electric Company, NEDE-31152-P.

4. Deleted.

5. General Electric, "Banked Position Withdrawal Sequence," NEDO-21231, January 1977.

6. Williamson, H.E., and Rowland, T.C., "Performance of Defective Fuel in the Dresden Nuclear Power Station," APED-3894, 1962.

7. L.C. Watson, et al, "Iodine Containment by Dousing NPD-II," AECL 1130, October 1960.

8. H.R. Diffey, et al., "Iodine Cleanup in a Steam Suppression System," International Symposium on Fission Product Release and Transport Under Accident Conditions," April 1956.

9. Dresden Unit 2 PDAR, AEC Docket 50-237.

10. "Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor," General Electric Company, APED-5756.

11. Letter from T.M. Ross (NRC) to T.J. Kovach (CECo), Revised SER for the Hydrogen Water Chemistry Control System, August 24, 1989.

12. SPC Document, Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A), Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, Richland, WA, March 1983.

13 Letter, B.L. Siegel to T.J. Kovach, "Revised Control Rod Sequencing Methods for the Dresden and Quad Cities Nuclear Power Stations," September 21, 1990.

14. Letter, P.L. Piet to T.E. Murley, "Dresden Nuclear Power Station Units 2 and 3, Quad Cities Nuclear Power Station Units 1 and 2, LaSalle County Nuclear Power Station Units 1 and 2, Revised Control Rod Sequencing Methods," January 27, 1993.

15. XCOBRA-T: A Computer Code for BWR Transient Thermal Hydraulic Core Analysis, XN-NF-84-105(A), Volume 1 and Supplements, Exxon Nuclear Company, Inc., Richland, WA, February 1988.

16. Deleted.

17. SPC document, "Exxon Nuclear Methodology for Boiling Water Reactors: Benchmark Results for the CASMO-3G/MICBURN-B Calculation Methodology," XN-NF-80-19(P)(A), Volume 1 Supplement 3, Advanced Nuclear Fuels Corporation, November 1990.

18. Deleted.

19. NRC SER Letter, February 27, 1992, Siegel (NRC) to Kovach (ComEd) "topical Report for Neutronics Methods for BWR Reload Design for Commonwealth Plants" (GE Methods).

20. NRC SER Letter, March 22, 1993, Patel (NRC) to Kovach (ComEd) "Commonwealth Edison Company Topical Report NFSR-0091, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods" (SPC Methods).

21. General Electric NEDO-31400A, Safety Evaluation for Eliminating the Boiler Water Reactor Main Steam Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor, " Class I October 1992.

22. Calculation QDC-9400-M0550, Rev. 1. "Control Room and Site Boundary Radiation Doses Following a CRDA".

23. "Dresden and Quad Cities Extended Power Uprate, Task T0900: Transient Analysis," GE-NE-A22-00103-10-01 Revision 0, October 2000.

24. Calculation BNDL: 98-003, Rev. 3, "Acceptance Review of GE Letter NSA98-071 on Rod Drive Speed, " July 8, 1998.

25. Letter to U.S. Nuclear Regulatory Commission, "Notification of Intent to Perform Analyses Using Vendor Safety Analyses Codes," Letter R3-03-174, September 19, 2003.

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

27. "Improved BPWS Control Rod Insertion Process," NEDO-33091-A, Revision 2, July 2004.

28. Nuclear Regulatory Commission Generic Letter 94-02, "Long Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors," July 11, 1994.

29. NEDO 31960, "BWR Owners Group Long-Term Stability Solutions Licensing Methodology," Supplement 1, March 1992.

30. NEDO 32465, "BWR Owners Group Reactor Stability Detect and Suppress Solution Licensing Basis Methodology and Reload Application," May 1995.

31. Calculation QDC-000-N-1020, "Impact of Extended Power Uprate on Site Boundary and Control Room Doses from LOCA and Non-LOCA Events," Revision 001A, Attachment E, "Evaluation for Impact of Westinghouse Optima2 Fuel on Existing UFSAR Design Basis Accidents."

32. "Reference Safety Report for Boiling Water Reactor Reload Fuel," Westinghouse Topical Report CENPD-300-P-A, July 1996.

33. "Westinghouse BWR Reload Licensing Methodology Basis for Exelon Generation Company Quad Cities Nuclear Power Station Units 1 and 2," WCAP-16334-P, June 2005.

34. "Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification," Westinghouse Topical Report, CENPD-284-P-A (Proprietary), July 1996.

35. U. S. Atomic Energy Commission, Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," 1962.

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

37. Letter from M. Banerjee (U. S. NRC) to C. Crane (Exelon Corporation), Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2 - Issuance of Amendments Re: Adoption of Alternative Source Term Methodology," dated September 11, 2006 [SER correction letter: D. Collins (U. S. NRC) to C. Crane (Exelon Corporation), "Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2 – Correction of Safety Evaluation for Amendment Dated September 11, 2006," dated September 28, 2006].

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

39. ORNL/TM-7175,"A Users' Manual for the ORIGEN2 Computer Code," A. G. Croff, July 1980.

40. GE Task Report No. GE-NE-A22-00103-64-01, Rev. 0, Project Task Report T0802, "Dresden and Quad Cities Asset Enhancement Program - Radiation Sources and Fission Products," August 2000.

41. NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," April 1998 and Supplement 1, June 1999.

42. U.S. Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988.

43. U.S. Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," 1993.

44. Atmospheric Dispersion Code System for Evaluating Accidental Radioactivity Releases from Nuclear Power Stations; PAVAN, Version 2; Oak Ridge National Laboratory; U.S. Nuclear Regulatory Commission; December 1997.

45. NRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," February 1983.

46. Atmospheric Relative Concentrations in Building Wakes; NUREG/CR-6331, PNNL-10521, Rev. 1; prepared by J. V. Ramsdell, Jr., C. A. Simmons, Pacific Northwest National Laboratory; prepared for U.S. Nuclear Regulatory Commission; May 1997 (Errata, July 1997).

47. NRC Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003.

48. QDC-0000-N-1268, "Reanalysis of Control Rod Drop Accident (CRDA) Using Alternative Source Term."

49. NF-BEX-08-133, Revision 1, "Evaluation of the Planned Implementation of Adjustable Speed Drives in Quad Cities Units 1 and 2," February 2009.

Table 15.4-1

Summary of Doses Calculated at LPZ, Control Room, and EAB for TID and Siemens Source Terms
Pre-Uprate Conditions

Fuel Type	Path	LPZ Dose (Rem)			Control Room Dose (Rem)			EAB Dose (Rem)		
		Thyroid	Whole Body	Beta	Thyroid	Whole Body	Beta	Thyroid	Whole Body	Beta
TID-14844	MVP	0.147	3.99E-2	1.51E-2	3.08	9.38E-3	0.188	1.34	0.361	0.137
	AOG	0	0.440	0.258	0	0.208	4.21	0	2.26	0.973
	Gland	0.892	2.69E-2	9.79E-3	18.7	7.15E-3	0.125	8.09	0.244	8.88E-2
	Total	1.04	0.507	0.282	21.8	0.224	4.53	9.43	2.86	1.20
Siemens 20 GWd/MTU	MVP	0.130	2.98E-2	1.02E-2	2.73	5.93E-3	0.114	1.18	0.270	9.27E-2
	AOG	0	0.285	0.152	0	0.123	2.38	0	1.59	0.619
	Gland	0.789	2.02E-2	6.75E-3	16.5	4.82E-3	7.86E-2	7.15	0.184	6.11E-2
	Total	0.919	.0335	0.169	19.2	0.134	2.57	8.33	2.04	0.773
Siemens 60 GWd/MTU	MVP	0.136	2.70E-2	8.92-E-3	2.86	4.50E-3	9.29E-2	1.23	0.244	8.08E-2
	AOG	0	0.226	0.126	0	8.80E-2	1.87	0	1.36	0.514
	Gland	0.823	1.85E-2	5.94E-3	17.3	3.94E-3	6.56E-2	7.46	0.168	5.39E-2
	Total	0.959	0.271	0.141	20.1	9.64E-2	2.03	8.69	1.77	0.649

Note: AXIDENT computer runs were made assuming all the release passing through one release path at a time. The results presented in the table were weighted with the actual release fraction through each path.

This dose data does not apply to current fuel cycles.

Table 15.4-1a
Control Rod Drop Accident
EAB, LPZ and Control Room Doses following EPU

(Historical Information)

Location	Organ	Dose (Rem)	Regulatory Dose Limit (Rem)
EAB			
	Thyroid	12.1	75
	Whole Body	3.41	6.25
LPZ			
	Thyroid	1.33	75
	Whole Body	0.60	6.25
Control Room			
	Thyroid	28	30
	Whole Body	0.27	5
	Beta	5.35	30

Table 15.4-1b

Control Rod Drop Accident
EAB, LPZ and Control Room Doses
(Alternative Source Term)

	EAB (Rem TEDE)	LPZ (Rem TEDE)	CR (Rem TEDE)
Acceptance Criteria	6.3	6.3	5
Main Condenser Leakage	0.161	0.0265	0.233
Gland Seal Condenser Leakage	1.43	0.109	0.792
SJAE Release	1.58	0.121	0.0413
Total Main Condenser and Gland Seal Condenser	1.591	0.136	1.03
Total SJAE and Gland Seal Condenser	3.01	0.230	0.833

15.5 INCREASE IN REACTOR COOLANT INVENTORY

[Start of HISTORICAL INFORMATION]

GE has determined that the analysis of Inadvertent HPCI Startup (IHPCIS) event is not required for reload licensing if the Loss of Feedwater Heating (LFWH) event is shown to be bounding based on more limiting core inlet subcooling. It is implicitly assumed in this methodology that the reactor water high level trip would not occur following an IHPCIS. Were the reactor water high level trip to occur, the IHPCIS event would be a pressurization (i.e., turbine trips on high water level) event superimposed on a subcooling event. In this case, it is likely that IHPCIS would not be bounded by LFWH, and may potentially be worse than the Feedwater Controller Failure (FWCF) event.

Starting with Quad Cities Unit 2 Cycle 18 reload analysis, GE has reviewed this event with respect to the HPCI trip and reactor water high level turbine trip setpoints and concluded that a high water level turbine trip may not be avoided. Therefore, inadvertent HPCI calculations were performed in all of the same operating domains (ICF, MELLL, FWTR, Bypass OOS) as performed for the FWCF event. The results show that this event is bounded by the FWCF event. GE concluded that the FWCF event with appropriate penalties, bound HPCI and no further OLMCPR adjustment was needed. This evaluation will be confirmed for the future reloads and the results will be documented in the supplemental reload analysis report.

[End of HISTORICAL INFORMATION]

Westinghouse has analyzed IHPCIS as a core-wide pressurization transient for all the operating domains licensed for Quad Cities. In this case, it is assumed that the HPCI pump trip would not occur following an IHPCI event where the reactor water level would reach the high water level trip setpoint. Then the reactor water high level would initiate a turbine trip and the IHPCI would become a pressurization event superimposed on a subcooling event like the FWCF. In this case, the IHPCI event could be potentially more limiting than the FWCF. The results are documented in the reload licensing report.

15.5.1 Inadvertent Initiation of High Pressure Coolant Injection During Power Operation

15.5.1.1 Identification of Causes and Frequency Classification

Inadvertent startup of the high pressure coolant injection (HPCI) system is postulated for this analysis, i.e., operator error. This transient disturbance is categorized as an incident of moderate frequency. [15.5-1]

Inadvertent startup of the high pressure coolant injection (HPCI) system requires multiple equipment failures or operator error.

MCPR operating limits are defined such that the MCPR fuel cladding integrity safety limit is not violated during the occurrence of this transient.

15.5.1.2 Sequence of Events and Systems Operation [15.5-2] [15.5-3]

For operation prior to EPU, various power-flow initial conditions were analyzed for the IHPCIS event. The 100% power, 100% recirculation flow case represented the most limiting of the cases analyzed.

The EPU transient analysis in Reference 5 contains information regarding this event as analyzed by GE.

The sequence of events for the event analyzed by Westinghouse is similar to the FWCF transient described in Section 15.1.2. The IHPCIS event also shows similar results compared to the FWCF event. Both sets of analyses include cases at rated and off-rated power and flow conditions to cover the expected range of operation. See Westinghouse cycle-specific reload reports for details applicable to the current analysis of this event.

15.5.1.3 Core and System Performance

[Start of HISTORICAL INFORMATION]

Prior to EPU, the analysis of this event was based on a number of assumptions. For the initial core, it was assumed that at time zero, 5600 gal/min of 40°F water was admitted to the vessel and mixed with the much warmer feedwater. The level control system, sensing additional water causing a level rise, cuts back on the feedwater flow entering the vessel. The colder water increases the core inlet subcooling, and due to the negative void reactivity coefficient, an increase in core power results. All analysis was done assuming the plant was initially on manual flow control; this results in no recirculation flow decrease and thus permits the maximum power increase due to the subcooling change. [15.5-3a]

The EPU transient analysis in Reference 5 contains information regarding this event as analyzed by GE.

[End of HISTORICAL INFORMATION]

For reload cores licensed with Westinghouse methods, the results and assumptions for this event are provided in the cycle-specific reload reports. At the start of the event the same initial HPCI flow rate is assumed (5600 gal/min), delivered to the feedwater system at near freezing conditions (32°F). No credit is taken for a reduction in feedwater flow in response to the increase in water level vessel inventory. Instead, it is conservatively assumed that feedwater and HPCI continue to operate until the time of high water level turbine trip and feedwater pump trip. As a result of this conservative modeling, the IHPCIS event can become a limiting pressurization transient similar to the FWCF event. Results are provided in the cycle-specific reload reports.

15.5.1.4 Barrier Performance

This transient is analyzed for each cycle since the MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit. See Section 15.5 for details. [15.5-4]

15.5.1.5 Radiological Consequences

The fuel cladding integrity safety limit would not be violated; therefore a radiological consequence analysis has not been performed.

15.5.2 References

1. "General Electric Standard Application for Reactor Fuel" (GESTAR II), June 2000, General Electric Co., NEDE-24011-PA-14, and U. S. Supplement, NEDE-24011-PA-14-US.

2. Deleted.

3. Deleted.

4. Deleted.

5. “Dresden and Quad Cities Extended Power Uprate, Task T0900: Transient Analysis,” GE-NE-A22-00103-10-01 Revision 0, October 2000.

15.6 DECREASE IN REACTOR COOLANT INVENTORY

This section covers postulated transients which involve an unplanned decrease in reactor coolant inventory. These events include inadvertent opening of a safety valve, relief valve, or safety relief valve, failure of an instrument line carrying reactor coolant outside primary containment, main steam line break outside primary containment, and loss of coolant resulting from the failure of pipes within the reactor coolant pressure boundary.

Because they continue to be bounded by analyses for previous fuel cycles, the events and radiological consequences described in this section have not been reanalyzed for the current fuel cycle. The conclusions of these radiological analyses are still valid; however, specific details contained in the descriptions and associated results and figures should be used only to understand the analysis and its conclusions. These specific details should not be used as sources of current fuel cycle design information.

For many of the following events, radiological analyses were performed by both CECo and the AEC/NRC for initial licensing of the plant. Since the initial licensing, Quad Cities has changed from 7x7 fuel arrays to 8x8 fuel arrays, 9x9 fuel arrays, and 10x10 fuel arrays. As described in this section, radiological evaluations have shown that the postulated releases for 8x8, 9x9, and 10x10 arrays would be less than those for 7x7 arrays.

15.6.1 Inadvertent Opening of a Safety Valve, Relief Valve, or Safety Relief

This event was not a specific consideration in the original licensing of Quad Cities, and the following evaluation shows that it is not of safety significance. The following is based on the NRC approved evaluation of a similar BWR/3 (Dresden Unit 2) performed during the Systematic Evaluation Program.

The inadvertent opening of a safety valve, relief valve, or safety relief valve (SRV) would result in a decrease in reactor coolant inventory and a decrease in reactor coolant system pressure. [15.6-1]

If a SRV or relief valve failed open, it would discharge to the suppression pool. The safety valves discharge directly to drywell atmosphere. Although a drywell high pressure reactor trip may occur if a safety valve failed open, the following analysis assumes a safety valve discharge would result in a sequence of events similar to a relief valve or SRV discharge.

15.6.1.1 Identification of Causes

The cause of an inadvertent opening of a safety valve, relief valve, or SRV is attributed to malfunction of the valve.

The inadvertent opening of a safety valve, relief valve, or safety relief valve are bounded by limiting events specified in the cycle-specific reload licensing documents and/or the COLR.

15.6.1.2 Sequence of Events and Systems Operation

The following sequence of events is assumed for this analysis.

The normal functioning of plant instrumentation and controls is assumed for this incident; specifically, the operation of the pressure regulator and vessel level control systems is assumed normal. On relief valve or SRV opening, the pressure regulator would sense the pressure decrease and would cause the turbine control valves to partially close. No reactor trip occurs, and the reactor conditions stabilize at a power level near the initial power. The feedwater system would be used to make up the continuing loss of reactor coolant inventory.

If the pressure regulator fails to respond, the decrease in main steam line pressure would cause the main steam isolation valves (MSIVs) to close. This event is discussed in Section 15.1.3.

If the feedwater system became unavailable due to a single failure or loss of offsite power, the HPCI system could provide makeup water. HPCI would be automatically actuated on low-low water level.

If a relief valve or SRV opens and fails to reclose, the torus would experience an increase in temperature. Closure of the MSIVs could not halt the blowdown since the relief valves and SRV are upstream of the MSIVs. [15.6-2]

15.6.1.3 Core and System Performance

This event is not limiting from a core performance standpoint.

Inadvertent opening of a safety valve, relief valve, or SRV would cause a negligible pressure reduction which could lead to partial closure of the turbine control valve by the pressure regulator. The net change in power level and coolant conditions within the fuel assemblies would be negligible, and operating thermal margins would be relatively unaffected. Therefore, minimum critical power ratio (MCPR) would not change significantly.

Refer to Section 6.2.1.3 for suppression pool temperature and pressure response to an opened relief valve or SRV.

15.6.1.4 Barrier Performance

The NRC has concluded,[1] based on the evaluation of other similar plants,[2, 3] there would not be any fuel failure resulting from a stuck open safety valve, relief valve or SRV event since MCPR would not change significantly. Therefore, the transient resulting from an inadvertently opened safety valve, relief valve, or SRV would not have a significant effect on the reactor coolant pressure boundary and would not violate the fuel clad integrity limit.

15.6.1.5 Radiological Consequences

The consequences of this event would not result in fuel failure. Discharge of normal coolant activity to the suppression pool via SRV or relief valve operation would result. This activity would be contained in the primary containment. Any discharges to the environment may be made under controlled release conditions. During purging of the containment, the release would be in accordance with the established limits; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.6.2 Break in Reactor Coolant Pressure Boundary Instrument Line Outside Containment

A break of a reactor coolant pressure boundary instrument line outside primary containment could result in the pressurization of secondary containment and the release of radioactive material to the environs. The response of the secondary containment to an instrument line break is discussed in Section 6.2.3.3. The following section describes an instrument line break and the potential radiological consequences.

See the introduction to Section 15.6 for information regarding use of details from this analysis description which may not be applicable to the current fuel cycle.

The remainder of Section 15.6.2 discusses the original GE disposition of this event.

15.6.2.1 Identification of Causes [15.6-3]

A postulated 1 inch reactor coolant pressure boundary instrument line break has been analyzed.

15.6.2.2 Sequence of Events and Systems Operation

The break was assumed to occur outside the primary containment but upstream of the flow check valve in the line. A manually operated stop valve is located outside the containment wall upstream of the break. This valve was not assumed to be closed until after the reactor was shutdown and depressurized. The reactor was considered to be shutdown manually by the operator upon detection of the break.

Routine surveillance on the part of the operator as given in the following items A through G has been a sufficient program for the periodic testing and examination of the valves in these small diameter instrument lines.

Such leaks could be detected by one or a combination of the following:

A. Comparison of readings among several instruments monitoring the same process variable such as reactor level, jet pump flow, steam flow, and steam pressure;

B. Annunciation of the failure of the affected control function, either high or low, in the control room;

C. Annunciation of a half-channel scram if the rupture occurred on a reactor protection system instrument line;

D. A general increase in the area radiation monitor readings throughout the reactor building;

E. Audible noise from the leakage heard either inside the turbine building or outside the reactor building on a normal tour;

F. Unexplained increased in floor drain collector tank water level; and

G. Increases in area temperature monitor readings in the reactor building.

Calculations of doses due to the released radioactive materials included the following assumptions. A coolant activity consistent with a plant off-gas release rate of 100,000 micro-Curies/sec was assumed to be released to the environment. While the release occurs at the top of the reactor building, it was assumed that downwash occurs resulting in an effective release height of 0-meters. No core uncovering occurs and no perforations occur, so only coolant activity is released. Iodine in the 30,000 pounds of water that flashed to steam was assumed to be transported with the steam.

The description that follows was a response to a follow-up question concerning a postulated 1” instrument line break within secondary containment, during initial licensing of Quad Cities Station. The AEC requested Quad Cities Station to specifically provide assurance that the integrity of secondary containment would be maintained and that the building filters (Standby Gas Treatment) would not be bypassed.

The instrument line break has been re-evaluated from the standpoint of expected pressure buildup inside the secondary containment of a two-unit plant with the use of the SBGTS. The second analysis described below (and in the following barrier performance section) is historical and is not currently relied upon for plant activities. The assumptions included the original proposed technical specification coolant activity of 20 micro-Curies/cc total Iodine, with an isolation of RB ventilation and SBGT system auto-start resulting from a vent duct high radiation trip signal.

[Start of historical analysis]

The SBGTS is assumed to be initiated on high radiation in the ventilation system and continuously remove 4000 cfm from the secondary containment.

The analysis assumed that over a 3-hour period 100,000 pounds of liquid is released to the reactor building from which 30,000 pounds of steam is formed.

Reactor building pressure would start to increase, thereby causing back pressure to be seen by the ventilation supply fan such that essentially no air flows into the reactor building.

While the blowdown would occur over a 3-hour period, a zero pressure differential between the reactor building and atmosphere would be reached in 3000 seconds, as shown in Figure 6.2-37, after which time the differential pressure would be negative and all release would be via the SBGTS and the main chimney. During the first 3000 seconds, 43,000 pounds of mass would be discharged to the secondary containment of which 14,000 pounds would be flashed to steam. Of the steam mass, 6,500 pounds would be released unfiltered and 7,450 pounds would be released via the SBGTS.

[End of historical analysis]

15.6.2.3 Barrier Performance

No core uncovering would occur and no perforations would occur, so only coolant activity would be released.

15.6.2.4 Radiological Consequences

This section describes instrument line break radiological dose calculations based on the above transient analyses which were performed for the initial licensing of the plant.

The iodine activity associated with the released liquid was 0.04 micro-Curies/cc of I-131 and 0.3 micro-Curies/cc of I-133. No further release of iodine from the water was assumed. Very stable 1 meter/sec meteorological conditions were assumed since this is the worst case for an equivalent ground level release. Calculated lifetime thyroid dose for the duration of the release of 0.3 Rem is well below the reference doses of 10CFR100 and is in fact less than the annual dose permitted in 10CFR20.

The iodine activity associated with the released liquid of 0.04 mciro-Curies/cc of I-131 and 0.3 micro-Curies/cc of I-133 equates to 0.121 I-131 dose equivalent. The Technical Specification shutdown Limiting Condition for Operation (LCO) limit for coolant activity of 4.0 micro-Curies/gram (1 gram = 1 cc of reactor coolant) has been used to determine dose consequences. Dose versus coolant activity is a linear relationship, and for coolant activity of 4.0 micro-Curies/cc of I-131 dose equivalent, the difference is conservatively a multiple of 40. The calculated lifetime thyroid dose for the duration of the release is therefore 12.0 Rem, which is well within 10CFR100 limits. An evaluation conducted using available emergency preparedness models confirmed that the original dose consequences analysis remains conservative with respect to current industry specifications on allowable concentrations of reactor coolant activity for I-131 dose equivalent.

[Start of historical analysis]

The analysis described below is historical and is not currently relied upon for plant activities.

Radiological dose calculations assumed the iodine activity associated with the released liquid was 20 micro-curies/cc. Iodine in the 30,000 pounds of water that flashed to steam was assumed to be transported with the steam.

The radiological consequences are based on the sum of the releases via normal inleakage paths (where the effective release height is assumed to be zero, and maximum building dilution effects of 1/3 with type F-1 m/sec meteorology are assumed) and releases via the SBGTS (where type B-1 m/sec meteorological conditions and 90% filter efficiency are assumed).

The maximum offsite exposure with complete air/steam mixing and with steam condensation is 0.02 rem to the thyroid at the site boundary. The maximum offsite exposure with no mixing of steam and air and no steam condensation is 6 rem to the thyroid. These exposures take into consideration the leakage via unfiltered leakage paths as well as leakage through the SBGTS and the main chimney.

[End of historical analysis]

Therefore, in the event of failure of the 1-inch instrument line, even assuming the very conservative assumptions outlined in the preceding paragraphs, the resulting dose would be a fraction of 10 CFR 100 limits. [15.6-5]

The specific activity of the primary coolant is limited by Technical Specification. In addition, there is no core uncovery and no perforations of the fuel during an instrument line break. Therefore, since only the coolant activity is released, the radiological dose calculations are independent of fuel type or design.

The reactor coolant released mass and flashed fraction for this analysis envelop those parameters for extended power uprate. Therefore, the calculated doses are not impacted by EPU.

15.6.3 Steam Generator Tube Failure

This section is not applicable to Quad Cities Station.

15.6.4 Steam System Line Break Outside Containment

15.6.4.1 Identification of Causes and Frequency Classification

The postulated accident is a sudden, complete severance of one main steam line outside containment with subsequent release of steam and water containing fission products to the pipe tunnel and the turbine building. This large flow of steam to the turbine building could fail the blowout panels and lead to the formation of a large steam cloud. [15.6-6]

This event is classified as a limiting fault, i.e. an event that is not expected to occur but is postulated because the consequences may result in the release of significant amounts of radioactive material. [15.6-7]

The steam system line break outside containment is not re-analyzed for reload cores.

15.6.4.2 Sequence of Events and Systems Operation

To evaluate the overall consequences of the postulated severance of one of the four main steam lines, the sequence of events following the break was investigated in detail. [15.6-8]

The initial conditions prior to the main steam line break were assumed to be:

Reactor Power	2511 MWt
Reactor Pressure	1020 psia
Reactor Water Level	Normal

The sequence of events assumed is:

Event	Time After Break (seconds)
Main steam line break	0
Main turbine control valves closure initiation	0.2
MSIV closure initiation	0.5
Reactor trip/control rod insertion started	1.5
Feedwater flow shut off	5.0
MSIVs closed	10.5

15.6.4.3 Core and System Performance

15.6.4.3.1 Main Steam Isolation Valve Closure

The steam blowdown flow rate through both ends of the postulated break would cause an increase in steam flow in each of the four lines to the maximum value allowed by critical flow considerations. Flow limiters (venturis) are sized in conjunction with isolation valve closure time so that core submergence is assured during blowdown and after termination of the accident. Therefore, venturi design would limit the maximum initial steam blowdown rate to 200% of rated steam flow. Rapid depressurization in the steam lines downstream of the flow limiters would initiate closure of the main turbine control valves within 0.2 second after the accident. The increased pressure differential across the flow limiters would indicate the severance immediately and would initiate MSIV closure (all 8 valves) within 0.5 second after the accident. Multiple flow limiter pressure differential sensors are provided in the primary containment isolation system to accomplish this function. [15.6-9]

15.6.4.3.2 Reactor Core Shutdown

A reactor scram would be initiated by a position switch on each MSIV at approximately 10% closure of the valve stem, as described in Section 7.2. Therefore, control rod insertion would begin within 1.5 seconds after the line break with a MSIV total elapsed closure time of 10.5 seconds (0.5 second detection plus 10 seconds closure). The MSIVs are designed to close against reactor operating pressure. In addition to the scram from MSIV closure, moderator voids generated in the core by depressurization caused by excess flow leaving the vessel would contribute sufficient negative reactivity to reduce reactor power immediately. Finally, as an additional backup, reactor low water level, which would occur later during the blowdown when the steam-water mixture density in the reactor vessel would be sufficiently low, would initiate a scram and isolate the reactor. [15.6-10]

15.6.4.3.3 Feedwater Flow

Since the design basis is the simultaneous loss of normal ac power, the analysis has been done assuming no feedwater flow. [15.6-11]

If normal ac power is available, the feedwater flow control valve would initially open fully due to the increased steam flow from the reactor pressure vessel. Within 1 second after the accident, the indicated high water level in the reactor vessel would initiate closure of the feedwater control valve. The feedwater flow would then decrease linearly to shut off at approximately 5 seconds after the accident. Following closure of the MSIVs, the reactor vessel water level would drop due to collapsing steam voids, thereby actuating the feedwater system to return the vessel water level to normal.

15.6.4.3.4 Reactor Coolant Blowdown

The two distinct intervals of blowdown are vapor blowdown before the steam-water mixture flows into the steam line, and steam-water mixture blowdown. The steam flow rate through the upstream side of the break would increase from the initial value of 680 lb/s in the line to 1360 lb/s (200% of initial) with critical flow occurring at the flow limiter in the steam line. The steam flow rate was calculated using an ideal nozzle model.[4] The flow model predicting the behavior of the flow limiter has been substantiated by tests conducted on a scale model over a variety of pressure, temperature, and moisture conditions. The steam flow rate through the downstream side of the break would consist of essentially equal flow components from the other three unbroken lines. In the three unbroken lines, critical flow would occur at their flow limiters since it was conservatively assumed that no friction losses exist. [15.6-12]

The steam flow rate in each of the three unbroken lines would increase from the initial value of 680 lb/s to 1360 lb/s (200% of initial). Total break flow is shown in Figure 15.6-1. The total steam flow rate leaving the vessel would be approximately 5500 lb/s which would be in excess of the generation rate of 2700 lb/s.

The simultaneous initial depressurization in the vessel would be at a rate of approximately 40 psi/s, as shown on Figure 15.6-2, which would cause flashing of the moderator throughout the reactor. Steam bubbles generated within the system would cause the reactor water level to rise at a rate determined by the difference between the rate at which bubbles are formed and the rate at which they break the water surface. Steam bubbles rise by buoyancy at an average velocity of 1 ft/s [5,6]relative to liquid, eventually separating from the mixture surface.

An analytical model was used to predict the rate of reactor water level rise. In a portion of the range of interest (i.e., steam blowdown) this model has been shown to be in reasonable agreement with level rise data obtained in a large vessel undergoing depressurization. When the reactor water level floods the steam dryers and reaches the vessel steam nozzles, the blowdown would change from single phase steam to a steam-water mixture.

A steam-water blowdown, would begin at 5 seconds after the break and would blow down at an average value of 12,000 lbm/s[7] as shown on Figure 15.6-1. At 8 seconds critical flow would be established because the MSIVs would be nearly closed.

15.6.4.3.5 Steam-Water Mixture Impact Forces

The maximum differential pressure which could be generated by continuous water flow past the MSIVs is 850 psi, that is, reactor vessel pressure when the valve is almost closed. This is below the differential pressure across the valve during hydrostatic testing. [15.6-13]

The surge pressure from the steam-water mixture in the steam line has been evaluated for the turbine-generator (T-G) design case, assuming that the MSIV closes in 10 seconds and assuming "instantaneous" deceleration of two-phase mixture at the valve. Line friction was ignored and a driving pressure of 850 psi was assumed. The reactor vessel pressure is less than 1000 psig due to vessel depressurization prior to mixed flow. By the time two-phase flow occurs in the steam line, the flow limiters would restrict flow. When flow is choked at the flow limiters, the pressure upstream of the valve (downstream from the limiter) is only 260 psig. Therefore, the maximum surge pressure is 260 psig when the steam-water mixture encounters the valve.

The resultant total transient differential pressure across the valve (520 psi) is well below the piping design pressure of 1250 psig. The line pressure downstream of the valve was conservatively assumed to be zero when calculating the differential pressure.

In addition to the preceding MSIV evaluation, a test program was conducted to demonstrate the ability of the MSIVs to withstand transient forces. This test program is described in Section 6.2.4.3.

15.6.4.3.6 Effect of Main Steam Isolation Valve Closure Time

A parametric analysis was performed to determine the effect of variable closing time of the MSIVs. The results are shown in Table 15.6-1. [15.6-14]

It would be necessary to lose approximately 120,000 pounds of water and steam before the top of the core would be exposed. As shown in Table 15.6-1, even if the MSIVs were closed in 10.5 seconds, the core would not be uncovered, even for the limiting condition of zero steam separation in the separators.

15.6.4.3.7 Core Cooling

The assumption of the simultaneous loss of normal ac power supply with the postulated break of one of the main steam lines would result in the coastdown of the recirculation drive pump flow as well as the feedwater flow. For the initial main steam line break analysis, core inlet flow was calculated using a computerized transient model which simulates the dynamics of the system including volume changes, heat addition, water level rise, and pump inertia. The initial analysis results of the core inlet flow transient are given on Figure 15.6-3. Approximately 5 seconds after the break, the core inlet plenum would begin to flash thereby causing an increase in the core inlet flow. [15.6-15]

For the main steam line break analysis performed for initial licensing a computerized transient model was used to evaluate the thermal hydraulic performance of the reactor core following this postulated break. The model includes as input: dimensions of the fuel assembly, thermal properties of the coolant and fuel, and time-varying pressure and flow, thereby allowing the evaluation of the condition of the fuel throughout the accident. The primary result of the initial thermal hydraulic performance analysis of the fuel bundle is the peak fuel clad temperatures. The evaluation of the accident was based on the reactor power being at 2511 MWt (corresponding to the steam flow rate for turbine design), pressure at approximately 1000 psig, and the combination of peaking factors resulting in a peak linear power density of 17.5 kW/ft for the 7x7 array such that the fuel is operating at its maximum warranted value. Furthermore, the fuel rod thermal performance was evaluated with a skewed cosine (to the top) axial power generation, (see Figure 15.6-4), causing an initial minimum critical heat flux ratio (MCHFR) close to the Technical Specification limit then in effect (circa 1970). The calculated MCHFR throughout the initial transient analysis is plotted on Figure 15.6-5 and shows the MCHFR of 1.6 approximately two seconds after the postulated accident occurs.

Subsequent to the preceding, an analysis was performed for 8x8 fuel arrays assuming a peak linear heat generation rate of 13.4 kW/ft for P/BP 8x8R fuel design and 14.4 for GE 8x8EB fuel design. The MCHFR and minimum critical power ratio (MCPR) resulting from a main steam line break outside containment for a typical BWR 4 are shown on Figure 15.6-6. GE has performed the main steam line break analysis for 9x9 and 10x10 fuel arrays as part of the LOCA analysis (see Section 6.3.3.2.2.3). This event is not re-analyzed for reload cores. [15.6-16]

15.6.4.4 Barrier Performance

The reduction in the vessel depressurization rate caused by the change from single-phase steam blowdown to steam-water mixture blowdown would result in an increase in MCPR and MCHFR primarily due to the reduction in flow caused by pump coastdown. The MCPR and the MCHFR through the initial part of the accident is shown on Figures 15.6-5 and 15.6-6. Following closure of the MSIVs the vessel would start to pressurize and the MCPR and MCHFR would continue to increase. The maximum MSIV closure time of 10.5 seconds would limit the total amount of liquid and steam lost from the primary system to prevent the core from being uncovered. Therefore, continuous cooling of the reactor core would be maintained throughout the transient and the subsequent initiation of the HPCI system, approximately 40 seconds after the accident occurs, would provide additional continuous core cooling. The peak (fuel rod) clad temperature (PCT) would never exceed its operating value and the fuel rod cladding would remain intact since no perforations of the fuel rod cladding would occur. [15.6-17]

15.6.4.5 Radiological Consequences

15.6.4.5.1 Application of Alternative Source Term Methodology

Regulation 10 CFR 50.67, "Accident Source Term," provides a regulatory mechanism for power reactor licensees to voluntarily replace the traditional accident source term used in design basis accident analyses (i.e., TID 14844, Reference 45) with an “Alternative Source Term” (AST). The methodology for this approach is provided in NRC Regulatory Guide 1.183 (Reference 46).

Accordingly, Quad Cities applied for the AST methodology for key Design Basis Accidents (DBAs). In support of a full-scope implementation of AST as described in Reference 46, AST radiological consequence are performed for the four DBAs that result in offsite exposure consequences: Loss of Coolant Accident (LOCA), Main Steam Line Break (MSLB), Fuel Handling Accident (FHA), and Control Rod Drop Accident (CRDA).

The NRC approved AST for Quad Cities in Reference 47.

Fission Product Inventory
No fuel damage is expected to result from a MSLB. Therefore, the activity available for release from the break is that present in the reactor coolant and steam lines prior to the event. Two cases are evaluated. Case 1 is for full power operation with a maximum equilibrium coolant concentration of 0.2 μCi/gm dose equivalent I-131. Case 2 is for a maximum coolant concentration of 4.0 μCi/gm dose equivalent I-131, based on a pre-accident iodine spike. This source term is consistent with the guidance contained in Reference 46. The MSIVs are assumed to isolate in 5.5 seconds. The radiological analysis assumes a conservative mass release of 1.4E5 pounds of liquid consistent with Standard Review Plan Section 15.6.4 (Reference 48).

Core Inventory Release Fractions
No fuel damage is expected to result from a MSLB. Therefore, the activity available for release from the break is that present in the reactor coolant and steam lines prior to the event.

Atmospheric Dispersion Factors
The Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) atmospheric dispersion factors (χ/Q) were determined using the guidance in NRC Regulatory Guide 1.5 (Reference 49), specifically:

$\chi/Q = 0.0133/\sigma_y\mu$, where

σ_y = horizontal standard deviation of the plume (meters)
μ = wind velocity (meters/second)

EAB χ/Q = 8.64E-04 sec/m^3
LPZ χ/Q = 8.69E-05 sec/m^3

Atmospheric dispersion factors were not calculated for the control room dose assessment. The plume was modeled as a hemispherical volume, the dimensions of which are determined based on the portion of the liquid reactor coolant release that flashed to steam. The activity of the plume is based on the total mass of water released from the break. This assumption is conservative because it considers the maximum release of fission products.

Release Paths
The entire radioactivity in the released coolant is assumed to be released to the atmosphere instantaneously as a ground level release. No credit is taken for holdup, plateout, or dilution within facility buildings. In addition, no credit is taken for control room isolation.

Dose Consequences
Dose consequences were determined using simplified models consistent with the requirements of Reference 46. Offsite dose consequences (i.e., EAB and LPZ) were determined based on the following factors: radioactivity releases (in Curies), dispersion factors, breathing rates, and dose conversions factors. The main control room doses are determined somewhat differently. Steam cloud concentrations are used rather than χ/Q multiplied by a curie release rate. No credit is taken for main control room filtration. Dose conversion factors were obtained from Federal Guidance Reports 11 and 12 (References 50 and 51).

15.6.4.5.2 Acceptance Criteria

The AST acceptance criteria for postulated major credible accident scenarios are provided by 10 CFR 50.67 and Regulatory Guide 1.183. For the main control room, adequate radiation protection is provided to permit access to and occupancy under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident. The AST acceptance criteria for an individual located at any point on the boundary of the exclusion area (the Exclusion Area Boundary or EAB) are provided by 10 CFR 50.67 as 25 rem TEDE for any 2-hour period following the onset of the postulated fission product release. The AST acceptance criteria for an individual located at any point on the outer boundary of the low population zone (LPZ) are provided by 10 CFR 50.67 as 25 rem TEDE during the entire period of passage of the radioactive cloud resulting from the postulated fission product release. Regulatory Guide 1.183 applies the following additional limits to events with a higher probability of occurrence including the MSLB. For the MSLB event (BWR), in the case of an accident assuming fuel damage (i.e., pre-incident iodine spike) doses at the EAB and LPZ should not exceed 25 rem TEDE. For MSLB accidents assuming normal equilibrium iodine activity, doses should not exceed 2.5 rem TEDE.

15.6.4.5.3 Computer Models

Dose consequences were determined using simplified models consistent with the requirements of Reference 46.

15.6.4.5.4 Key Plant Assumptions

Mass of liquid water released:	140,000 lb (conservative relative to expected release)
Flashing fraction:	40%
Mass of steam in cloud:	56,000 lb
CREV mitigation:	Not credited
Reactor coolant activity:	Case 1: 0.2 μCi/gm; Case 2: 4.0 μCi/gm

15.6.4.5.5 Results

The reanalysis of the MSLB accident event using AST methodology is documented in calculation QDC-0000-N-1266 (Reference 52). The results are summarized in Table 15.6-2a.

Section 15.6.4.5.6 describes the radiological assessment performed prior to the adoption of an updated accident source term in accordance with 10 CFR 50.67, "Accident Source Term."

15.6.4.5.6 Historical Information

The following describes main steam line break radiological dose analyses performed for the initial licensing of the plant, i.e., 7x7 fuel arrays.

The predominant activity in the discharged coolant would be N-16, which would be significantly reduced by decay due to its short half-life (about 7 seconds). If the reactor contained fuel with cladding leaks, the water released through the break would contain some fission products. [15.6-18]

During 1964 the Dresden Unit 1 reactor was operated with a significant number of cladding leaks. Analysis of reactor water samples indicated the following yearly average fission product content:

Fission Product	Activity (μCi/cc)
I-131	0.025
I-133	0.1
Other halogens	0.25
Other fission products	0.25

With a separate reactor cleanup system for each unit, it is estimated that the maximum coolant activity would be approximately 2.3 μCi/cc at 100,000 μCi/s release rate, and would have the following fission product contents:

Fission Product	Activity (μCi/cc)
I-131	0.067
I-132	0.38
I-133	0.40
I-134	0.53
I-135	0.49
Other halogens	0.14
Other fission products	0.28

Measurements of halogen concentrations in the Dresden Unit 1 reactor water and condensate showed that the steam to water halogen concentration ratio was in the range of 3×10^{-5} to 1×10^{-5}. The only halogens carried out through the break would therefore be those absorbed in the water. Thus, 116 curies, including 3.4 curies of I-131, 19.1 curies of I-132, 20.2 curies of I-133, 27.0 curies of I-134, and 24.9 curies of I-135 would be carried out through the break.

Based on operating experience, the above fission product concentrations in the reactor coolant would occur when the off-gas emission was at about 100,000 μCi/s measured after 30 minutes decay in the off-gas system. The noble gas activity discharged from the break, assuming a 10.5 seconds MSIV closure time, would be 5.4 Ci (calculated for 2 minutes decay time).

Superheated steam would discharge from the break and exhaust into the turbine building. At atmospheric pressure the total coolant discharged is assumed to separate to 55,000 pounds of steam and 45,000 pounds of water. The steam would travel upward through the turbine building, partially condensing on walls and equipment in the building. Pressure in the turbine building would be released through special sections of siding which have been installed in both the reactor building and turbine building. No damage to building structural members would occur. A realistic value for the water-steam mixture which would be lost during a main steam line break accident is 85,000 pounds. The value of 100,000 pounds stated previously is used to calculate radiological exposures as a result of the main steam line break accident, the 15,000 pounds additional representing additional margin of conservatism.

Depending upon the release rate from the turbine building, the steam cloud could rise to a height of 4600 feet or more in a 2-mph wind, or 920 feet in a 10-mph wind.[8] Due to the uncertainty in predicting the actual steam release rate the conservative assumption is made that the release occurs at the top of the turbine building with no additional rise. Experiments[9, 10] on halogen partitions indicate that 99 — 99.99% of the halogens would remain absorbed in the water which does not evaporate. Nevertheless, for this analysis, it was assumed that all of the halogens contained in the low quality steam-water mixture were released to the environs. The use of these conservative assumptions results in calculated radiological exposures which are at least four orders of magnitude greater than the actual case, however, even with such a conservative approach the radiological doses, as shown in Table 15.6-2 are well below the guidelines set forth in 10 CFR 100.

The break of a main steam line outside both the drywell and the reactor building was also evaluated by the AEC[27] during the initial licensing of the plant. For the analysis the MSIVs were assumed to start to close within 0.5 seconds after the steam line break was sensed and fully closed in a maximum time of 5 seconds, the limit established in the Technical Specifications. During the 5 second closure time, approximately 30,000 pounds of primary coolant would be lost through the break. The iodine concentration resulting from the total fission product inventory of the primary coolant system was assumed to be 20 μCi/ml. In calculating the consequences of the steam line break accident, the AEC followed Safety Guide 5 entitled "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors." Based on these assumptions the AEC determined that the potential offsite 2-hour dose at the site boundary would be 22 rem to the thyroid and less than 1 rem whole body. [15.6-19]

Subsequent to and separate from the preceding radiological dose analyses, NRC staff calculations reported showed the resultant radiological dose at the site boundary to be less than 30 rem to the thyroid. This dose was calculated on the basis of the radioiodine concentration limit of 5 μCi of I-131 dose equivalent per gram of water, atmospheric diffusion from an elevated release at 30 meters under fumigation conditions for Pasquill Type F, 1 meter per second wind speed, and a steam line isolation valve closure time of 5 seconds. [15.6-20]

The Technical Specification limitations on the specific activity of the primary coolant ensure that the two hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR 100. Therefore, the radiological dose consequences are not increased by fuel type or design changes.

The radiological consequences for a main steam line break outside of containment are not increased by the extended power uprate because the mass and energy releases remain the same and the activity released is the reactor coolant activity at Technical Specification concentrations which are not impacted by EPU.

[END OF HISTORICAL INFORMATION]

15.6.5 Loss-of-Coolant Accidents Resulting from Piping Breaks Inside Containment

A LOCA would result in the heating and pressurization of containment, a challenge of the emergency core cooling systems, (ECCS), and the potential release of radioactive material to the environs. The response of the containment to a LOCA is discussed in Section 6.2.1.3.2. The fuel thermal response and ECCS performance are described in Section 6.3.3, and additional evaluations referenced in section 4.2. The following section describes the potential radiological consequences due to a LOCA. Detailed containment and fuel thermal responses for LOCAs have been reanalyzed since original licensing, using updated models, codes, and assumptions. The radiological consequences addressed in the following LOCA discussion, however, are based on the initial 7x7 core, using original models and codes that are relatively conservative. The overall radiological consequences in the following discussion, therefore, are conservative with respect to fuel cladding perforations and source terms. The original UFSAR licensing basis prior to Extended Power Uprate utilized the TID 14844 methodology, which establishes source term based on rated core thermal power. The impact of Extended Power Uprate on the radiological consequences of a LOCA is addressed at the end of Section 15.6.5.5.

See the introduction to Section 15.6 for information regarding use of details from this analysis description which may not be applicable to the current fuel cycle.

15.6.5.1 Identification of Causes and Frequency Classification

The full range of LOCAs has been analyzed, from a small rupture where the makeup flow is greater than the coolant loss rate, to a highly improbable circumferential recirculation line break. The analysis shows that the circumferential recirculation line break would result in the maximum fuel temperature and containment pressure. [15.6-21]

This event is classified as a limiting fault, i.e. an event that is not expected to occur but is postulated because the consequences may result in the release of significant amounts of radioactive material. [15.6-22]

15.6.5.2 Sequence of Events and Systems Operation

The HPCI, residual heat removal (RHR), and core spray systems would act to cool the core following the accident. [15.6-23]

For breaks in small liquid lines up to about 0.12 ft^2 in area, HPCI can supply sufficient coolant to depressurize the vessel and cool the core, depending only on the core spray system and the LPCI mode of RHR for long-term cooling. For breaks in liquid lines between 0.12 ft^2 and 0.2 ft^2 in area, the depressurizing function of the HPCI and the coolant makeup function of either the core spray subsystem or the LPCI mode of RHR would act in conjunction to provide effective core cooling. In the event of a LOCA without HPCI capability (i.e., if the normal feedwater and HPCI are assumed to be unavailable), the ADS would cause the reactor vessel blowdown to occur in a time interval sufficiently short to permit operation of the core spray system and LPCI mode of RHR to assure adequate core cooling. [15.6-24]

For breaks in liquid line larger than about 0.2 ft^2 in area, where no depressurization assistance is required, such as the design basis recirculation line break described previously, the core spray subsystem in combination with the LPCI mode of the RHR system would be capable of cooling the core independently of the HPCI or ADS.

The coolant lost through the rupture would be condensed in the pressure suppression pool, thus reducing primary containment pressure. Energy would be removed from the pressure suppression pool by the containment cooling system. [15.6-25]

15.6.5.3 Core and System Performance

The methodology to analyze the consequences of a LOCA depends upon the particular break size and the location being evaluated. The analyses for LOCAs originally were performed using calculational models and techniques different from the models and techniques currently used. Since the original analysis, additional information and results of tests related to the performance of the ECCS systems also have become available. [15.6-26]

Section 6.3.3 and additional evaluations referenced in section 4.2 discuss the fuel thermal response, the ECCS performance, and the current analysis models.

Section 6.2.1.3.2 discusses the containment and coolant blowdown responses.

For 10 CFR 50 Appendix K analyses, the changing thermal and hydraulic phenomena that are associated with a design basis LOCA may be described in five phases: (a) temperature changes and heat removal during reactor blowdown with associated flow coastdown, (b) achievement of critical heat flow at any point on the fuel rod cladding and associated temperature rise of fuel and clad material, (c) lower plenum flashing causing a temporary resurgence of core flow, (d) temperature rise of fuel and cladding with diminished cooling and complete depressurization, and (e) temperature changes and heat removal during ECCS operation.

The first phase of the LOCA is the short-term blowdown period during which energy is removed from the core by coolant passing through the core and exiting through the postulated break, causing the reactor coolant system pressure to decrease rapidly. Initial conditions are nearly the same as during normal operation and nucleate boiling continues undisturbed.

A short time later, the core flow and system pressure decrease sufficiently that nucleate boiling cannot be sustained. The short-term blowdown phase ends when the coolant flow through the core reaches the inlet of the jet pump.

In the third phase of the LOCA, lower plenum flashing may occur. This is a flow phenomenon during the blowdown wherein a sudden transient increase in the core flow begins a few seconds after the core flow has decayed to near zero. The increase in core flow results when the liquid level in the vessel drops below the recirculation line suction nozzles causing the flow out the break to change from a liquid phase to a steam phase and increasing the rate of depressurization of the system.

This rapid depressurization results in a rapidly changing thermodynamic state of the fluid in the primary system. Because the fluid in the lower plenum beneath the core was initially in the subcooled state, it does not change thermodynamic state during early blowdown as does the rest of the fluid system; however, when the system pressure decreases to the level where this fluid flashes to steam a large increase in steam flow through the core results. This period of the LOCA is called "lower plenum flashing". Calculation of flows, temperatures and pressures during this phase depends on the knowledge of the flashing process, the effect of flow maldistribution, the resistance to flow of a two-phase mixture through the core and jet pump diffusers, and the rate of blowdown through the break.

Following the period of lower plenum flashing, heat generation, produced by the radioactive decay of the fission products, and thermal radiation among the fuel rod causes the core to heat up.

Although the loss of water level or the increase in drywell pressure resulting from a pipe break is sensed immediately and the ECCS is signaled to start, the actual injection of water by the low pressure systems does not occur for about 30 seconds. This time is required for the diesel generators to start and accept load, the reactor pressure to fall below the ECCS pump discharge pressure and the ECCS pumps to achieve full flow. Water is injected into the reactor through both the LPCI system and the core spray system.

15.6.5.4 Barrier Performance

The discussion in this subsection represents evaluations performed for initial licensing. Results of extensive LOCA experimental programs since 1974 have demonstrated the conservatism which the LOCA models have with respect to modeling the vessel inventory, inventory distribution, and core heat transfer. A new thermal-hydraulic model (SAFER) and a new fuel rod thermal-mechanical model (GESTR-LOCA) have been developed to provide more realistic calculations for LOCA analyses. Analysis of LOCA conditions using the SAFER-GESTR[12] codes results in lower values; hence, the following discussion of dose calculations are considered conservative. [15.6-27]

The following calculations are for 7x7 arrays. The original UFSAR licensing basis prior to Extended Power Uprate utilized the TID 14844 methodology, which establishes source term based on rated core thermal power. The radiological release is unchanged from 7x7 to 8x8 to 9x9 to 10x10 fuel assemblies. The impact of Extended Power Uprate on radiological consequences is discussed at the end of relevant sections.

[BEGIN HISTORICAL INFORMATION]

Section 15.6.5.4.1 presents historical information. The radiological consequences following a LOCA have been reassessed using a new accident source term in accordance with 10 CFR 50.67, "Accident Source Term." 10 CFR 50.67 provides a regulatory mechanism for power reactor licensees to voluntarily replace the traditional accident source term used in design basis accident analyses (i.e., TID 14844, Reference 45) with an Alternative Source Term (AST). Since the publication of TID 14844, significant advances have been made in understanding the timing, magnitude and chemical form of fission product releases from severe accidents. The methodology for this approach is provided in NRC Regulatory Guide 1.183 (Reference 46). The radiological assessment for AST is provided in Section 15.6.5.5.

15.6.5.4.1 Historical Information

Fission Product Release from the Fuel

Calculations performed for initial licensing show that about 8% of the fuel rods in the core might experience cladding perforation, based on a 1500°F perforation temperature; however, the conservative assumption is made that 15% of the fuel rods would experience cladding perforation. The thermal analyses also show that none of the fuel would reach melting conditions. A maximum of 1% of the noble gas activity and 0.5% of the halogen activity contained in a fuel rod is in the plenums and would be available for release if the cladding were to be perforated. Negligible solid or particulate activity would be released from the perforated rods. The amount of the total reactor fission product inventory released from the fuel would be about 0.15% of the noble gases and about 0.075% of the halogens. The release would occur as the cladding is perforated. [15.6-28]

Fission Product Release to the Drywell

The fallout and plateout of fission products within the reactor vessel and piping would reduce the amount of fission products available for transport to the drywell. Of the halogens that would be released from the fuel, 5% are assumed to be instantly converted to organic halides, principally methyl iodide.

Because organic halogens are both less soluble in water and more difficult to filter than uncombined halogens, a conservatively large fraction of halogens was assumed to be organic. Fuel melting experiments[13,14,15] have shown that 0.1% — 3% of the released halogens are organic. For the LOCA analysis, 5% of the halogens released from the fuel are assumed to be organic. This assumption is conservative by a factor of 1.5 to 50.

All organic halogens are assumed to be released and not fallout or plateout. Of the remaining 95% (which are inorganic), 50% will be subject to plateout on metal surfaces. The fallout and plateout in the reactor vessel and piping assumed in the analysis is:

Fallout and Plateout

Fission Product Group	Percent
Noble gases	0
Halogens, organic	0
Halogens, inorganic	50

The pressure suppression pool contains approximately 112,000 ft^3 of water for absorption of halogens. The containment air-to-water volume ratio is about 2.5. All the organic halogens are assumed to remain airborne; although at an air-to-water ratio of 2.5, about half would be expected to be absorbed in water.[16] In Oak Ridge Reactor (ORR) in-pile UO_2 melting experiments, the condensation of the steam in the gas stream removed essentially all halogens from the gas stream.[17] The inorganic halogen partition factor according to Allen would be greater than 10^4.[18] Watson[19] also reports the partition factor to be greater than 10. These experiments, including both steam condensation in vapor suppression systems and in air, correspond to the conditions accompanying a LOCA.

The initial blowdown through the suppression pool would be mostly air with the trailing phases of blowdown essentially being all steam. Most fission product release would accompany the final steam release and would be efficiently scrubbed by the condensing steam. Airborne inorganic halogen and solid fission products in the drywell would be rapidly removed by the containment spray and steam condensation then mixed with water in the suppression pool. For the accident analysis, a partition factor of 100 for inorganic halogens is used. Inorganic halogens are assumed to be re-evolved from the water as leakage from the containment reduces the inventory of airborne halogens. The assumption of a high fraction of organic halogens with no absorption in water and the conservative water to air partition factor for inorganic halogens results in a conservative high fraction of halogens remaining airborne which could leak from the containment. The inventory of airborne fission products in the drywell which could leak into the reactor building is shown in Table 15.6-3.

The following discussion further defines the assumptions and mechanisms concerning the time-dependent airborne fission-product inventory (Table 15.6-3).

The fission product airborne activity in the primary containment is dependent upon the fraction of contained activity released, plateout, fallout, washout, and conversion effects appropriate to a given species or group of isotopes. For the LOCA, a maximum of 8% of the fuel rods would be perforated; however, it is conservatively assumed that 15% of the fuel rods are perforated.

Those rods experiencing clad damage are assumed to release 1% of their noble gas and 0.5% of their iodine activity. Of the iodine activity released, 5% is assumed to be instantaneously converted to methyl iodide and the remaining 95% is assumed to experience a reduction factor of two due to plateout. That fraction of inorganic iodine which escapes plateout is carried to the suppression pool where it is assumed to instantaneously form and maintain an equilibrium condition with the air in the primary containment. Since the analysis shows the total number of fuel perforations would occur within approximately 5 minutes after initiation of the accident, the actual mechanism which transports activity from the drywell to the wetwell is unimportant as long as a transport mechanism exists. Less than 2% of the total assumed number of perforations occurs prior to approaching an equilibrium pressure condition between the wetwell and drywell. Therefore, less than 2% of the released activity would be driven to the suppression pool in the initial pressure transient.

The differential pressure between the reactor vessel and drywell would also be approximately zero after 30 seconds. Therefore, the only mechanisms available for fission product transport out of the reactor vessel would be by diffusion or confinement and transport in the liquid reactor coolant. The iodine activity released to the drywell after the initial pressure transient would therefore be released primarily with the liquid discharging through the break in the recirculation line.

The core spray would be actuated in approximately 60 seconds, liquid level in the reactor vessel would be at the top of the jet pumps after 125 seconds, and the liquid level would be at the bottom of the core after 180 seconds. Also, since the core is being sprayed from above by a two-phase mixture, fission product migration would be through the jet pumps to the drywell via the water.

Less than 4% of the assumed maximum number of damaged rods will experience cladding damage in the 125 seconds. It can therefore be concluded that the iodine activity would reach the suppression pool and that the mode of transport to the pool is not important.It is irrelevant whether or not the noble gas activity reaches the suppression pool as the decontamination effects of water for the noble gases is negligible. It is therefore assumed that all of the released noble gases are airborne in the primary containment.

Fission Product Release from Drywell to the Reactor Building

The primary containment leakage rates were calculated assuming that the primary containment leaks 0.5% of the contained free volume per 24 hours at 62 psig; the turbulent rough passage equation[20] was used for interpolation to lower pressures. The long-term primary containment pressure is shown in Section 6.2.1.3.2. The corresponding containment leakage for case d, shown in Figure 6.2-14 represents a highly faulted condition of one core spray, three RHR LPCI pumps, and one heat exchanger all inoperable. [15.6-29]

If fission products leak from the drywell, high pressure or reactor building high radiation signals would isolate the affected zone of the reactor building and start the standby gas treatment system. The SBGTS fan maintains the reactor building below atmospheric pressure and discharges a volume equivalent to 100% of the building volume per day through high-efficiency filters and charcoal absorbers to the 310-foot chimney. The analysis assumed all the noble gases and halogens released into the reactor building remain airborne. [15.6-30]

The airborne fission product inventory in the reactor building, which was evaluated considering the leakage from the drywell to the reactor building, radioactive decay, fallout and plateout, and an air change rate of 100% of the building volume per day, is shown in Table 15.6-4.

Fission Product Release from Reactor Building to Atmosphere

The halogens which leak from the pressure suppression containment into the reactor building are exhausted by the SBGTS through a demister, air heater, roughing filter, charcoal filter, and a high efficiency filter. The reactor building exhaust air is treated to reduce the humidity so that the filters will be effective for removal of organic halogens. Tests on filter efficiencies have shown that inorganic halogens are removed by charcoal filters with efficiencies greater than 99.99%.[21,22] Tests on filter efficiencies have also shown that organic halogens are removed by charcoal filters at a relative humidity less than 30% with filter efficiencies from 99.9 — 99.9999%.[22,23,24,25,26] The charcoal filter on the EVESR at Vallecitos Atomic Power Laboratory retained organic halogens produced at power operation with a filter efficiency from 99.8 — 99.9% at a relative humidity of 10 — 15%. The latest experimental results were used as the basis of the original SBGTS design. The system is designed to provide the necessary residence time in the filters. Thus, the analysis assumption of 99% filter efficiency for the removal of inorganic and organic halides in the SBGTS is conservative by approximately four orders of magnitude.

The compounding of conservative assumptions used in the LOCA analysis results in calculated doses from halogens that are 20 to 1000 times higher than would actually be expected. Discharge rates from the LOCA to the elevated release point are shown in Table 15.6-5.

[END OF HISTORICAL INFORMATION]

15.6.5.5 Radiological Consequences

15.6.5.5.1 Application of Alternative Source Term Methodology

Regulation 10 CFR 50.67, "Accident Source Term," provides a regulatory mechanism for power reactor licensees to voluntarily replace the traditional accident source term used in design basis accident analyses (i.e., TID 14844, Reference 45) with an "Alternative Source Term" (AST). The methodology for this approach is provided in NRC Regulatory Guide 1.183 (Reference 46).

Accordingly, Quad Cities applied for the AST methodology for key Design Basis Accidents (DBAs). In support of a full-scope implementation of AST as described in Reference 46, AST radiological consequence are performed for the four DBAs that result in offsite exposure consequences: Loss of Coolant Accident (LOCA), Main Steam Line Break (MSLB), Fuel Handling Accident (FHA), and Control Rod Drop Accident (CRDA).

The NRC approved AST for Quad Cities in Reference 47.

Fission Product Inventory

The inventory of reactor core fission products is based on maximum full power operation at a power level of 3016 MWth, the Extended Power Uprate (EPU) thermal power of 2957 MWth plus 2% to account for uncertainties in accordance with NRC Regulatory Guide 1.49 (Reference 53). The AST source term values for this analysis were derived using the ORIGEN2 computer code (Reference 54) and the guidance outlined in NRC Regulatory Guide 1.183. Core isotopic nuclides and their curie per megawatt activities were utilized from Appendix D of GE task report GE-NE-A22-00103-64-01 (Reference 55) for input into the RADTRAD computer code, which was used to determine dose consequences (note that the CO-58 and CO-60 activities were obtained from the RADTRAD User's Manual). The isotopic inventories (containment and reactor building) are provided in Tables 15.6-5a and 15.6-5b respectively. The SVEA-96 Optima2 fuel was also evaluated and determined to be bounded by the GE results.

Core Inventory Release Fractions

The core inventory release fractions were determined using the guidance in NRC Regulatory Guide 1.183, Section 3.2 (Release Fractions).

Atmospheric Dispersion Factors

The main control room and offsite dose χ/Q values were calculated using the methodology described in UFSAR Section 2.3.6. The atmospheric relative concentrations used for the LOCA analysis are as presented in Table 15.6-9.

Release Paths

Timing of Release Phases: The fission product inventory is assumed to be released in phases following a design basis LOCA. Regulatory Guide 1.183 specifies the onset and duration of each sequential release phase. The onset of fuel gap release occurs in 2 minutes and is assumed to last 0.5 hours. The early in-vessel phase begins in 0.5 hours and lasts for 1.5 hours. The inventory of each phase is released at a constant rate over the duration of the phase. Once dispersed in the primary containment, the release to the environment is assumed to occur through the following three pathways.

- Primary Containment
- Main Steam Isolation Valves (MSIVs)
- Emergency Core Cooling Systems

Primary Containment Leakage: The radioactive release from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of primary containment. A containment leakage of 3% volume per day is assumed, which is the sum of the primary-to-secondary leakage and leakage through the MSIVs. Leakage from primary containment is assumed to mix in 50% of the reactor building free volume which is treated by the standby gas filtration system. Reduction in containment leakage after 24 hours is not credited.

MSIV Leakage: The MSIVs are postulated to leak at a total design leak rate of 150 scfh (at 48 psig) as follows:

- 60 scfh through a steam line with a failed inboard MSIV; and
- the other three steam lines are assumed to leak at 60 scfh, 30 scfh, and 0 scfh respectively.

The radiological consequences from postulated MSIV leakage are combined with the consequences for the other fission product release paths.

Emergency Core Cooling System Leakage: Systems that circulate suppression pool water outside primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. The analysis conservatively assumes leakage begins at the onset of the accident and to continue throughout the 30-day duration. The leakage from all components is 1 gpm into the reactor building (the radiological assessment conservatively assumes 2 gpm).

<u>Source Term Mitigation</u>

Reduction in Airborne Activity Inside Containment: The activity of elemental iodine and aerosols released from the core into the drywell is reduced by deposition (i.e., plate-out) and settling in the drywell utilizing the natural deposition values identified in the RADTRAD code. Iodine removal by suppression pool scrubbing is not credited because the bulk core activity is released to containment well after the initial mass and energy release. Containment spray is also not credited. The Decontamination Factor (DF) for elemental iodine is consistent with SRP 6.5.2 and is limited to a DF 200.

Aerosol Deposition in Main Steam Lines: Main steam line pipe deposition was modeled using the RADTRAD code with removal coefficients based on gravitational settling. Two-node treatment is used for each steam line in which flow occurs. No credit is taken for holdup or plate-out in the main steam lines beyond the outboard MSIV. Only horizontal sections of piping are credited. Additionally, no credit is taken for holdup and plate-out in the main condenser. Main steam line deposition is based on using the shortest line (i.e., most rapid transport) for the worst case line (i.e., the one with the assumed failed inboard isolation valve).

Ventilation Cleanup Systems: Containment leakage into the reactor building is collected by the standby gas treatment (SGT) System which exhausts the reactor building, via filters, and reduces releases. The SGT exhaust charcoal and HEPA filter efficiencies are assumed to be 50% for elemental iodine, 50% for organic iodide, and 99% for particulate aerosols. The control room emergency ventilation (CREV) system mitigates dose to the main control room operators. The CREV charcoal and HEPA filter efficiencies are assumed to be 99% for elemental iodine, 99% for organic iodide, and 99% for particulate aerosols. The CREV is assumed to be actuated 40 minutes following a LOCA. Prior to initiation, unfiltered inleakage is assumed to be 60,000 cfm. Following actuation, unfiltered inleakage is assumed to be 400 cfm, which includes 10 cfm for normal ingress/egress.

Suppression Pool Post-LOCA pH Control: The guidance in NRC Regulatory Guide 1.183 provides that the iodine species released to containment include 95% cesium iodide, 4.85% elemental iodine, and 0.15% iodine in organic forms. This assumption is valid only if the suppression pool water is maintained at a pH of 7.0 or higher to ensure against the re-evolution of elemental iodine. The standby liquid control system is used to control pH by injecting sodium pentaborate. Following a DBA LOCA, sodium pentaborate is injected into the RPV where it will be mixed with ECCS flow and migrate to the suppression pool. Credit for standby liquid injection is based on operation of one pump, manually initiated and injected within 24 hours following a DBA LOCA (assumed to be injected within 2 hours of an accident where there is indication of fuel damage).

Dose Consequences

As per NRC Regulatory Guide 1.183, Total Effective Dose Equivalent (TEDE) offsite doses are determined as the sum of the committed effective dose equivalent (CEDE) from inhalation and deep dose equivalent (DDE) from external exposure from radionuclides. Dose conversion factors are taken from Federal Guidance Report No. 11 (Reference 50) and Federal Guidance Report No. 12 (Reference 51). Control room dose consequences are determined assuming a buildup of contamination in the control room. Included in this assessment is radiation shine from external sources (i.e., the radioactive plume and secondary containment) and sources inside the main control room (due to the buildup of radionuclides). No credit is taken for issuance of potassium iodide pills or respirators.

15.6.5.5.2 Acceptance Criteria

The AST acceptance criteria for postulated major credible accident scenarios are provided by 10 CFR 50.67 and Regulatory Guide 1.183. For the main control room, adequate radiation protection is provided to permit access to and occupancy under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident. The AST acceptance criteria for an individual located at any point on the boundary of the exclusion area (the Exclusion Area Boundary or EAB) are provided by 10 CFR 50.67 as 25 rem TEDE for any 2-hour period following the onset of the postulated fission product release. The AST acceptance criteria for an individual located at any point on the outer boundary of the low population zone (LPZ) are provided by 10 CFR 50.67 as 25 rem TEDE during the entire period of passage of the radioactive cloud resulting from the postulated fission product release.

15.6.5.5.3 Computer Models

AST calculations for the DBA LOCA were prepared to simulate the radionuclide release, transport, removal, and dose estimates associated with the postulated accident scenario.

Source term calculations were determined using the ORIGEN2 computer code. The ORIGEN2 computer code was designed for reactor fuel cycle mass and radioactivity inventory calculations. The ORIGEN2 computer code is widely recognized for calculating fission product inventories.

Offsite χ/Qs were calculated with the PAVAN computer code (Reference 56) using the guidance of NRC Regulatory Guide 1.145 (Reference 57); control room χ/Qs were calculated with the ARCON96 (Reference 58) and PAVAN computer codes consistent with the guidance of Regulatory Guide 1.194 (Reference 59). The PAVAN and ARCON96 codes are generally used to calculate relative concentrations in plumes from nuclear power plants at offsite locations and control room air intakes, respectively.

The RADTRAD computer code (Reference 60) was used for determining dose consequences for the DBA LOCA. The RADTRAD program is a radiological consequence analysis code used to estimate post-accident doses at plant offsite locations and in the control room.

15.6.5.5.4 Key Plant Assumptions

The key input assumptions are presented in Table 15.6-7.

15.6.5.5.5 Results

The reanalysis of the LOCA accident event using AST methodology is documented in calculation QDC-0000-N-1481 (Reference 61). The radiological consequences are presented in Table 15.6-8a.

Section 15.6.5.5.6 describes the radiological assessment performed prior to the adoption of an updated accident source term in accordance with 10 CFR 50.67, “Accident Source Term.”

15.6.5.5.6 Historical Information

The discussion in this subsection represents evaluations performed for initial licensing. Analysis of LOCA conditions using the SAFER-GESTR[12] codes results in lower values; hence, the following discussion of dose calculations are considered conservative. The original UFSAR licensing basis prior to Extended Power Uprate utilized the TID 14844 methodology, which establishes source term based on rated core thermal power. The impact of Extended Power Uprate and the use of SVEA-96 Optima2 fuel on the radiological consequences is discussed at the end of relevant sections. [15.6-31]

Offsite Dose Rates

The offsite radiological effects of a design basis LOCA for the initial core (7x7 arrays) based on the preceding barrier performance analysis and 99% SBGTS efficiency are shown in Table 15.6-6. These doses are far below the guideline radiation doses listed in 10 CFR 100. The original UFSAR licensing basis prior to Extended Power Uprate utilized the TID 14844 methodology, which establishes source term based on rated core thermal power. This table was valid before the Extended Power Uprate, regardless the change of fuel design from 7x7 to 8x8 to 9x9 to 10x10 fuel assemblies. It is retained for historical information. [15.6-32]

The consequences of the postulated design basis LOCA were calculated by the AEC[27] using the conservative assumptions presented in the Commission's Safety Guide 3 entitled "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors". For the AEC analysis, the primary containment was assumed to leak at a constant rate of 1.3% of the containment volume per day at accident conditions for the duration of the accident without consideration of the effects of post accident decreases in pressure. The leakage was assumed to pass directly to the SBGTS (assuming 90% halogen removal efficiency) without mixing with the reactor building atmosphere and then to the 310-foot chimney. The results of the AEC analysis indicate 2-hour doses at the site boundary of 6 rem for the whole body, and 150 rem to the thyroid. [15.6-33]

An evaluation of the design basis LOCA was based on the primary containment maximum allowable accident leak rate of 1.0%/day at 48 psig. The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90% for halogens and 95% for particulates, and assuming the fission product release fractions stated in TID 14844, the maximum total whole body passing cloud dose is about 5 rem, and the maximum total thyroid dose is about 120 rem at the site boundary over an exposure duration of 2 hours. The resultant doses that would occur for the duration of the accident at the low population distance of 3 miles are lower than those stated due to the variability of meteorological conditions that would be expected to occur over a 30-day period. Thus, the doses reported are the maximum that would be expected in the unlikely event of a design basis LOCA. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected offsite doses and 10 CFR 100 guidelines. [15.6-34]

The Extended Power Uprate increases the 2 hour thyroid dose from 120 rem to 152 rem and the whole body dose from 5 rem to 6 rem. An evaluation of the post-LOCA offsite doses for SVEA-96 Optima2 fuel shows that the existing EPU doses are either bounded, maintained, or increased slightly [44]. The SVEA-96 Optima2 doses are shown in Table 15.6-8a.

Thus, there is adequate margin in the design of the reactor and containment to limit the consequences of large postulated accidents and still adequately protect the public. [15.6-35]

Site Dose Rates

The radiological effects in the reactor building a week after a design basis LOCA would be greater than 500 R/hr, restricting access to short durations for life saving purposes only. The turbine building would be accessible on a limited basis depending on location.[28] [15.6-36]

Control Room Dose Rates

Subsequent to the offsite and site dose analyses, a control room dose analysis was performed in accordance with the guidance of NUREG 0737[29] Item III.D.3.4 to determine compliance with the radiological requirements of General Design Criterion (GDC) 19 and Standard Review Plan (SRP) 6.4.[30] The LOCA was considered in the analysis to be the radiological design basis accident (DBA). The results of this analysis are considered conservative. Several natural mechanisms will reduce or delay the radioactivity prior to release to the environment. Credit was taken only for iodine plateout on surfaces of the steam lines and condenser and radioactive decay prior to release. [15.6-37]

Methodology

The guidelines given in SRP 6.4[30] and Regulatory Guide 1.3[31] have been used with the exceptions of the chi/Q for the control room and plateout of iodines during transportation within pipes. Realistically, the components of the main steam lines and the turbine-condenser complex would remain intact following a design basis LOCA. Therefore, plateout of iodines on surfaces of the main steam lines and the turbine-condenser complex would be expected.

Figure 15.6-7 shows the radiological control room model used for activity released through the SBGTS and through the MSIVs. The total control room 30-day integrated dose would be equal to the sum of the two dose models. The input parameters used to develop the activity levels in the control room are shown on Table 15.6-7. [15.6-38]

Assumptions and Bases

Regulatory Guide 1.3 [31] has been used to determine activity levels in the containment following a design basis LOCA. Activity releases are based on a containment leakage rate of 1% per day. Table 15.6-7 lists the assumptions and parameters used in the analysis and dose point locations. The majority of the containment leakage will be collected in the reactor building and exhausted to the atmosphere through the SBGTS as an elevated release from the main stack. Any SBGTS bypass leakage has been quantified by assuming that all MSIVs leak 11.5 scfh per main steam line when tested at 25 psig (Technical Specification limit for all MSIV leakage paths is less than or equal to 46 scfh). The leak-rate was corrected to the containment design pressure using the laminar flow extrapolation factors of ORNL NSIC-5[41]. [15.6-39]

Leakage past the isolation valves could be released through the outboard MSIV stems into the steam tunnel, or continue down the steam lines to the stop valves and into the turbine-condenser complex. The steam tunnel is exhausted by the SBGTS filtration system, thus eliminating it as a bypass pathway. The MSIV leakage travels down the steam piping to the turbine-condenser complex where it is released as a ground level release at a rate of 1% of the turbine-condenser volume per day. This leak rate is consistent with the assumptions used for the control rod drop accident in SRP 15.4.9.[32] This assumption is conservative since the volumetric leakage from the condenser at 1% per day is greater than the MSIV leakage from the drywell. The MSIV leakage passes through three different volumes which provide holdup and plateout. The first volume consists of the steam lines between the inboard and outboard isolation valves, the second volume consists of the steam lines between the outboard isolation valves and the turbine stop valves, and the third volume includes the steam lines after the turbine stop valves and the turbine-condenser complex. The leakage path was conservatively treated as a single volume with a volume of 1.7 x 10^5 ft^3 and a surface area of 6.5 x 10^5 ft^2. The iodine removal rates were calculated for elemental and particulate iodines using a deposition velocity of 0.012 cm/s. The removal of organic iodine through plateout is not credited. Elemental and particulate iodine decontamination factors of over 100 can be calculated for the small travel distances and large travel times down the steam lines, refer to NUREG/CR-009 Section 5.1.2[33].

The MSIVs will leak to the turbine building which would be exhausted by the heating, ventilating, and air conditioning (HVAC) system if it was operational. Additional plateout on ductwork, fans, and unit coolers would further minimize the iodine releases. If the HVAC system was not operational, then any bypass flow would tend to collect in the building and be subject to additional decay and plateout which are not credited in the analysis.

The activity which enters the main control room may be the result of bypass leakage, the SBGTS exhaust in the outside air, or both, depending on wind direction. It is possible for the control room intake to be exposed to activity from both sources at the same time. Because the SBGTS exhaust is elevated, the concentrations from this source at the intake will be less than those due to bypass leakage. It is conservatively assumed that the activity concentration at the intake is due to concurrent bypass leakage and chimney releases for the duration of the event.

Infiltration Analysis

During emergency operation, the control room ventilation system supplies 1800-2200 scfm of outdoor air to maintain the control room at 1/8 in w.g. positive overpressure with respect to the adjacent areas. Intentionally admitting outdoor air into the habitable zone prevents infiltration through the habitable zone boundary by assuring that air is exfiltrating from the zone at a fairly significant velocity (a velocity through the habitable zone boundary penetrations of approximately 1400 ft/min is required to develop a backpressure of 1/8 in w.g.). Prior to start of the AFU (during the first 1 hour 50 minutes of the accident), 1000 cfm is assumed to infiltrate through the zone boundary based on SRP 6.4[30] which recommends a base infiltration rate equal to 50% of the makeup rate required to pressurize the zone to 1/8 in w.g. [15.6-40]

The infiltration of unfiltered air into the control room habitable area is assumed to enter through three different pathways subsequent to emergency isolation: [15.6-41]

1. Through the habitable zone boundary;

2. Through the system components located outside of the habitable zone; and

3. Through backflow which could occur when the control room doors are opened.

In accordance with SRP 6.4[30], the infiltration through the zone boundary is assumed to be zero when the system is pressurized.

Infiltration through the components located outside the habitable zone occurs through joints and seams in the ductwork, damper shafts, joints and penetrations in the air handling units and the dampers that isolate the habitable zone from the nonhabitable areas. Figure 6.4-1 identifies the system components which are located outside of the habitable zone that are under a negative pressure. The inleakage through the damper shafts and blades was based on damper specification requirements and vendor test data. The Train A ductwork and air handling unit was assumed to leak at the maximum adjusted leakage rate provided in ANSI N-509 of, 0.2 cfm/ft^2 at 10 in $H_{\{2\}}O$. The Train A ductwork leak rate was used in the analysis because it was more conservative than the Train B ductwork leak rate of 0.005 cfm/ft^2 at 10 in H_2O gauge.

The opening and closing of boundary doors can induce infiltration to the habitable zone. Standard Review Plan 6.4[30] recommends that backflow through the control room doors be assumed to be 10 cfm.

The infiltration analysis resulted in a total unfiltered infiltration rate of 260 cfm. A breakdown of the infiltration through the different leakage paths is as follows:

Leakage Path	Infiltration (cfm)
Through ductwork	150.1
Through equipment housings	32
Through isolation dampers	45.9
Through damper shafts	21.3
Through zone doors (backflow)	10
Through habitable zone boundary	0
Total infiltration	259.3

Atmospheric Dispersion Factor (chi/Q)

The following discussion is an explanation of the reasons for the use of the Halitsky chi/Q methodology and a value of $K_{\{C\}} = 2$, instead of the Murphy methodology[34] which SRP 6.4[30] suggests as an interim position. [15.6-42]

Historically, the preliminary work on building wake chi/Qs was based on a series of wind tunnel tests by J. Halitsky, et al.[35]. In 1974, K. Murphy and K. Campe of the NRC published their paper based on a survey of existing data. This chi/Q methodology, which presented equations without derivation or justification, was adopted as the interim methodology in SRP 6.4 in 1975. Since then, a series of actual building wake chi/Q measurements have been conducted at Rancho Seco[36] and several other papers have been published documenting the results of additional wind tunnel tests.

Murphy[34] suggested the following equation for the calculation of chi/Q:

$$chi/Q = K_{\{C\}}/AU$$

where

$K_{\{C\}} = K + 2$

$K = 3/(s/d)^{1.4}$

A = Cross-sectional area of the building

U = Wind speed

This formulation was derived from the Halitsky[35] data in Figure 37 from Murphy's paper.[34] The Halitsky data were from wind tunnel tests on a model of the EBR-II rounded (PWR type) containment and the validity of the data was limited to 0.5 < s/d <3. The origin and reason for the +2 in K + 2 is not known. All other formulations use K only, and for the situation where K is less than 1, the use of K + 2 imposes an unrealistic limit on the chi/Q.

For Quad Cities, the building complex is composed of square-edged buildings and not a round-topped cylindrical containment as was used in the Halitsky experiments. For an HVAC intake located near the south wall of the control room at elevation 633 feet 0 inches, the intake will be subject to a building wake caused by a combination of the reactor building and the turbine building for any bypass leakage escaping from the turbine building. There will not be any reactor building bypass leakage because the building is kept at a negative pressure by the SBGTS which exhausts to the main chimney.

Because the Murphy methodology could not be applied, a survey of the literature was undertaken. It was found that the Halitsky wind tunnel test data[35] conservatively overestimated $K_{\{C\}}$ values "by factors of up to possibly 10." Given this conservatism, it was felt that the use of a reasonable $K_{\{C\}}$ value from the Halitsky data on square-edged buildings should be acceptable. A review of Figure 5.27 from the Halitsky data [35] resulted in $K_{\{C\}}$ values in the 0.5 to 2 range. A value of $K_{\{C\}} = 2$ was chosen to determine a chi/Q for the control room. A building cross-sectional area of 1,550 m^2^ was conservatively used. This corresponds to a projected area of one reactor building above grade. The use of a 1,550 m^2 area is very conservative because both of the reactor buildings are adjacent to each other and the combined projected area could be larger than the value used. Information from other sources, as indicated in the following, has also shown that this should be a conservative value.

In a paper by D. H. Walker,[37] control room chi/Qs were experimentally determined for floating power plants in wind tunnel tests. Different intake and exhaust combinations were considered. Using the data for intake 6 and stack A exhaust,[37]chi/Q values of 1.77×10^{-5} and 2.24×10^{-5} were found after adjusting the wind speed from the 1.5 m/s to 1 m/s. These values are approximately two orders of magnitude lower then the conservatively calculated value for Quad Cities.

In a wind tunnel test by P. N. Hatcher,[38] a model industrial complex was used to test dispersions due to a wake. Data obtained from these tests show that K_C has a value less than 1, and decreases as the test points are moved closer to the structure. In a study to determine optimum stack heights, R. N. Meroney and B. T. Yang[39] show that for short stacks (6/5 of building height), K_C reaches a value of approximately 0.2 and decreases closer to the building. They concluded that the Halitsky methodology was "overly conservative." These recent experimental tests show that K_C = 2 used to determine the chi/Q for Quad Cities is a conservative estimate by at least a factor of 2 and possibly by a factor of 10 or more.

Field tests were made on the Rancho Seco facility,[36] and chi/Q values were obtained. The data indicates that the use of K_C = 2 is conservative.

It was concluded that sufficient data and field tests exist to give a reasonable assurance that the chosen chi/Q is a conservative one, over and above the conservatism implied by using the fifth percentile wind speed and wind direction factors. Based on the preceding discussion, the following equation is used in the calculation of chi/Q values.

chi/Q = 2/AU

Mechanisms for Reducing Iodine Releases

The following four mechanisms could result in significant quantities of iodine being removed before they would be released to the environment. However, credit for the plateout mechanisms is the only credit taken in the calculation of radiological consequences.

1. Drywell sprays, suppression pool-to-air partitioning and condensation effects — The drywell sprays would reduce the iodine source term if actuated. Even without the spray system, condensation would occur in the drywell and suppression chamber. The iodines in the air and suppression pool would be expected to reach equilibrium due to this phenomenon. Because the iodines have a preference to remain in solution due to the equilibrium partition factor of over 400 established by the physical conditions in the containment, the iodines available for release by air leakage would be reduced significantly. The NSAC-14, Workshop on Iodine Releases in Reactor Accidents has indicated that the iodine release assumption may be overly conservative. Most of the iodine may be released as cesium iodide instead of elemental iodine. The cesium iodide has a much higher solubility and tendency to plateout than elemental iodine.

2. Plateout — The plateout removal constant used in this analysis is based on the lowest deposition velocity quoted in NUREG/CR-009.[33] The NOAA's Technical/Memorandum, "Rancho-Seco Building Wake Effects on Atmospheric Diffusion,"[36] indicates that the deposition velocities could be higher by a factor of 4, which would tend to increase the plateout.

3. Removal through valves and leakage holes — Because the bypass leakage paths would be through minute holes in valves and valve seats, the leakage would be subjected to filtration effects. Larger particulates could tend to plug the leak paths.[40]

4. Condensate within pipes — Condensation would occur within the pipes when the pipes cool down to ambient temperature. This could result in removal of iodines and particulates from the gas phase.

Results

The 30-day control room doses using the inputs in Table 15.6-7 with the control filter unit initiated at 1 hour 50 minutes are shown in Table 15.6-8.[43] [15.6-43]

The extended power uprate increases the calculated control room thyroid dose by 30%, and the whole body and beta doses by 20%. The post-EPU doses are shown in Table 15.6-8a and are within the regulatory limits.

An evaluation of the post-LOCA control room doses for Optima2 fuel shows that the existing EPU doses are either bounded, maintained, or increased slightly [44]. The Optima2 doses are shown in Table 15.6-8a.

[END OF HISTORICAL INFORMATION]

15.6.6 References

1. Letter from W.A. Paulsen (NRC) to L. Del George (CECo), January 4, 1982, Subject: Dresden 2 SEP Topic XV-15, Inadvertent Opening of a BWR Safety/Relief Valve.

2. Letter from D. Crutchfield (NRC) to W. Counsil (Northeast Nuclear Energy Co.), October 28, 1981, Subject: SEP Topic XV-15 for Millstone 1 (Docket 50-245).

3. Letter from D. Crutchfield (NRC) to I. Finfrock (Jersey Central Power & Light Co.), December 4, 1981, Subject: SEP Topic XV-15 for Oyster Creek (Docket 58-219).

4. Moody, F. J., "Maximum Flowrate of a Single Component. Two Phase Mixture." Journal of Heat Transfer, Trans. ASME Series C, Vol. 87, p 134.

5. Wilson, J.F., et al., "The Velocity of Rising Steam in a Bubbling Two Phase Mixture." ANS Transactions, Vol. 5, No. 1, p 151, 1962.

6. Moody, F.J. "Liquid-Vapor Action in a Vessel During Blowdown,"APED-5177, June 1966.

7. Moody, F.J., "Maximum Two Phase Vessel Blowdown from Pipes, ASME Paper No. 65-WA/HT-1.

8. Singer, I. A., Frijjolo, J. A. and Smith, M. E. "The Prediction of the Rise of a Hot Cloud from Field Experiments," Journal of the Air Pollution Control Association, November 1964.

9. Watson, L.C., et. al., "Iodine Containment by Dousing in NPD-II," AECL 1130, October 1960.

10. Diffey, H.R., et. al., "Iodine Cleanup in a Steam Suppression System," International Symposium on Fission Product Release and Transport Under Accident Conditions, April 1956.

11. Deleted.

12. Deleted

13. Collins, D.A., et al., International Symposium on Fission Product Release and Transport Under Accident Conditions, Oak Ridge Tennessee Paper 59, April 1965.

14. Parker, et al., Volume II, "Fission Product Release," SIFTOR Draft, Chapter 18.

15. Collins, R.D., and Hillary, International Symposium on Fission Product Release and Transport Under Accident Conditions, Oak Ridge, Tennessee Paper 44, April 1965.

16. Diffey, et al., International Symposium on Fission Product Release and Transport Under Accident Conditions, Oak Ridge, Tennessee Paper 41, April 1965.

17. Miller, et al., International Symposium on Fission Product Release and Transport Under Accident Conditions, Oak Ridge, Tennessee Paper 12, April 1965.

18. Allen, T.L., and Keefer, R. M., "The Formation of Hypoidus Acid and Hydrated Iodine Cation by the Hydrolysis of Iodine," JACS 77, No. 11, June 1955.

19. Watson, Bancroft, and Hoelke, "Iodine Containment by Dousing in NPD-II," AECL-1130, 1960.

20. Maccary, R.L., et al., "Leakage Characteristics of Steel Containment Vessels and the Analysis of Leakage Rate Determinations." TID-20483, May 1964.

21. Keiholts, G.W., and Barton, C.J., "Behavior of Iodine in Reactor Containment Systems," ORNL-NSIC-4, p 64, February 1965.

22. Adams and Browning, International Symposium on Fission Product Release and Transport Under Accident Conditions, Oak Ridge, Tennessee, Paper 46, April 1965.

23. Collins, R.D., and Hillary, International Symposium on Fission Product Release and Transport Under Accident Conditions, Oak Ridge, Tennessee, Paper 44, April 1965.

24. Collins, and Eggleton, "Behavior of Iodine in Reactor Containment Systems," ORNL-NSIC-4, p 65 February 1965.

25. Adams and Browning, "Behavior of Iodine in Reactor Containment Systems," ORNL-NSIC-4, p 65 February 1965.

26. Collins, D.A., et al., International Symposium on Fission Product Release and Transport Under Accident Conditions, Oak Ridge, Tennessee Paper 45, April 1965.

27. U.S. Atomic Energy Commission, Division of Reactor Licensing, Safety Evaluation for Quad Cities Station, Units 1 and 2, Docket Nos. 50-254 and 50-265, August 25, 1971.

28. Brtis, and Lanti, "Post-Accident Radiation Levels," Sargent & Lundy Engineers, Chicago, IL Quad Cities Station Project Number 5954-00 December 1979.

29. U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG 0737, November 1980.

30. U. S. Nuclear Regulatory Commission, "Standard Review Plan 6.4 for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 6.4, "Habitability Systems", NUREG 0800 Rev. 1, December 1978.

31. U. S. Nuclear Regulatory Commission Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," Rev. 2, June 1974.

32. U. S. Nuclear Regulatory Commission Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Section 15.4.9 "Spectrum of Rod Drop Accidents" (BWR), NUREG-0800 Rev. 1, December 1978.

33. "Technological Basis for Models of Spray Washout of Airborne Contaminants in Containment Vessels," NUREG/CR-009, Posta, A.K., Sherry, R.R., Tam, P.S., October 1978.

34. Murphy K.G. and Campe, K.M. Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19, 13th AEC Air Cleaning Conference.

35. Slade, D. H. ed., Meteorology and Atomic Energy, TID, 24190 (1968).

36. Start, G.E., Cate, J.H., Dickenson, C.R., Ricks, N.R., Ackerman, G.H., and Sagendorf, J.F., "Rancho-Seco Building Wake Effects on Atmospheric Diffusion," NOAA Technical Memorandum, ERL ARL-69, 1977.

37. Walker, D.H., Nassano, R.N., Capo, M.A., 1976, Control Room Ventilation Intake Selection for the Floating Nuclear Power Plant 14th ERDA Air Cleaning Conference.

38. Hatcher, R.N., Meroney, R.N., Peterka, J.A., and Kothari, K. "Dispersion in the Wake of a Model Industrial Complex," NUREG 0373 1978.

39. Meroney, R.N. and Yang, B.T. "Wind Tunnel Study on Gaseous Mixing Due to Various Stack Heights and Injection Rates Above an Isolated Structure," FDDL Report CER 71-72RNM-BTY16, Colorado State University 1971.

40. Morewitz, H.A., Johnson, R.P., Nelson, C.T., Vaughn, E.V., Guderjahn, C.A., Hillard, R.K., McCormack, J.D., and Posta, A.K., "Attenuation of Airborne Debris from Liquid-Metal Fast Breeder Reactor Accident," Nuclear Technology, Volume 46, December 1979.

41. ORNL NSIC-5, "U.S. Containment Technologies", Oak Ridge National Laboratory and Bechtel Corp., August 1965.

42. "LOCA Break Spectrum Analysis for Quad Cities Units 1 and 2," EMF-96-184(P), December 1996.

43. QDC-9400-M-0348, Revision 1, Assessment of Control Room Habitability with Increased SGTS Filter Efficiency, April 28, 1997.

44. Calculation QDC-000-N-1020, "Impact of Extended Power Uprate on Site Boundary and Control Room Doses from LOCA and Non-LOCA Events," Revision 001A, Attachment E, "Evaluation for Impact of Westinghouse Optima2 Fuel on Existing UFSAR Design Basis Accidents."

45. U. S. Atomic Energy Commission, Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," 1962.

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

47. Letter from M. Banerjee (U. S. NRC) to C. Crane (Exelon Corporation), Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2 - Issuance of Amendments Re: Adoption of Alternative Source Term Methodology," dated September 11, 2006 [SER correction letter: D. Collins (U. S. NRC) to C. Crane (Exelon Corporation), "Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2 – Correction of Safety Evaluation for Amendment Dated September 11, 2006," dated September 28, 2006].

48. NUREG-0800 (Standard Review Plan), Section 15.6.4, "Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)."

49. NRC Regulatory Guide (Safety Guide) 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors," March 1971.

50. U.S. Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988.

51. U.S. Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," 1993.

52. QDC-0000-N-1266, "Re-analysis Main Steam Line Break (MSLB) Accident Using Alternative Source Term."

53. NRC Regulatory Guide 1.49, "Power Levels of Nuclear Power Plants," December 1973.

54. ORNL/TM-7175, "A Users' Manual for the ORIGEN2 Computer Code," A. G. Croff, July 1980.

55. GE Task Report No. GE-NE-A22-00103-64-01, Rev. 0, Project Task Report T0802, "Dresden and Quad Cities Asset Enhancement Program - Radiation Sources and Fission Products," August 2000.

56. Atmospheric Dispersion Code System for Evaluating Accidental Radioactivity Releases from Nuclear Power Stations; PAVAN, Version 2; Oak Ridge National Laboratory; U.S. Nuclear Regulatory Commission; December 1997.

57. NRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," February 1983.

58. Atmospheric Relative Concentrations in Building Wakes; NUREG/CR-6331, PNNL-10521, Rev. 1; prepared by J. V. Ramsdell, Jr., C. A. Simmons, Pacific Northwest National Laboratory; prepared for U.S. Nuclear Regulatory Commission; May 1997 (Errata, July 1997).

59. NRC Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003.

60. NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," April 1998 and Supplement 1, June 1999.

61. QDC-0000-N-1481, "Quad Cities Units 1 & 2 Post-LOCA EAB, LPZ, and CR Dose – AST Analysis."

Table 15.6-1

EFFECT OF MAIN STEAM ISOLATION VALVE CLOSURE TIME

(Historical Information)

Steam Line Isolation Valve Closure Time (Seconds)	Net Mass of Water and Steam Loss from Pressure Vessel (Pounds)	
(Includes 0.5 detection time)	With Feedwater	Without Feedwater
3.5 seconds	10,000	20,000
10.5 seconds	60,000	85,000

Table 15.6-2*

RADIOLOGICAL EFFECTS OF THE STEAM LINE BREAK ACCIDENT (INITIAL CORE ANALYSIS)
(Original analysis, retained for historical purpose)

Distance (Miles)	First 2 — Hour Dose						Total Dose					
	VS-1	MS-1	N-1	N-5	U-1	U-5	VS-1	MS-1	N-1	N-5	U-1	U-5
	Passing Cloud Whole Body Dose (rem)											
1/4	1.1×10^{-3}	1.1×10^{-3}	1.1×10^{-3}	2.2×10^{-4}	5.8×10^{-4}	1.4×10^{-4}						
1	7.3×10^{-4}	6.8×10^{-4}	4.6×10^{-4}	8.8×10^{-5}	1.1×10^{-4}	2.9×10^{-5}						
3	3.9×10^{-4}	3.1×10^{-4}	1.2×10^{-4}	2.4×10^{-5}	1.6×10^{-5}	5.0×10^{-6}			Not			
5	2.4×10^{-4}	1.8×10^{-4}	5.1×10^{-5}	1.1×10^{-5}	5.7×10^{-6}	2.0×10^{-6}			Applicable			
10	1.1×10^{-4}	6.7×10^{-5}	1.2×10^{-5}	3×10^{-6}	1.2×10^{-6}	5.3×10^{-7}						
	Lifetime Thyroid Dose (rem)											
1/4	2.3×10^{-4}	4.0×10^{-1}	5.2×10^{-1}	8.8×10^{-2}	2.4×10^{-1}	5.9×10^{-2}						
1	3.8×10^{-3}	2.6×10^{-1}	1.2×10^{-1}	3.6×10^{-2}	2.3×10^{-2}	6.0×10^{-3}						
3	2.5×10^{-2}	9.6×10^{-2}	1.9×10^{-2}	5.7×10^{-3}	3.1×10^{-3}	8.1×10^{-4}			Not			
5	3.6×10^{-2}	5.1×10^{-2}	7.5×10^{-3}	2.3×10^{-3}	1.2×10^{-3}	3.1×10^{-4}			Applicable			
10	3.3×10^{-2}	1.9×10^{-2}	2.2×10^{-3}	6.5×10^{-4}	3.3×10^{-4}	8.7×10^{-5}						
	Fallout Whole Body Dose (rem)											
1/4	1.7×10^{-7}	5.2×10^{-4}	1.1×10^{-3}	9.0×10^{-4}	1.0×10^{-3}	1.3×10^{-3}	5.7×10^{-7}	1.7×10^{-3}	3.6×10^{-3}	3.0×10^{-3}	3.3×10^{-3}	4.1×10^{-3}
1	2.8×10^{-6}	3.4×10^{-4}	2.5×10^{-4}	3.7×10^{-4}	9.7×10^{-5}	1.3×10^{-4}	9.1×10^{-6}	1.1×10^{-3}	8.4×10^{-4}	1.2×10^{-3}	3.2×10^{-4}	4.2×10^{-4}
3	1.9×10^{-5}	1.3×10^{-4}	3.9×10^{-5}	5.9×10^{-5}	1.3×10^{-5}	1.7×10^{-5}	6.1×10^{-5}	4.1×10^{-4}	1.3×10^{-4}	2.0×10^{-4}	4.3×10^{-5}	5.7×10^{-5}

Table 15.6-2

RADIOLOGICAL EFFECTS OF THE STEAM LINE BREAK ACCIDENT (INITIAL CORE ANALYSIS) (Cont'd)

	Fallout Whole Body Dose (rem)											
5	2.6×10^{-5}	6.6×10^{-5}	1.5×10^{-5}	2.4×10^{-5}	5.1×10^{-6}	6.7×10^{-6}	8.7×10^{-5}	2.2×10^{-4}	5.1×10^{-4}	7.8×10^{-5}	1.7×10^{-5}	2.2×10^{-5}
10	2.4×10^{-5}	2.6×10^{-5}	4.4×10^{-6}	6.7×10^{-6}	1.4×10^{-6}	1.8×10^{-6}	7.9×10^{-5}	8.3×10^{-5}	1.5×10^{-5}	2.2×10^{-5}	4.7×10^{-6}	6.1×10^{-6}

* The Technical Specification limitations on the specific activity of the primary coolant ensure that the two hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR 100. Therefore, the radiological dose consequences are not increased by fuel type or design changes.

	Meteorology	
VS-2	Very stable	2 mph
MS-2	Moderately stable	2 mph
N-2	Neutral	3 mph
N-10	Neutral	10 mph
U-2	Unstable	2 mph
U-10	Unstable	10 mph

Table 15.6-2a

RADIOLOGICAL EFFECTS OF THE STEAM LINE BREAK ACCIDENT
(Alternative Source Term)

	Case 1 (normal equilibrium – 0.2 µC/g)	Case 2 (iodine spike – 4.0 µC/g)
Regulatory Limits:	Control Room: 5.0; EAB & LPZ: 2.5	Control Room 5.0; EAB & LPZ: 25
Location	Dose (Rem TEDE)	Dose (Rem TEDE)
EAB	0.167	3.32
LPZ	0.0168	0.335
CR	0.189	3.79

Table 15.6-3*

LOSS-OF-COOLANT ACCIDENT
PRIMARY CONTAINMENT AIRBORNE FISSION PRODUCT INVENTORY
(INITIAL CORE)

(Historical Information)

Time After Accident	Noble Gases (curies)	Halogens (curies)
1 minute	1.6×10^6	4.3×10^4
30 minutes	5.2×10^5	2.7×10^4
1 hour	4.7×10^5	2.5×10^4
2 hours	4.5×10^5	2.4×10^4
12 hours	3.3×10^5	1.3×10^4
15 hours	3.2×10^5	1.2×10^4
1 day	2.7×10^5	1.0×10^4
2 days	2.5×10^5	9.3×10^3
10 days	7.5×10^4	2.4×10^3
25 days	1.2×10^4	4.7×10^2

* 1. The values in this table were determined to be unchanged due to the introduction of ATRIUM-9B fuel since the source terms are based on the TID 14844.

2. This table was valid prior to Extended Power Uprate. It is retained for historical purpose.

Table 15.6-4*

LOSS-OF-COOLANT ACCIDENT
REACTOR BUILDING AIRBORNE FISSION PRODUCT INVENTORY
(INITIAL CORE)

(Historical Information)

Time After Accident	Noble Gases (curies)	Halogens (curies)
1 minute	8.8×10^{0}	2.4×10^{-1}
30 minutes	4.8×10^{1}	2.5×10^{0}
1 hour	8.9×10^{1}	4.8×10^{0}
2 hours	1.6×10^{2}	8.6×10^{0}
12 hours	6.0×10^{2}	2.4×10^{1}
15 hours	6.6×10^{2}	2.6×10^{1}
1 day	7.4×10^{2}	2.8×10^{1}
2 days	8.3×10^{2}	3.1×10^{1}
10 days	2.0×10^{2}	6.5×10^{0}
25 days	2.5×10^{1}	9.5×10^{-1}

* 1. The values in this table were determined to be unchanged due to the introduction of ATRIUM-9B fuel since the source terms are based on the TID 14844.

2. This table was valid prior to Extended Power Uprate. It is retained for historical purpose.

Table 15.6-5*

LOSS-OF-COOLANT ACCIDENT
FISSION PRODUCT RELEASE RATE FROM CHIMNEY (INITIAL CORE)

(Historical Information)

Time After Accident	Noble Gases (curies/sec)	Halogens (curies/sec)
1 minute	1.0×10^{-4}	2.8×10^{-8}
30 minutes	5.6×10^{-4}	2.9×10^{-7}
1 hour	1.0×10^{-3}	5.5×10^{-7}
2 hours	1.9×10^{-3}	1.0×10^{-6}
12 hours	7.0×10^{-3}	2.8×10^{-6}
15 hours	7.7×10^{-3}	3.0×10^{-6}
1 day	8.6×10^{-3}	3.2×10^{-6}
2 days	9.7×10^{-3}	3.6×10^{-6}
10 days	2.4×10^{-3}	7.5×10^{-7}
25 days	2.9×10^{-4}	1.1×10^{-7}

* 1. The values in this table were determined to be unchanged due to the introduction of ATRIUM-9B fuel since the source terms are based on the TID 14844.

2. This table was valid prior to Extended Power Uprate. It is retained for historical purpose.

Table 15.6-5a

ISOTOPIC CORE INVENTORY (Ci/MWt)
(Alternative Source Term)

Isotope	Ci/MW$_t$	Isotope	Ci/MW$_t$	Isotope	Ci/MW$_t$
CO-58	1.529E+02	RU-103	4.311E+04	CS-136	2.379E+03
CO-60	1.830E+02	RU-105	3.034E+04	CS-137	4.928E+03
KR-85	4.364E+02	RU-106	1.837E+04	BA-139	4.888E+04
KR-85M	6.772E+03	RH-105	2.882E+04	BA-140	4.714E+04
KR-87	1.291E+04	SB-127	2.999E+03	LA-140	5.055E+04
KR-88	1.815E+04	SB-129	8.877E+03	LA-141	4.447E+04
RB-86	7.096E+01	TE-127	2.986E+03	LA-142	4.286E+04
SR-89	2.428E+04	TE-127M	4.060E+02	CE-141	4.465E+04
SR-90	3.528E+03	TE-129	8.735E+03	CE-143	4.101E+04
SR-91	3.081E+04	TE-129M	1.300E+03	CE-144	3.682E+04
SR-92	3.362E+04	TE-131M	3.955E+03	PR-143	3.963E+04
Y-90	3.625E+03	TE-132	3.850E+04	ND-147	1.800E+04
Y-91	3.155E+04	I-131	2.710E+04	NP-239	5.587E+05
Y-92	3.377E+04	I-132	3.914E+04	PU-238	1.768E+02
Y-93	3.942E+04	I-133	5.501E+04	PU-239	1.474E+01
ZR-95	4.443E+04	I-134	6.035E+04	PU-240	2.001E+01
ZR-97	4.497E+04	I-135	5.157E+04	PU-241	6.700E+03
NB-95	4.464E+04	XE-133	5.282E+04	AM-241	9.857E+00
MO-99	5.121E+04	XE-135	2.144E+04	CM-242	2.285E+03
TC-99M	4.484E+04	CS-134	8.009E+03	CM-244	1.621E+02

Table 15.6-5b

REACTOR BUILDING ISOTOPIC CORE INVENTORY (Ci/MWt)
(Alternative Source Term)

Post-LOCA Reactor Building Isotopic Inventory

Isotope	Post-LOCA Reactor Building Isotopic Inventory (Ci) Containment + ESF Leakage						Total Activity (Ci)
	0.667 hr	2.0 hr	4.0 hrs	8.0 hrs	16 hrs	24 hrs	
Co-58	9.423E-03	6.372E-01	1.267E+00	9.365E-01	3.895E-01	1.583E-01	3.398E+00
Co-60	1.128E-02	7.633E-01	1.519E+00	1.124E+00	4.691E-01	1.912E-01	4.078E+00
Kr-85	3.477E+01	9.031E+02	3.193E+03	6.467E+03	9.829E+03	1.113E+04	3.155E+04
Kr-85m	4.866E+02	1.028E+04	2.669E+04	2.910E+04	1.283E+04	4.212E+03	8.360E+04
Kr-87	7.151E+02	8.981E+03	1.068E+04	2.443E+03	4.743E+01	6.856E-01	2.286E+04
Kr-88	1.229E+03	2.305E+04	5.003E+04	3.817E+04	8.233E+03	1.323E+03	1.220E+05
Rb-86	3.777E+00	3.420E+01	5.980E+01	4.344E+01	1.787E+01	7.194E+00	1.663E+02
Sr-89	1.197E+01	8.093E+02	1.609E+03	1.188E+03	4.935E+02	2.003E+02	4.312E+03
Sr-90	1.740E+00	1.177E+02	2.343E+02	1.734E+02	7.236E+01	2.950E+01	6.290E+02
Sr-91	1.447E+01	8.885E+02	1.528E+03	8.448E+02	1.967E+02	4.472E+01	3.517E+03
Sr-92	1.398E+01	6.726E+02	8.025E+02	2.136E+02	1.152E+01	6.066E-01	1.715E+03
Y-90	2.003E-02	2.432E+00	9.351E+00	1.396E+01	1.136E+01	6.708E+00	4.383E+01
Y-91	1.559E-01	1.071E+01	2.190E+01	1.699E+01	7.503E+00	3.147E+00	6.041E+01
Y-92	4.795E-01	1.420E+02	4.975E+02	3.582E+02	5.532E+01	6.056E+00	1.060E+03
Y-93	1.857E-01	1.147E+01	1.989E+01	1.119E+01	2.697E+00	6.349E-01	4.607E+01
Zr-95	2.190E-01	1.481E+01	2.945E+01	2.176E+01	9.048E+00	3.675E+00	7.897E+01
Zr-97	2.158E-01	1.382E+01	2.534E+01	1.592E+01	4.786E+00	1.405E+00	6.150E+01
Nb-95	2.201E-01	1.490E+01	2.964E+01	2.194E+01	9.154E+00	3.731E+00	7.958E+01
Mo-99	3.135E+00	2.092E+02	4.076E+02	2.893E+02	1.110E+02	4.160E+01	1.062E+03
Tc-99m	2.765E+00	1.868E+02	3.703E+02	2.693E+02	1.076E+02	4.165E+01	9.784E+02
Ru-103	2.656E+00	1.796E+02	3.568E+02	2.633E+02	1.092E+02	4.427E+01	9.558E+02
Ru-105	1.685E+00	9.261E+01	1.349E+02	5.347E+01	6.399E+00	7.482E-01	2.898E+02
Ru-106	1.132E+00	7.661E+01	1.524E+02	1.128E+02	4.704E+01	1.916E+01	4.092E+02
Rh-105	1.777E+00	1.198E+02	2.353E+02	1.666E+02	6.125E+01	2.156E+01	6.062E+02
Sb-127	3.679E+00	2.465E+02	4.832E+02	3.471E+02	1.364E+02	5.236E+01	1.269E+03
Sb-129	9.834E+00	5.373E+02	7.757E+02	3.022E+02	3.494E+01	3.945E+00	1.664E+03
Te-127	3.674E+00	2.474E+02	4.888E+02	3.560E+02	1.438E+02	5.674E+01	1.296E+03
Te-127m	5.006E-01	3.388E+01	6.742E+01	4.993E+01	2.084E+01	8.498E+00	1.811E+02
Te-129	1.020E+01	6.005E+02	9.525E+02	4.324E+02	1.016E+02	2.874E+01	2.126E+03
Te-129m	1.603E+00	1.085E+02	2.159E+02	1.596E+02	6.624E+01	2.683E+01	5.787E+02

Table 15.6-5b (Continued)

REACTOR BUILDING ISOTOPIC CORE INVENTORY (Ci/MWt)
(Alternative Source Term)

Post-LOCA Reactor Building Isotopic Inventory

Isotope	Post-LOCA Reactor Building Isotopic Inventory (Ci) Containment + ESF Leakage						Total Activity (Ci)
	0.667 hr	2.0 hr	4.0 hrs	8.0 hrs	16 hrs	24 hrs	
Te-131m	4.802E+00	3.150E+02	5.986E+02	4.040E+02	1.401E+02	4.748E+01	1.510E+03
Te-132	4.719E+01	3.155E+03	6.169E+03	4.408E+03	1.713E+03	6.506E+02	1.614E+04
I-131	1.474E+03	1.508E+04	2.696E+04	2.062E+04	1.034E+04	6.041E+03	8.051E+04
I-132	1.880E+03	1.626E+04	1.981E+04	7.778E+03	2.104E+03	7.834E+02	4.862E+04
I-133	2.934E+03	2.883E+04	4.856E+04	3.297E+04	1.303E+04	6.001E+03	1.323E+05
I-134	1.943E+03	6.953E+03	2.576E+03	8.455E+01	7.809E-02	8.407E-05	1.156E+04
I-135	2.622E+03	2.342E+04	3.419E+04	1.744E+04	3.888E+03	1.010E+03	8.257E+04
Xe-133	4.207E+03	1.089E+05	3.815E+05	7.563E+05	1.100E+06	1.191E+06	3.542E+06
Xe-135	1.776E+03	4.546E+04	1.453E+05	2.205E+05	1.817E+05	1.113E+05	7.061E+05
Cs-134	4.267E+02	3.872E+03	6.790E+03	4.963E+03	2.066E+03	8.419E+02	1.896E+04
Cs-136	1.266E+02	1.145E+03	1.999E+03	1.449E+03	5.929E+02	2.374E+02	5.550E+03
Cs-137	2.626E+02	2.382E+03	4.178E+03	3.055E+03	1.272E+03	5.185E+02	1.167E+04
Ba-139	1.724E+01	5.966E+02	4.342E+02	4.300E+01	3.212E-01	2.343E-03	1.091E+03
Ba-140	2.321E+01	1.566E+03	3.102E+03	2.276E+03	9.325E+02	3.733E+02	8.272E+03
La-140	2.948E-01	4.263E+01	1.788E+02	2.752E+02	2.215E+02	1.276E+02	8.460E+02
La-141	1.950E-01	1.043E+01	1.458E+01	5.332E+00	5.426E-01	5.395E-02	3.114E+01
La-142	1.566E-01	5.819E+00	4.712E+00	5.775E-01	6.605E-03	7.379E-05	1.127E+01
Ce-141	5.504E-01	3.722E+01	7.398E+01	5.460E+01	2.263E+01	9.160E+00	1.981E+02
Ce-143	4.986E-01	3.280E+01	6.259E+01	4.260E+01	1.503E+01	5.178E+00	1.587E+02
Ce-144	4.539E-01	3.071E+01	6.110E+01	4.521E+01	1.885E+01	7.678E+00	1.640E+02
Pr-143	1.955E-01	1.327E+01	2.655E+01	1.986E+01	8.424E+00	3.471E+00	7.176E+01
Nd-147	8.861E-02	5.975E+00	1.183E+01	8.664E+00	3.540E+00	1.413E+00	3.151E+01
Np-239	6.832E+00	4.548E+02	8.831E+02	6.224E+02	2.355E+02	8.701E+01	2.290E+03
Pu-238	2.180E-03	1.475E-01	2.935E-01	2.173E-01	9.067E-02	3.696E-02	7.881E-01
Pu-239	1.817E-04	1.230E-02	2.448E-02	1.813E-02	7.572E-03	3.089E-03	6.576E-02
Pu-240	2.467E-04	1.669E-02	3.322E-02	2.459E-02	1.026E-02	4.183E-03	8.919E-02
Pu-241	8.260E-02	5.589E+00	1.112E+01	8.233E+00	3.435E+00	1.400E+00	2.986E+01
Am-241	4.862E-05	3.291E-03	6.552E-03	4.856E-03	2.031E-03	8.301E-04	1.761E-02
Cm-242	1.127E-02	7.622E-01	1.516E+00	1.122E+00	4.674E-01	1.902E-01	4.069E+00
Cm-244	7.994E-04	5.409E-02	1.076E-01	7.968E-02	3.325E-02	1.355E-02	2.890E-01

Table 15.6-6*

OFFSITE RADIOLOGICAL EFFECTS OF THE LOSS-OF-COOLANT ACCIDENT (INITIAL CORE)
(This table is retained for historical purposes)

	First 2 — Hour Dose						Total Dose					
Distance (Miles)	VS-1	MS-1	N-1	N-5	U-1	U-5	VS-1	MS-1	N-1	N-5	U-1	U-5
Passing Cloud Whole Body Dose (rem)												
1/4	1.2×10^{-5}	1.2×10^{-5}	1.2×10^{-5}	2.0×10^{-6}	1.7×10^{-5}	2.5×10^{-6}	3.6×10^{-4}	3.6×10^{-4}	3.7×10^{-4}	6.3×10^{-5}	5.3×10^{-4}	7.9×10^{-5}
1	6.8×10^{-6}	6.8×10^{-6}	8.2×10^{-6}	1.2×10^{-6}	6.2×10^{-6}	9.8×10^{-7}	2.1×10^{-4}	2.1×10^{-4}	2.6×10^{-4}	3.8×10^{-5}	1.9×10^{-4}	3.1×10^{-5}
3	3.0×10^{-6}	3.2×10^{-6}	2.9×10^{-6}	4.8×10^{-7}	1.2×10^{-6}	2.4×10^{-7}	9.5×10^{-5}	1.0×10^{-4}	9.1×10^{-5}	1.5×10^{-5}	3.7×10^{-5}	7.5×10^{-6}
5	1.9×10^{-6}	2.1×10^{-6}	1.4×10^{-6}	2.5×10^{-7}	4.7×10^{-7}	1.1×10^{-7}	6.0×10^{-5}	6.6×10^{-5}	4.3×10^{-5}	7.9×10^{-6}	1.5×10^{-5}	3.5×10^{-6}
10	9.5×10^{-7}	1.0×10^{-6}	3.9×10^{-7}	9.4×10^{-8}	1.2×10^{-7}	3.9×10^{-8}	3.0×10^{-5}	3.2×10^{-5}	1.2×10^{-5}	3.0×10^{-6}	3.0×10^{-6}	1.2×10^{-6}
Lifetime Thyroid Dose (rem)												
1/4	a	a	8.0×10^{-10}	a	3.2×10^{-6}	2.0×10^{-7}	a	a	3.3×10^{-8}	a	1.3×10^{-4}	8.2×10^{-6}
1	a	9.6×10^{-9}	1.8×10^{-6}	1.6×10^{-7}	1.6×10^{-6}	2.2×10^{-7}	a	4.0×10^{-7}	7.4×10^{-5}	6.6×10^{-6}	6.8×10^{-5}	9.2×10^{-6}
3	a	2.3×10^{-7}	6.4×10^{-7}	1.1×10^{-7}	2.8×10^{-7}	4.8×10^{-8}	a	9.4×10^{-6}	2.6×10^{-5}	4.5×10^{-6}	1.2×10^{-5}	2.0×10^{-6}
5	a	3.5×10^{-7}	3.1×10^{-7}	5.8×10^{-8}	1.3×10^{-7}	2.3×10^{-8}	a	1.4×10^{-5}	1.3×10^{-5}	2.4×10^{-6}	5.3×10^{-6}	9.4×10^{-7}
10	a	3.5×10^{-7}	1.1×10^{-7}	2.3×10^{-8}	4.4×10^{-8}	8.2×10^{-9}	3.9×10^{-9}	1.4×10^{-5}	4.6×10^{-6}	9.4×10^{-7}	1.8×10^{-6}	3.4×10^{-7}
	First 2 — Hour Dose						Total Dose					

Table 15.6-6* (Cont'd)

Distance (Miles)	VS-1	MS-1	N-1	N-5	U-1	U-5	VS-1	MS-1	N-1	N-5	U-1	U-5
	Whole Body Fallout Dose (rem)											
1/4	a	a	a	a	5.5×10^{-9}	1.7×10^{-9}	a	a	a	a	5.3×10^{-7}	1.7×10^{-7}
1	a	a	1.4×10^{-9}	6.1×10^{-10}	2.9×10^{-9}	1.9×10^{-9}	a	4.0×10^{-10}	1.3×10^{-7}	5.9×10^{-8}	2.7×10^{-7}	1.9×10^{-7}
3	a	a	4.9×10^{-10}	4.2×10^{-10}	5.0×10^{-10}	4.2×10^{-10}	a	9.5×10^{-9}	4.7×10^{-8}	4.0×10^{-8}	4.8×10^{-8}	3.1×10^{-8}
5	a	1.5×10^{-10}	2.4×10^{-10}	2.3×10^{-10}	2.2×10^{-10}	2.0×10^{-10}	a	1.5×10^{-8}	2.3×10^{-8}	2.2×10^{-8}	2.1×10^{-8}	1.9×10^{-8}
10	a	1.5×10^{-10}	a	a	a	a	a	1.5×10^{-8}	8.3×10^{-9}	8.3×10^{-9}	7.3×10^{-9}	6.8×10^{-9}

(a) denotes less than 10

* 1. The values in this table were determined to be unchanged due to the introduction of ATRIUM-9B fuel since the source terms are based on the TID 14844.

	Meterology	Wind Speed (m/sec)
VS-1	Very Stable	1
MS-1	Moderately Stable	1
N-1	Neutral	5
N-5	Neutral	1
U-1	Unstable	1
U-5	Unstable	5

2. This table is valid prior to Extended Power Uprate. It is retained for historical purpose.

Table 15.6-7

LOSS-OF-COOLANT ACCIDENT INPUT PARAMETERS
(Alternative Source Term)

		Design Basis Assumption
I.	**Data and Assumptions Used to Estimate Radioactive Source from Postulated Accidents.**	
A.	Power level, MWt	3016*
B.	Fuel Burnup	
	GE14: Peak Fuel Burnup, MWD/MTU	62,000
	Optima2: Average Fuel Burnup, MWD/MT	37,000
C.	Fission Product Release Fractions	(NRC Regulatory Guide 1.183, Table 1)

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

		Design Basis Assumption
D.	Iodine chemical forms	
	Organic	0.15%
	Elemental	4.85%
	Aerosol (CsI)	95%
E.	Fission product release timing	(NRC Regulatory Guide 1.183, Table 4)

Phase	Onset	Duration
Gap Release	2 min	0.5 hr
Early In-vessel Release	0.5 hr	1.5 hr

Table 15.6-7 (continued)

LOSS-OF-COOLANT ACCIDENT INPUT PARAMETERS
(Alternative Source Term)

			Design Basis Assumption
II.	Data and Assumptions Used to Estimate Activity Released		
	A.	Primary containment leak rate (total), %/day	3.00
	B.	Total MSIV leakage (at 48 psig)	150 scfh (0 to 30 days); 60 scfh max per line
	C.	Volume of primary containment (drywell plus suppression chamber free air volume), cu. ft.	2.69×10^5
	D.	Drywell surface area, sq. ft	32,430
	E.	Volume of suppression pool water, cu. ft	110,000
	F.	Primary containment leak rate which goes to secondary, %/day	2.187
	G.	Primary containment leak rate which goes through MSIVs, %/day	0.813
	H.	SGTS adsorption and filtration efficiencies, %	
		Organic Iodide	50
		Elemental Iodine	50
		Particulate Aerosols	99
	I.	Secondary containment leak rate, %vol/day	0

Table 15.6-7 (continued)

LOSS-OF-COOLANT ACCIDENT INPUT PARAMETERS
(Alternative Source Term)

			Design Basis Assumption
	J.	Main Steam Line Deposition	
		Two-node treatment, each well mixed, is used for each steam line in which flow occurs. The first node is from the reactor vessel to the inboard MSIV. The second node is from the inboard MSIV to the outboard MSIV. Gravitational settling applied to aerosols on horizontal pipe projected areas. For Elemental Iodine Deposition, a DF of 2 or elemental removal efficiency of 50% is used per AEB 98-03, Appendix B. No credit is taken for holdup or plate-out in the main steam lines beyond the outboard MSIVs. No credit is taken for holdup and plate-out in the main condenser.	
III.	Data for Control Room		
	A.	Volume of control room habitable zone, cu. ft.	1.84×10^5
	B.	Volume of control room proper, cu. ft.	5.83×10^4
	C.	Control room intake flow, scfm	2000 +/- 10%
	D.	Control room intake charcoal adsorption efficiencies, %	
		Organic Iodide	99
		Elemental Iodine	99
		Particulate Aerosols	99
	E.	Control room intake, isolation (following LOCA), min	0
	F.	Control room filter unit start (following LOCA), min	40
	G.	Unfiltered inleakage (0 to 40 min), scfm	60,000
	H.	Unfiltered inleakage (40 min to 720 hrs), scfm	400

Table 15.6-7 (continued)

LOSS-OF-COOLANT ACCIDENT INPUT PARAMETERS
(Alternative Source Term)

		Design Basis Assumption
I.	Control room cleanup recirculation flowrate, scfm	0
J.	Occupancy factors	
	0 to 1 day	1.0
	1 to 4 days	0.6
	4 to 30 days	0.4
K.	CR Operator Breathing Rates, m^3/sec	3.5E-04

* The reactor power level after Extended Power Uprate is 2957 MWt. The radiological consequences are evaluated at 3016 MWt to include the 2% instrument error.

Table 15.6-8*

LOSS-OF-COOLANT ACCIDENT
CONTROL ROOM RADIOLOGICAL EFFECTS (rem) [Pre-uprate]

(Historical Information)

	Thyroid	Wholebody	Beta
MSIV Leakage			
Activity inside control room	7.58	$1.36x\ 10^{-2}$	0.55
Plume shine	__	2.03×10^{-3}	--
Direct shine	--	5.70×10^{-2}	--
Stack Release			
Activity inside control room	15.17	2.25×10^{-1}	8.16
Plume shine	--	1.66×10^{-2}	--
Total Control Room Doses	22.8	0.314	8.71
SRP 6.4 Guidelines	30	5	30

* The values in this table were determined to be unchanged due to the introduction of ATRIUM-9B fuel since the source terms are based on the TID 14844.

* This table is retained for historical purposes.

15.6-8a
Loss-of-Coolant Accident
EAB, LPZ and Control Room Doses following EPU
(Alternative Source Term)

Location	Duration	TEDE (rem)	Regulatory Limit TEDE (rem)
Control Room	30 days	4.08*	5
EAB	Maximum, 2 hours	8.47	25
LPZ	30 days	2.63	25

* The doses here include the external cloud shine, control room filter shine, and inhalation doses from radioactivity drawn into the control room.

Table 15.6-9

Atmospheric Dispersion Factors for LOCA
(Alternative Source Term)

Receptor	Time Interval	Ground Level (χ/Q) (s/m^3)	Elevated (χ/Q) (s/m^3)
EAB	0 - 0.5 hr	----	1.57E-4
	0.5 - 2 hr	----	6.38E-6
	0 - 2 hr	1.36E-3	----
LPZ	0 - 0.5 hr	----	3.01E-5
	0.5 - 2 hr	----	2.05E-5
	0 - 2 hr	1.04E-4	----
	2 - 8 hr	4.14E-5	8.76E-6
	8 - 24 hr	2.62E-5	5.73E-6
	24 - 96 hr	9.96E-6	2.28E-6
	96 - 720 hr	2.52E-6	6.07E-7
Control Room	0 - 2 hr	1.02E-3	5.84E-6
	2 - 8 hr	8.23E-4	2.68E-6
	8 - 24 hr	3.55E-4	1.81E-6
	24 - 96 hr	2.32E-4	7.77E-7
	96 - 720 hr	1.38E-4	2.30E-7

15.7 RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT

Because they continue to be bounded by generic analyses or analyses for previous fuel cycles, the events described in Sections 15.7.1 and 15.7.3 have not been reanalyzed for the current fuel cycle or EPU. These events, including the associated assumptions and conclusions, continue to be part of the plant's licensing basis. The conclusions of these analyses are still valid; however, specific details contained in the descriptions and associated figures should be used only to understand the analysis and its conclusions. These specific details should not be used as sources of current fuel cycle design information.

15.7.1 Postulated Liquid Releases Due to Liquid Tank Failures

The offsite consequences of a failure of the river discharge tank concurrent with a failure of the basin or a failure of the drain line below this basin would be the eventual disposal of the tank contents into the river. Assuming entry of 2×10^5 μCi (Table 11.2-1 shows the full tank contents as 2×10^4 μCi) into the river over a period of 1 hour, the activity concentration in the river would be about 4×10^{-8} μCi/cc based on an average river flow of 47,000 ft^3/s. The maximum permissible concentration (MPC) for an unidentified mixture is 10^{-7} μCi/cc. Thus, in the worst case the river concentration over an hour would still be lower than the MPC for the mixture. [15.7-1]

Table 11.2-5 depicts the results of an analysis for a failure of the tank farm. It was assumed that a tornado scattered the contents of all tank farm radwaste tanks on the ground and in the catch basin around the tanks. The entire volume of water was assumed to drain into the river in 1 hour. It can be seen from the table that the resultant concentrations of radionuclides mixed with an average river flow of 47,000 ft^3/s are approximately four orders of magnitude under the MPC values set forth in 10 CFR 20.

Tables 11.2-1 and 11.2-5 are based on operating data from 1980 and include additional data to that contained in the original estimates provided in the FSAR. However, the 1980 data would result in lower releases than those assumed in the FSAR, and the conclusions provided in the FSAR and discussed above are still bounding and valid.

15.7.2 Design Basis Fuel Handling Accidents Inside Containment and Spent Fuel Storage Buildings

See the introduction to Section 15.7 for information regarding use of details from this analysis description which may not be applicable to the current fuel cycle.

15.7.2.1 Identification

During a refueling operation the primary containment (drywell and suppression chamber) and the reactor vessel are open; the secondary containment (reactor building) serves as the major barrier to the release of radioactive materials. The accident is assumed to occur when a fuel assembly is accidentally dropped onto the top of the core during fuel handling operations. [15.7-2]

15.7.2.2 Designed Safeguards

The reactor core is designed to remain subcritical with one control rod fully withdrawn and all other control rods fully inserted, even if it is assumed that a fresh fuel assembly is dropped into an empty fuel space in an otherwise fully constituted core. At least two control rods adjacent to the empty fuel space would have to be withdrawn for a nuclear excursion to occur. Electrical interlocks that meet the single failure criteria are included in the design to prevent such a configuration from occurring. [15.7-3]

With the reactor mode switch in REFUEL or STARTUP a rod withdrawal interlock prevents any withdrawal whenever the travel limit switch on the refueling platform indicates that the platform is carrying fuel over the reactor core.

With the reactor mode switch in REFUEL, a rod withdrawal interlock prevents the withdrawal of more than one control rod. When any one rod position indicator shows that a rod is withdrawn from the fully inserted position, the interlock prevents the withdrawal of any other rod.

When any rod position indicator shows a rod is withdrawn, an interlock prevents the movement of the refueling platform toward a position over the reactor core while the hoists are carrying fuel.

Each fuel hoist is equipped with a load limit switch and two independent travel limit switches and a mechanical stop to prevent damage due to upward vertical movement. To drop the fuel assembly, either the assembly bale, the fuel grapple, or the grapple cable would have to break. Section 9.1.4 provides additional details regarding refueling platform controls and interlocks.

15.7.2.3 Procedural Safeguards

Procedures require the reactor control operator to observe rod position instrumentation and to be in communication with the refueling operator during all fuel loading operations. The reactor is verified to be subcritical by observing the source range monitor count rate which is maintained greater than the minimum value specified in the Technical Specifications, when fuel is being moved in the core. [15.7-4]

15.7.2.4 Accident Analysis

Dropping a fuel assembly onto the reactor core is assumed to occur under nonoperating conditions for an 8x8, 9x9, or 10x10 fuel array. The key assumption of this postulated occurrence is the inadvertent mechanical damage to the fuel rod cladding as a consequence of the fuel assembly being dropped on the core while in the cold condition. Therefore, fuel densification considerations do not affect the accident analysis results. [15.7-5]

15.7.2.4.1 Methods, Assumptions and Conditions

The assumptions and analyses applicable to this type of fuel handling accident are as follows:

A. The fuel assembly is dropped 34 feet to impact the core (from the maximum height allowed by the fuel handling equipment).

B. The entire amount of potential energy, including the energy of the entire fuel assembly falling to its side from a vertical position (referenced to the top of the reactor core), is available for application to the fuel assemblies involved in the accident. This assumption neglects the dissipation of some of the mechanical energy of the falling fuel assembly in the water above the core and requires that the assembly separates from the grapple head.

C. None of the energy associated with the dropped fuel assembly is absorbed by the fuel material (uranium dioxide).

D. All fuel rods, including tie rods, are assumed to fail by 1% strain in compression, the same mode as ordinary fuel rods. For the fuel designs considered here, there is no propensity for preferential failure of tie rods.

E. As stated above, the fuel handling accident (FHA) evaluates the consequences of the inadvertent dropping of a fuel assembly onto other fuel assemblies already loaded in the core. However, the current regulatory guidance, Standard Review Plan (SRP) 15.7.4 (Reference 8), assumes the dropped mass to include the weight of the refueling mast. As such, General Electric performs the FHA evaluation with a 619 lb. mast, the approximate weight of the NF-500 (Reference 6). [15.7-6]

F. It is assumed that 50% of the energy is absorbed by the dropped fuel assembly and that the remaining 50% is absorbed by the struck fuel assemblies in the core during each impact.

G. It is assumed that the refueling mast adds kinetic energy (when included in the analysis), and dissipates all of this energy during impact. Upon impact, half the energy is absorbed by the struck fuel assemblies and half is absorbed by the dropped fuel assembly.

H. The current GE analysis for the Fuel Handling Accident (FHA) as found in GESTAR II[6] analyzes the drop of a GE11 fuel assembly. The GE11 fuel assembly has a 9x9 fuel rod array as opposed to the 8x8 fuel rod array in a GE10 fuel assembly. Previous versions of GESTAR have analyzed the 8x8 fuel assembly, but assumed a lower drop height and a lighter refueling mast. Therefore, the analysis used is for a GE11 fuel assembly with a 9x9 array of fuel rods. After the number of failed fuel rods is determined, the results are ratioed using methodology specified in Reference 6 to make the results applicable to the GE10 fuel assembly.

15.7.2.4.2 Analysis and Results

Because of the complex nature of the impact and the resulting damage to fuel assembly components, a rigorous predication of the number of failed rods is not possible. For this reason, a simplified energy approach is taken and numerous conservative assumptions are made to assure a conservative estimate of the number of failed rods.

The number of failed fuel rods is determined by balancing the energy of the dropped fuel assembly against the energy required to fail a rod.

No energy is considered to be absorbed by the fuel pellets (i.e., the energy is absorbed entirely by the nonfuel components of the assemblies). The energy available for cladding deformation is considered to be proportional to the mass ratio:

$$\frac{\text{mass of cladding}}{\text{mass of assembly - mass of fuel pellets}}$$

The kinetic energy acquired by the falling fuel assembly/refueling mast and the amount absorbed by the cladding of the struck assemblies during the first impact is: [15.7-7]

	Energy Dissipated By The First Impact	Ratio of Cladding Mass to Non-Fuel Assembly Components Mass	Energy Absorbed By The Cladding of The Impacted Fuel Assemblies
GE: GE11 Assembly Plus Refueling Mast (ft-lb)	(562 + 619) * 34 = 40154 ft-lb	0.51	0.5 * 40154 * 0.51 = 10239.27
GNF: GE14 Assembly plus Refueling Mast	(629 + 619) * 34 = 42432 ft-lb.	0.526	0.5 * 42432 * 0.526 = 11160 ft-lb.
Westinghouse: SVEA-96 Optima2 Assembly plus Refueling Mast	(629 + 619) * 34 = 42432 ft-lb	Not Evaluated	4899 ft-lb*

* Half of the energy is assumed absorbed by the falling assembly and all 96 rods are assumed to fail. Of the remaining energy, most is absorbed by the fuel channel. The remaining amount (4899 ft-lb) is absorbed by the cladding in the impacted fuel assembly and results in approximately 20 additional rod failures.

where:

562	=	weight of the GE11 fuel assembly
619	=	weight of the mast (GE)
629	=	weight of the GE14 and SVEA-96 Optima2 fuel assemblies
34	=	drop height

The dropped fuel assembly was considered to impact at a small angle, subjecting all the fuel rods in the dropped assembly to bending moments. The fuel rods are expected to absorb little energy prior to failure as a result of bending. For this reason, it is assumed that all the rods in the dropped fuel assembly fail.

The assembly was assumed to tip over and impact horizontally on top of the core. The remaining energy was used to predict the number of additional rod failures.

The number of fuel rod failures caused by compression is determined as follows:

Evaluation Results for SVEA-96 Optima2 Fuel, GE14 Fuel and GE11 Fuel

	Fuel Rod Failures Caused By Compression During First Impact	Fuel Rod Failures During The First Impact	Fuel Rod Failures Caused By Tipping Impact	Total Number of Fuel Rod Failures
GE: GE11 Assembly Plus Refueling Mast (ft-lb)	$\frac{0.5 * 40154 * 0.51}{200 \text{ ft-lb}}$ = 51 rods	74 rods + 51 rods 125 rods failed	$\frac{0.5 * 562 * 0.51 * 6.67}{200 \text{ ft-lb}}$ + $\frac{0.5 * 619 * 0.51 * 13.3}{200 \text{ ft-lb}}$ = 15 rods	First Impact 125 rods + Tipping Impact 15 rods 140 rods failed
GNF: GE14 Assembly plus Refueling Mast	$\frac{0.5*42432*0.526}{175}$ = 64 rods	92 rods + 64 rods 156 rods failed	$\frac{0.5*629*0.526*6.67}{175}$ + $\frac{0.5*619*0.526*13.3}{175}$ = 20 rods	First Impact 156 rods + Tipping Impact 20 rods 176 rods failed*
Westinghouse: SVEA-96 Optima2 Assembly plus Refueling Mast	$\frac{4899}{248}$ = 20 rods	96 rods + 20 rods 116 rods failed	0 rods**	First Impact 116 rods + Tipping Impact 0 rods 116 rods failed

* The number of failed rods calculated for the GE14 fuel is greater than the number provided in NEDC-32868P Rev. 1 due to extra conservatism in the analysis shown here.

** The SVEA-96 Optima2 fuel handle is attached directly to the channel, so any additional energy caused by tipping will first be absorbed by the fuel channels of the affected assemblies. No fuel rod failures are expected due to tipping because this energy would be distributed over a number of fuel channels.

where:

0.526 = ratio of cladding material mass to non-fuel assembly components mass (GE14)
0.51 = ratio of cladding material mass to non-fuel assembly components mass (GE11)
248 ft-lb is the energy absorbed by the cladding prior to failure (SVEA-96 Optima2)
200 ft-lb is the energy absorbed by the cladding prior to failure (GE11)
175 ft-lb is the energy absorbed by the cladding prior to failure (GE14)
0.5 = fraction absorbed by the struck assemblies
15 = change in height of grapple when tipping occurs (SPC)

13.3	=	change in height of the grapple when tipping occurs (GE)
6.67	=	midplane change in height of the assembly when tipping on its side (GE)
562	=	weight of the GE11 fuel assembly
629	=	weight of the GE14 and SVEA-96 Optima2 fuel assemblies
619	=	weight of the mast (GE and SVEA-96 Optima2)

The current GE analysis which determines the number of GE fuel rods failing as the result of a fuel bundle drop uses methodology presented in Reference 6. This analysis is based on a GE11 assembly which has a 9x9 fuel rod configuration.

The impact of the transition to Westinghouse SVEA-96 Optima2 fuel on the fuel handling accident consequences has been determined in accordance with the methodology described in Reference 32. The analysis for the Westinghouse SVEA-96 Optima2 fuel determined that a total of 116 fuel rods would be damaged (96 rods in the dropped assembly and 20 rods in the impacted assembly). Since the number of failed fuel rods is significantly fewer than estimated for the GE14 design and the rods that fail do not increase the release, the radiological consequences developed based upon the 7x7 fuel design key inputs and assumptions will be bounding for the Westinghouse SVEA-96 Optima2 fuel design.

15.7.2.5 Radiological Effects

15.7.2.5.1 Application of Alternative Source Term Methodology

Regulation 10 CFR 50.67, "Accident Source Term," provides a regulatory mechanism for power reactor licensees to voluntarily replace the traditional accident source term used in design basis accident analyses (i.e., TID 14844, Reference 18) with an "Alternative Source Term" (AST). The methodology for this approach is provided in NRC Regulatory Guide 1.183 (Reference 19).

Accordingly, Quad Cities applied for the AST methodology for key Design Basis Accidents (DBAs). In support of a full-scope implementation of AST as described in Reference 19, AST radiological consequence are performed for the four DBAs that result in offsite exposure consequences: Loss of Coolant Accident (LOCA), Main Steam Line Break (MSLB), Fuel Handling Accident (FHA), and Control Rod Drop Accident (CRDA).

The NRC approved AST for Quad Cities in Reference 20.

Fission Product Inventory
The fuel source term is based on the reactor source terms used for the LOCA dose assessment. The inventory of reactor core fission products is based on maximum full power operation at a power level of 3016 MWth, the Extended Power Uprate (EPU) thermal power of 2957 MWth plus 2% to account for uncertainties in accordance with NRC Regulatory Guide 1.49 (Reference 21). The AST source term values for this analysis were derived using the ORIGEN2 computer code (Reference 22) and the guidance outlined in NRC Regulatory Guide 1.183. Core isotopic nuclides and their curie per megawatt activities were utilized from Appendix D of GE task report GE-NE-A22-00103-64-01 (Reference 23) for input into the RADTRAD computer code, which was used to determine dose consequences (note that the CO-58 and CO-60 activities were obtained from the RADTRAD User's Manual). The Optima2 fuel was also evaluated and determined to be bounded by the GE results.

Core Inventory Release Fractions
The core inventory release fractions were determined using the guidance in NRC Regulatory Guide 1.183, Section 3.2 (Release Fractions).

Atmospheric Dispersion Factors
The main control room and offsite dose χ/Q values were calculated using the methodology described in UFSAR Section 2.3.6. The atmospheric relative concentrations used are as follows:

Main Control Room
$\chi/Q = 5.82E\text{-}04\ sec/m^3$ (0-2 hours)

Exclusion Area Boundary (EAB)
$\chi/Q = 1.36E\text{-}03\ sec/m^3$ (0-2 hours)

Low Population Zone (LPZ)
$\chi/Q = 1.04E\text{-}04\ sec/m^3$ (0-2 hours)

Release Paths

The FHA accident analysis postulates that a spent fuel assembly is dropped on top of irradiated fuel. The FHA is assumed to occur 24 hours following a reactor shutdown. The effective decontamination factor is 200 for depths greater than 23 ft., consistent with NRC Regulatory Guide 1.183 (the activity is released within two hours). The activity would be released from the reactor building via the reactor building ventilation stack. No credit is for reactor building holdup or dilution. In addition, the standby gas and control room treatment/filtration systems are not credited.

Dose Consequences

RADTRAD (Reference 24) is used to determine the dose consequences for the FHA. The FHA assessment takes no credit for control room isolation, emergency ventilation or filtration of intake air for the duration of the accident event. Dose conversion factors were obtained from Federal Guidance Reports 11 and 12 (References 25 and 26).

15.7.2.5.2 Acceptance Criteria

The AST acceptance criteria for postulated major credible accident scenarios are provided by 10 CFR 50.67 and Regulatory Guide 1.183. For the main control room, adequate radiation protection is provided to permit access to and occupancy under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident. The AST acceptance criteria for an individual located at any point on the boundary of the exclusion area (the Exclusion Area Boundary or EAB) are provided by 10 CFR 50.67 as 25 rem TEDE for any 2-hour period following the onset of the postulated fission product release. The AST acceptance criteria for an individual located at any point on the outer boundary of the low population zone (LPZ) are provided by 10 CFR 50.67 as 25 rem TEDE during the entire period of passage of the radioactive cloud resulting from the postulated fission product release. Regulatory Guide 1.183 applies the following additional limits to events with a higher probability of occurrence (including the FHA). For the FHA, doses at the EAB and LPZ should not exceed 6.3 rem TEDE.

15.7.2.5.3 Computer Models

AST calculations for the FHA were prepared to simulate the radionuclide release, transport, removal, and dose estimates associated with the postulated accident scenario.

Source term calculations were determined using the ORIGEN2 computer code. The ORIGEN2 computer code was designed for reactor fuel cycle mass and radioactivity inventory calculations. The ORIGEN2 computer code is widely recognized for calculating fission product inventories.

Offsite χ/Qs were calculated with the PAVAN computer code (Reference 27) using the guidance of NRC Regulatory Guide 1.145 (Reference 28); control room χ/Qs were calculated with the ARCON96 (Reference 29) and PAVAN computer codes consistent with the guidance if Regulatory Guide 1.194 (Reference 30). The PAVAN and ARCON96 codes are generally used to calculate relative concentrations in plumes from nuclear power plants at offsite locations and control room air intakes, respectively.

The RADTRAD computer code was used for determining dose consequences for the FHA. The RADTRAD program is a radiological consequence analysis code used to estimate post-accident doses at plant offsite locations and in the control room.

15.7.2.5.4 Key Plant Assumptions

Bounding fuel assembly:	7X7 49 pin bundle
Bounding fuel damage:	111 pins
Peaking Factor:	1.7
Secondary containment isolation/filtration:	Not credited
CREV mitigation:	Not credited
Refuel floor normal ventilation rate:	4 air changes per hour
Release point:	Reactor building exhaust stack

15.7.2.5.5 Results

The reanalysis of the FHA accident event using AST methodology is documented in calculation QDC-0000-N-1267 (Reference 31). Dose results for the main control room, EAB and LPZ are tabulated in Table 15.7-4b.

Section 15.7.2.5.6 describes the radiological assessment performed prior to the adoption of an updated accident source term in accordance with 10 CFR 50.67, “Accident Source Term.”

15.7.2.5.6 Historical Information

The description in this section is for a reactor core and fuel assemblies prior to extended power uprate (EPU). A more recent Fuel Handling Accident Analysis was performed in Reference 11 and the results are presented below in the Section titled “Meteorology and Dose Rates.” The impact of EPU and the use of Optima2 fuel on the calculated doses is addressed below in the Section titled “Meteorology and Dose Rates.”

Fission Product Release from Fuel

Fission product release estimates for the expected 89 fuel rod failures are based on the following assumptions: [15.7-8]

A. The reactor fuel has an average irradiation time of 1000 days at 2511 MWt up to 24 hours prior to fuel assembly drop;

B. A maximum of 1% of the noble gas activity and a maximum of 0.5% of the halogen activity is in the fuel rod plenums. Negligible solid or particulate activity would be released from the fuel and any such release would be absorbed in the reactor pool water; and

C. The peaking factor used to get the curie content per rod is 1.0.

The quantities of fission products calculated to be released from the failed fuel to the water from the initial core analysis of 111 fuel rod failures for 7x7 fuel is bounding and are:

Fission Product	Amount Released (curies) from Fuel
Noble gases (Xe, Kr)	5.7×10^3
Halogens (Br, I)	3.5×10^3

Fission Product Inventory in the Reactor Building

All of the noble gas fission products are assumed to be released from the reactor water to the reactor building.

The halogens released are assumed to be absorbed in the pool. They are assumed to be evolved from the pool into the air to establish an equilibrium partition factor. At the halogen concentration in the water (10^{-7}gm mol/l) for the refueling accident the partition factor has been evaluated in three different studies as 1×10^4 [1], 2×10^3 [2], and 1×10^4 [3]. In the analysis of this accident the partition factor is conservatively assumed to be 1×10^2.

Halogen fission products would also fall out and plate out in the reactor building, but additional halogens would be evolved from the reactor water to maintain equilibrium concentration in the reactor building air. If a true equilibrium condition were established as is assumed in this analysis, there would be little fallout or plateout of the noble gas fission products.

Based on the preceding assumptions and a discharge rate of 100% of building volume per day through the standby gas treatment system (SBGTS), the calculated building fission product inventory as a function of time following the accident is shown in Table 15.7-1. These results are from the initial core analysis which assumed a 7x7 fuel assembly configuration and 111 fuel rod failures.

Chimney Release Rate

The standby gas treatment system is actuated automatically on high area radiation in the reactor building in order to control the release of fission products to the atmosphere. Monitors are located near the fuel pool, and the SBGTS would be initiated prior to the escape of fission products through the regular ventilation system. While the SBGTS is in operation, it will produce a reduced pressure of 0.25 in H_2O in the reactor building while discharging at a minimum one building volume per day through the filters to the elevated release point. The amount of fission products that are released to the atmosphere is calculated assuming that the efficiency of the filters is only 99%. The activity release rate for the initial core analysis (111 fuel rod failures, 7x7 fuel) is shown in Table 15.7-2.

An analysis performed by the AEC[4] using a 90% SBGTS efficiency, also results in calculated radiological doses well within the guideline values given in 10 CFR 100. This 90% SBGTS efficiency is bounded by the requirements in Technical Specifications. [15.7-9]

Meteorology and Dose Rates

Table 15.7-3 summarizes the calculated radiological effects for the following six meteorological conditions: Very stable conditions with a 2-mph wind speed (VS-2), moderately stable condition with a 2-mph wind speed (MS-2), neutral stability conditions with 2 and 10-mph wind speeds (N-2 and N-10), and unstable conditions with 2 and 10-mph wind speeds (U-2 and U-10). The results given are for the initial core.

Radiological Reassessment

A radiological reassessment using 34 feet for a drop height was performed by GE (Reference 6). This reassessment compared the number of fuel rod failures assuming a fuel assembly and the refueling mast falls.

	GE 9x9, GE11	ATRIUM-9B	GE8	GE9 & GE10	GE14
Fuel rods per assembly	74	72	62	60	87.333

Per Reference 6, the radiological consequences for 8x8 fuel will be less than 84% of those values presented in Table 15.7.3 for a 7x7 core.

Per Reference 6, the radiological consequences for GE11 fuel will be less than 84% of those values presented in Table 15.7.3 for a 7x7 core.

The ATRIUM-9B assembly is in a 9x9 configuration with 72 fuel rods per assembly. For the purposes of this evaluation, it is conservatively assumed that the fractional plenum activity for any 9x9 rod will be 49/72, or 0.68 times the activity in a 7x7 rod. Based on the assumption that 132 9x9 rods fail compared to 111 for a 7x7 core, the relative amount of activity released for ATRIUM-9B is (132/111) (0.68)=0.81 times the activity released for a 7x7 core. In other words, the radiological consequences for ATRIUM-9B will be less than 81% of those values presented in the Table 15.7.3 for a 7x7 core.

The GE14 assembly is in a 10x10 configuration with 78 full length rods and 14 part length rods or 87.333 rods per assembly. For the purposes of this evaluation, it is conservatively assumed that the fractional plenum activity for a GE14 assembly will be 49/87.333, or 0.561 times the activity in a 7x7 assembly. Based on the assumption that 176 rods in a GE14 core fail compared to 111 for a 7x7 core, the relative amount of activity released for GE14 is (176/111)(0.561)=0.89 times the activity released for a 7x7 core (Reference 16). In other words, the radiological consequences for GE14 will be less than 89% of those values presented in Table 15.7.3 for a 7x7 core.

Mixed Fuel Types

The results of an SPC assembly dropping on a GE assembly or visa versa are bounded by the results of the above analyses due to the similarity in assembly weights, height and materials.

Radiological Effects of a Refueling Accident in the Spent Fuel Pool on Containment

A radiological reassessment [9] has been performed to determine the effect of the minimum refueling pool water depth allowed by Technical Specifications on radiological consequences following a fuel handling accident over a spent fuel storage pool or over the reactor core. The radiological doses to the control room operator and a person located at the exclusion area boundary and low population zone boundary were calculated for a normal water level of 23 feet and a minimum level of 19 feet above the spent fuel.

The methodology used to calculate the doses is consistent with the assumptions and methodologies presented in Regulatory Guide 1.25[10] and Standard Review Plan 15.7.4[8]. The code library data used the conservative inhalation dose conversion factors that were in effect and used for the 10 CFR 100 type reactor siting analyses, namely TID-14844[11] and ICRP Publication 2[12]. The refueling pool decontamination factor was calculated using a methodology developed by G. Burley[13], which is referenced in SRP 15.7.4. The release source term was calculated using the methodologies of TID-14844 and Regulatory Guide 1.25. Additionally, the source term for the refueling accident in the spent fuel pool was conservatively based on 111 failed rods, consistent with the amount of fuel failure described in Section 15.7.2.5.1 and the 1971 SER for a fuel bundle drop over the reactor core. The analysis is applicable to a fuel handling accident inside containment or in the spent fuel pool.

Table 15.7-4 lists the calculated doses and applicable regulatory limits for the control room operator and a person located at the exclusion area boundary and low population zone boundary following a fuel handling accident with a minimum refueling water depth of 19 feet above the spent fuel. A design basis fuel handling accident with the minimum refueling pool water level allowed by Technical Specifications will have consequences below the regulatory limits.

At the uprated power of 3016 MWth (including 2% instrument error) and with GE14 fuel (87.33 effective full length rods per assembly), the number of damaged fuel rods was estimated to be 176 rods. The equivalent number of damaged fuel bundles (176/87.33, or 2.02 bundles) is bounded by that based on the conservatively assumed 111 failed rods in 7x7 fuels (111/49, or 2.26 bundles) used in the pre-EPU analysis. The extended power uprate will increase the halogen and noble gas inventory in the spent fuel. The calculated thyroid, whole body, and beta doses listed in Table 15.7-4 are expected to increase by 27%, 18% and 18% respectively. The post-EPU doses are noted in Table 15.7-4a and are below the regulatory limits.

Westinghouse SVEA-96 Optima2 fuel has different core radionuclide inventories than GE fuel, with some isotopes released in either higher or lower quantities when evaluated on an equivalent basis. An evaluation of the post-FHA doses for Optima2 fuel shows that the existing EPU doses are bounded, maintained, or increased slightly [17]. The Optima2 doses are shown in Table 15.7-4a.

[END OF HISTORICAL INFORMATION]

15.7.3 Spent Fuel Cask Drop Accidents

The Spent Fuel Cask Drop scenario has been reanalyzed and is not considered a credible "Design Basis Accident". This is documented in a letter from Dennis L. Ziemann, Chief Operating Reactors Branch #2, to Mr. R. L. Bolger, Assistant Vice President, Commonwealth Edison, dated January 27, 1977. Enclosure 2 to this letter is the safety evaluation prepared by the Office of Nuclear Reactor Regulation supporting approval to facility modifications to reduce the probability of a fuel cask drop accident to an acceptably low level. This evaluation states that modifications to the Reactor Building crane were made to preclude drops of a 100-ton spent fuel shipping cask. The modifications to the crane combined with administrative controls prevent all postulated single component failures. Therefore, the Spent Fuel Cask Drop Accident is no longer considered a credible "Design Basis Accident". [15.7-10]

[Historical Information]

The following discussion reflects the results of the original cask drop evaluation. This evaluation was completed prior to the reactor building overhead crane being classified as single-failure-proof.

See the introduction to Section 15.7 for information regarding use of details from this analysis description which may not be applicable to the current fuel cycle.

Analyses have been made to determine the damage effects on the fuel pool should the fuel cask be dropped from the maximum height that the cask could be raised above the pool. The results are summarized at the end of this section. [15.7-10a]

The fuel pool design is such that major leakage due to an accident should never occur. The heaviest object to be handled near the pool is the fuel cask. This cask weighs about 85 tons; however, the analysis is based on a 100-ton cask load, a drop from the maximum height that the cask could be raised above the pool, and a minimum water level (33 feet) at the time of the drop.

Calculations show that the cask would have a maximum velocity at impact with the pool bottom of 41 ft/s. It is estimated that limited cask deterioration would occur. The cask utilized for this analysis has 8-inch radial steel fins at its base. These fins would be displaced about 6 inches under impact. In addition, limited damage would occur to the concrete from the impacting cask. It is estimated that the localized penetration of the cask into the concrete pool bottom would be 4 inches.

The cask design considered in the analysis utilizes bottom radial fins. These fins project approximately 8 inches from the cask and are 1 inch in width. The total linear length of the fins is approximately 800 inches. This provides a total bearing area of 800 square inches. [15.7-11]

The impact factor calculated in the following paragraphs is 31 g. The resulting compressions in the fins would be (200,000 x 31 / 800) psi, or approximately 8000 psi.

The penetration of the cask fins into the concrete can be obtained by use of the Modified Petry Formula[5]:

$$D = K_o A_p \log_{10}\left(1 + \frac{v^2}{215{,}000}\right)$$

where: D = Depth of penetration into an infinite thickness (feet)

K_o = A coefficient, 0.00426, for reinforced concrete used at Quad Cities

v = Velocity of missile (ft/s) = 41 ft/s

A_p = Weight of missile per ft^2 of projected area (lb/ft^2)

$$A_p = \frac{200{,}000}{800} \text{ x } 144 = 36{,}000 \text{ lb/ft}^2$$

Substituting in the above formula provides the result of approximately 6 inches penetration without fin deformation. On this basis, the assumed total deformation of 10 inches due to penetration of the concrete and crushing of the fins is reasonable (6 inches concrete, 4 inches fin).

The calculations for the impact factor are given below:

To quantify the analysis it is necessary to assume uniform deceleration. [15.7-12]

$$a = \frac{u^2}{2s}$$

where

$$\begin{aligned} a &= \text{Average deceleration} \\ s &= \text{Stopping velocity} \\ u &= \text{Impact velocity} \end{aligned}$$

$$a = \frac{(41)^2}{(2)(10/12)} = 1010 \text{ ft/s}^2 \text{ (approximately 31g)}$$

The equivalent static loading (P_E) can now be calculated to verify the adequacy of the structure:

$$P_E = (200)(31) = 6200 \text{ kips}$$

To calculate slab shear (An explanation of terms and definitions can be found in the Uniform Building Code, UBC-2617h.), If:

A = Diameter of loaded area = diameter of cask = 60 inches, and

b = Diameter of "critical" section = A + d = 122 inches,

d = Distance from extreme compression fiber to centroid of tension reinforcement = 62 inches

then

$$\nu_u = \frac{V_u}{b_o d}$$

where

$$V_u = \text{Total shear force} = 6200 \text{ kips}$$
$$b_o = \text{Periphery of critical section} = \pi \text{ x } 122$$
$$d = 62 \text{ inches}$$

and

$$\nu_u = 261 \text{ psi} = \text{Applied shear}.$$

Since

$$\nu_c = 4\,\phi\,\sqrt{f'_c}$$
$$= 4 \text{ x } 0.85\sqrt{4000} \text{ psi}$$

then

$$\nu_c = 215 \text{ psi} = \text{Check shear.}$$

With complete web reinforcement, v_c = 322 psi. Although the pool design was never intended to qualify for v_c = 322 psi, some reinforcement is provided and it can be concluded that the 6-foot 3-inch thick concrete floor would not catastrophically fail.

Although these calculations show that the cask would not penetrate the pool floor, the reinforcing steel would locally approach yield stress. On the assumption that local areas of the steel are at 40,000 psi, an estimate of leakage paths can be made:

$$\text{Unit Strain} = \frac{40{,}000}{30 \text{ x } 10^6} = 0.0013 \text{ in / in}$$

In evaluating the capability of the pool floor to retain its integrity under the postulated cask drop, assumptions have been made with respect to structural behavior. The principal assumption involves the manner in which the cask load is taken by the pool floor acting as a beam-type structure. The cask loading position is adjacent to one wall of the pool. For purposes of flexural loading, it is assumed that the cask load is divided between an assumed cross beam and a strip parallel to the pool wall. With the 7-foot-diameter, 100-ton cask, 20% of the load is taken on a beam 7 feet wide and 41-feet long with the remainder taken directly into the supporting wall. [15.7-13]

The load is concentrated 4 feet from one end and end conditions are assumed fixed. The maximum moment in such a beam is the end moment nearest the load; however, any rotation of the supports would reduce this moment.

The ultimate strength design methods outlined in Chapter 26 of the Uniform Building Code were used to check the design. The results of this analysis show a maximum stress in the steel of approximately 40,000 psi. This is the basis for the preceding discussion relating to strain and consequent cracking.

It has been shown that the structural elements resisting the load will not fail in a catastrophic fashion. With respect to the energy absorption characteristics of the resisting system, it should be recognized that most of the energy would be taken in the elastic response of the major interacting structural elements. Relatively small amounts of energy are absorbed in the work of damaging the cask and the destruction of a small amount of concrete in the impact zone.

If a 5-foot square area of the floor slab is subjected to this strain, and if the cracking is assumed to pass completely through the slab, then the flow area is: [15.7-14]

$$A = 60 \text{ in} \times 0.0013 \times 60 \text{ in} = 5 \text{ in}^2$$

It should be noted that there is no mechanism to provide gross slab straining completely through the floor. Also, these crack conditions exist only for the time during which the slab is subjected to the dynamic loading of the cask—less than 1 second. Even if it is assumed that the residual crack area is as great as one-fourth of the above estimate (0.25 x 5 = 1.25 in^2), the resulting leakage would be limited to:

$$\text{Leakage } q = CA\sqrt{2\,gh}$$

where:

$$C = \text{discharge coefficient} = 0.25$$

$$q = 0.25 \times \frac{1.25}{144}\sqrt{(2)\,(32.2)\,(38)}$$

$$q = 0.11 \text{ ft}^3/\text{s} = 50 \text{ gal/min}$$

The choice of a discharge coefficient is a matter of judgment. The cracks are assumed to pass uniformly through the slab. It should be noted that there is no mechanism to provide gross cracking completely through the floor. This mode of cracking is inconsistent with structural conditions existing in the slab and is very conservative. The paths would be rough and distorted. Under these conditions, 0.25 is believed to be adequately conservative. However, even if a coefficient of 1.0 were chosen, the estimated leakage would be 200 gal/min, still well within the capacity (275 gal/min each) of one of three condensate transfer pumps available.

In summary, the analysis results in an estimate on the order of 10 to 80 gal/min leakage rate through crack paths that could develop as a result of the above postulated accident. The water would leak onto the floor beneath the pool and subsequently to the reactor building floor drain sumps. The sump capacity and the normal makeup capability are both greater than this calculated leakage. Depending upon the plant operating conditions at the time a leak is postulated to develop, there are various methods of supplying makeup water to the pool to prevent the pool level from decreasing to an unsafe level above the fuel. The condensate transfer system is normally used to supply makeup water to the pool. There are three condensate transfer pumps serving both units and under normal conditions, one pump can supply all the necessary plant makeup requirements. Should a circumstance occur which requires more than the capacity of one pump (275 gal/min) the other pumps can be started. This provides up to 825 gal/min makeup capability, which is in excess of any leakage that could conceivably occur.

In conclusion, the energy absorption capability of the cask housing, fins, and concrete fuel pool is such that the expected damage to the floor would not result in a leakage rate greater than the pool makeup capability. No energy absorbing system is being investigated or is considered necessary. [15.7-15]

[End of Historical Information]

15.7.4 Radioactive Gas Waste System Leak or Failure

When the Off-Gas system was modified at Quad Cities and Dresden, an accident analysis was performed for the hypothetical failure of each of the three following major components[14]: the charcoal beds, the prefilter, or the holdup piping. This accident analysis was based upon the follow parameters:

A. Reactor rated at 2527 MWt (Dresden);
B. 18.5 standard ft^3/min air inleakage;
C. 100,000 μCi/s diffusion bed gas mixture after a 30-minute delay; and
D. 12 active carbon beds – 74,000 pounds of activated carbon.

The analysis assumes the following equipment characteristics with respect to the retention of daughter products prior to the failure of the off-gas equipment:

A. Off-gas condenser – 100% but washed out;
B. Water separator – 100% but washed out;
C. Holdup pipe – 60% but washed out;
D. Prefilter – 100%;
E. Activated carbon beds – 100%; and
F. Postfilter – 100%

To provide an estimate of hypothetical radiological doses from off-gas system equipment failures, certain percentages of the activity contained in the most significant off-gas system components are assumed to be released to the environment under very stable 1 m/s meteorological conditions with an effective release height of zero meters. Percentages of primary activities released from components and the corresponding estimated radiological exposures are presented in Table 15.7-5 and are compared to the limits in the Standard Review Plant (SRP) and 10 CFR 100.

Additional analysis[15] was performed to evaluate the effects of an increase of 3.5 times in the radioactive inventory after holdup (350,000 μCi/s versus 100,000 μCi/s) to bound the off-gas activity allowed by plant Technical Specifications. The resulting estimated radiological exposures from this analysis are also presented in Table 15.7-5 and remain less than the original limits of both the SRP and 10 CFR 100.

This accident analysis indicates there is no undue hazard to the health and safety of the public resulting from installation and operation of the off-gas treatment system. The failure of equipment in the off-gas treatment system would result in only a fraction of the offsite doses presently permitted.

15.7.5 References

l. Miller, et al., "International Symposium on Fission Product Release and Transport Under Accident Conditions." Paper 12, Oak Ridge, Tennessee, April 1955.

2. Allen, T. L., and Keefer, R. M., "The Formation of Hypoidus Acid and Hydrated Iodine Cation by the Hydrolysis of Iodine," JACS 77, No. 11, June 1955.

3. Watson, Bancroft and Hoelke, "Iodine Containment by Dousing in NPD-11," AECL-1130, 1960.

4. U.S. Atomic Energy Commission, Division of Reactor Licensing, "Safety Evaluation for Quad Cities Station, Units 1 and 2," Docket Nos. 50-254 and 50-265, August 25, 1971.

5. Russell, C. R., Reactor Safeguards, The MacMillian Co., New York, N.Y., 1962

6. NEDE-24011-P-A, "GESTAR II - General Electric Standard Application for Reactor Fuel," Revision 14, June 2000.

7. EMF-96-173(P), "Dresden/Quad Cities Fuel Handling Accident Analysis for ATRIUM-9B Fuel," October 1996, NFS NDIT 9600173.

8. USNRC Standard Review Plan 15.7.4, "Radiological Consequences of Fuel Handling Accidents," Revision 1, July 1981.

9. Calculation No. QDC-9400-M-0908, Rev. 0, Quad Cities Units 1 and 2, Effect of Reduced Pool water Level on FHA Radiological Consequences.

10. Regulatory Guide 1.25, Assumptions Used for Evaluating the Potential Radiological Consequences of Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors, Revision 0, March 23, 1972.

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

12. ICRP Publication 2, “Report of Committee II, Permissable Dose for Internal Radiation,.” 1959.

13. G. Burley, “Evaluation of Fission Product Release and Transport for Fuel handling Accident,” Radiological Safety Branch, Division of Reactor Licensing, October 5, 1971.

14. Quad Cities Units 1 and 2 Special Report No. 1, Modified Off-Gas System.

15. Letter E-DAS-00-048 from D. A. Studley (Scientech) to Robert Tsai (ComEd) “submittal of Calculation in Support of Improved Tech. Spec. Program,” dated February 17, 2000.

16. Calculation Number BNDQ: 01-012, Revision 0, “Fuel Handling Accident at Quad Cities for 8x8, 9x9, and 10x10 Fuel,” 1/13/2002.

17. Calculation QDC-000-N-1020, "Impact of Extended Power Uprate on Site Boundary and Control Room Doses from LOCA and Non-LOCA Events," Revision 001A, Attachment E, "Evaluation for Impact of Westinghouse Optima2 Fuel on Existing UFSAR Design Basis Accidents."

18. U. S. Atomic Energy Commission, Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," 1962.

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

20. Letter from M. Banerjee (U. S. NRC) to C. Crane (Exelon Corporation), Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2 - Issuance of Amendments Re: Adoption of Alternative Source Term Methodology,” dated September 11, 2006 [SER correction letter: D. Collins (U. S. NRC) to C. Crane (Exelon Corporation), “Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2 – Correction of Safety Evaluation for Amendment Dated September 11, 2006,” dated September 28, 2006].

21. NRC Regulatory Guide 1.49, “Power Levels of Nuclear Power Plants,” December 1973.

22. ORNL/TM-7175, “A Users’ Manual for the ORIGEN2 Computer Code,” A. G. Croff, July 1980.

23. GE Task Report No. GE-NE-A22-00103-64-01, Rev. 0, Project Task Report T0802, “Dresden and Quad Cities Asset Enhancement Program - Radiation Sources and Fission Products,” August 2000.

24. NUREG/CR-6604, “RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation,” April 1998 and Supplement 1, June 1999.

25. U.S. Federal Guidance Report No. 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” 1988.

26. U.S. Federal Guidance Report No. 12, “External Exposure to Radionuclides in Air, Water, and Soil,” 1993.

27. Atmospheric Dispersion Code System for Evaluating Accidental Radioactivity Releases from Nuclear Power Stations; PAVAN, Version 2; Oak Ridge National Laboratory; U.S. Nuclear Regulatory Commission; December 1997.

28. NRC Regulatory Guide 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” February 1983.

29. Atmospheric Relative Concentrations in Building Wakes; NUREG/CR-6331, PNNL-10521, Rev. 1; prepared by J. V. Ramsdell, Jr., C. A. Simmons, Pacific Northwest National Laboratory; prepared for U.S. Nuclear Regulatory Commission; May 1997 (Errata, July 1997).

30. NRC Regulatory Guide 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” June 2003.

31. QDC-0000-N-1267, “Re-analysis of Fuel Handling Accident (FHA) Using Alternate Source Term.”

32. Westinghouse Topical Report CENPD-300-P-A (Proprietary), CENPD-300-NP-A (Non-Proprietary), “Reference Safety Report for Boiling Water Reactor Reload Fuel,” July 1996.

Table 15.7-1

REACTOR BUILDING AIRBORNE FISSION PRODUCT INVENTORY
(Original analysis, retained for historical purposes)

Time After Accident	Noble Gases (curies/s)	Halogens (curies/s)
1 minute	5.6×10^{3}	5.1×10^{2}
30 minutes	5.4×10^{3}	4.7×10^{2}
1 hour	5.2×10^{3}	4.5×10^{2}
2 hours	4.8×10^{3}	4.0×10^{2}
12 hours	2.7×10^{3}	2.3×10^{2}
15 hours	2.3×10^{3}	2.1×10^{2}
1 day	1.5×10^{3}	1.8×10^{2}
2 days	4.5×10^{2}	1.2×10^{2}
10 days	5.3×10^{-2}	1.6×10^{1}
25 days	2.6×10^{-9}	5.0×10^{-1}

Table 15.7-2

REFUELING ACCIDENT
FISSION PRODUCT RELEASE RATE FROM CHIMNEY
(Original analysis, retained for historical purposes)

Time After Accident	Noble Gases (curies)	Halogens (curies)
1 minute	6.5×10^{-2}	5.9×10^{-5}
30 minutes	6.2×10^{-2}	5.5×10^{-5}
1 hour	6.0×10^{-2}	5.2×10^{-5}
2 hours	5.6×10^{-2}	4.6×10^{-5}
12 hours	3.2×10^{-2}	2.7×10^{-5}
15 hours	2.7×10^{-2}	2.5×10^{-5}
1 day	1.7×10^{-2}	2.0×10^{-5}
2 days	5.3×10^{-3}	1.4×10^{-5}
10 days	6.2×10^{-7}	1.8×10^{-6}
25 days	0	5.8×10^{-8}

Table 15.7-3

RADIOLOGICAL EFFECTS OF THE REFUELING ACCIDENT
(Original analysis, retained for historical purposes)

	First 2-Hour Dose						Total Dose					
Distance (Miles)	VS-2	MS-2	N-2	N-10	U-2	U-10	VS-2	MS-2	N-2	N-10	U-2	U-10
	Passing Cloud Whole Body Dose (rem)											
1/4	8.3×10^{-4}	8.2×10^{-4}	8.3×10^{-4}	1.4×10^{-4}	1.2×10^{-3}	1.8×10^{-4}	4.0×10^{-3}	4.0×10^{-3}	4.1×10^{-3}	7.0×10^{-4}	5.9×10^{-3}	8.8×10^{-4}
1	4.8×10^{-4}	4.8×10^{-4}	5.9×10^{-4}	8.7×10^{-5}	4.4×10^{-4}	7.0×10^{-5}	2.4×10^{-4}	2.4×10^{-3}	2.9×10^{-3}	4.3×10^{-4}	2.2×10^{-3}	3.4×10^{-4}
3	2.2×10^{-4}	2.3×10^{-4}	2.1×10^{-4}	3.4×10^{-5}	8.4×10^{-5}	1.7×10^{-5}	1.1×10^{-3}	1.1×10^{-3}	1.1×10^{-3}	1.7×10^{-4}	4.1×10^{-4}	8.3×10^{-5}
5	1.4×10^{-4}	1.5×10^{-4}	9.7×10^{-5}	1.8×10^{-5}	3.4×10^{-5}	8.0×10^{-6}	6.7×10^{-4}	7.3×10^{-4}	4.7×10^{-4}	8.8×10^{-5}	1.6×10^{-4}	3.9×10^{-5}
10	6.8×10^{-5}	7.2×10^{-5}	2.8×10^{-5}	6.8×10^{-6}	8.6×10^{-6}	2.8×10^{-6}	3.3×10^{-4}	3.5×10^{-4}	1.4×10^{-4}	3.3×10^{-5}	4.2×10^{-5}	1.4×10^{-5}
	Lifetime Thyroid Dose (rem)											
1/4	*	*	1.5×10^{-7}	*	6.1×10^{-4}	3.8×10^{-5}	*	2.8×10^{-10}	1.0×10^{-6}	2.5×10^{-10}	4.1×10^{-3}	2.6×10^{-4}
1	*	1.9×10^{-6}	3.4×10^{-4}	3.1×10^{-5}	3.2×10^{-4}	4.3×10^{-5}	*	1.3×10^{-5}	2.3×10^{-3}	2.1×10^{-4}	2.1×10^{-3}	2.9×10^{-4}
3	*	4.4×10^{-5}	1.2×10^{-4}	2.1×10^{-5}	5.5×10^{-5}	9.3×10^{-6}	*	2.9×10^{-4}	8.2×10^{-4}	1.4×10^{-4}	3.7×10^{-4}	5.2×10^{-5}
5	*	6.7×10^{-5}	5.9×10^{-5}	1.1×10^{-5}	2.4×10^{-5}	4.4×10^{-6}	*	4.5×10^{-4}	4.0×10^{-4}	7.5×10^{-5}	1.6×10^{-4}	3.0×10^{-5}
10	1.8×10^{-8}	6.7×10^{-5}	2.2×10^{-5}	4.4×10^{-6}	8.5×10^{-6}	1.6×10^{-6}	1.2×10^{-7}	4.5×10^{-4}	1.5×10^{-4}	3.3×10^{-6}	5.7×10^{-5}	1.1×10^{-5}

(Sheet 1 of 2)

Table 15.7-3

RADIOLOGICAL EFFECTS OF THE REFUELING ACCIDENT
(Original analysis, retained for historical purposes)

	First 2-Hour Dose						Total Dose					
	VS-2	MS-2	N-2	N-10	U-2	U-10	VS-2	MS-2	N-2	N-10	U-2	U-10
Whole Body Fallout Dose (rem)												
1/4	*	*	*	*	6.0×10^{-7}	1.9×10^{-7}	*	*	1.2×10^{-9}	*	1.0×10^{-5}	3.5×10^{-6}
1	*	4.6×10^{-10}	1.5×10^{-7}	6.6×10^{-8}	3.1×10^{-7}	2.1×10^{-7}	*	8.5×10^{-9}	2.8×10^{-6}	1.2×10^{-6}	5.8×10^{-6}	3.9×10^{-6}
3	*	1.1×10^{-8}	5.3×10^{-8}	4.5×10^{-8}	5.3×10^{-8}	4.5×10^{-8}	*	2.0×10^{-7}	9.9×10^{-7}	8.5×10^{-7}	1.0×10^{-7}	8.5×10^{-7}
5	*	1.7×10^{-8}	2.6×10^{-8}	2.4×10^{-8}	2.4×10^{-8}	2.1×10^{-8}	*	3.1×10^{-7}	4.8×10^{-7}	4.5×10^{-7}	4.5×10^{-7}	4.0×10^{-7}
10	*	1.7×10^{-8}	9.3×10^{-9}	9.4×10^{-9}	8.2×10^{-9}	7.7×10^{-9}	*	3.1×10^{-7}	1.7×10^{-7}	1.8×10^{-7}	1.5×10^{-7}	1.5×10^{-7}

	Meteorology	Wind Speed (mph)
VS-2	Very stable	2
MS-2	Moderately stable	2
N-2	Neutral	2
N-10	Neutral	10
U-2	Unstable	2
U-10	Unstable	10

* Denotes less than 10^{-10}.

Table 15.7-4

RADIOLOGICAL EFFECTS OF THE REFUELING ACCIDENT IN THE SPENT FUEL POOL OR CONTAINMENT
Pre-Uprate Conditions

Summary of Calculated Doses in the Control Room, Low Population Zone, and Exclusion Area Boundary following a postulated Fuel Handling Accident with the Spent Fuel Pool Water at a minimum level of 19 feet.

Location		Thyroid Dose (Rem)	Whole Body Dose (Rem)	Beta Dose (Rem)
Control Room	Calculated Value	7.66	1.20E-2	0.462
	SRP 6.4 Limit	30	5	30
Low Population Zone	Calculated Value	0.687	3.80E-2	9.60E-2
	SRP 15.7.4 Limit	75	6	NA
Exclusion Area Boundary	Calculated Value	9.92	0.358	0.900
	SRP 15.7.4 Limit	75	6	NA

(This table is retained for historical purposes)

Table 15.7-4a

Fuel Handling Accident in the Spent Fuel Pool or in Containment
EAB, LPZ, and Control Room Doses Following EPU

(Historical Information)

Location	Organ	Non-Optima2 Dose (Rem)	Optima2 Dose (Rem) *	Regulatory Dose Limit (Rem)
EAB	Thyroid	12.6	12.6	750
	Whole Body	0.42	0.44	6.25
LPZ	Thyroid	0.87	0.87	75
	Whole Body	0.045	0.047	6.25
Control Room	Thyroid	9.73	9.68	30
	Whole Body	0.014	0.014	5
	Beta	0.55	0.55	30

* Optima2 dose per QDC-0000-N-1020, Revision 001A (Reference 17 of Section 15.7).

Table 15.7-4b

Fuel Handling Accident in the Spent Fuel Pool or in Containment
EAB, LPZ, and Control Room Doses
(Alternative Source Term)

Location	Case 1 (23 feet water coverage) Dose (rem TEDE)	Case 2 (19 feet water coverage) Dose (rem TEDE)
Limits	CR 5.0; EAB&LPZ 6.3	
EAB	3.84	5.24
LPZ	0.294	0.401
CR	1.22	1.80

Table 15.7-5

RADIOLOGICAL EFFECTS OF AN
OFF-GAS SYSTEM COMPONENT FAILURE

(This assessment has not been updated using Alternative Source Term)

Component Failed	Primary Activity Released	Percent Released	Resultant Exposure for 100,000 mCi/s (mR) [Ref. 14]	Resultant Exposure for 350,000 mCi/s (mR) [Ref.15]	SRP[1] Limits (mR)	10 CFR 100 Limits (mR)
First Carbon Bed	Iodine	1	17.4	60.9	500	300,000
12 Carbon Beds	Noble Gas	10	1.77	6.20	500	25,000
Prefilter	Particulates	1	48.0	168.0	500	25,000
Holdup Pipe	Particulates	20	53.0	185.5	500	25,000

1. Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, Postulated Radioactive Releases Due to Waste Gas System Leak or Failure.

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

This section covers the events, which result in an anticipated transient without scram (ATWS). Anticipated transient without scram events are beyond design basis accidents. Anticipated transients without scram are those low probability events in which an anticipated transient occurs and is not followed by an automatic reactor shutdown (scram) when required. The failure of the reactor to scram quickly during these transients can lead to unacceptable reactor coolant system pressures and to fuel damage. Mitigation of the lack of scram must involve insertion of negative reactivity into the reactor, thereby terminating the long-term aspects of the event.

The occurrence of a common-mode failure, which completely disables the reactor scram function, is a very low probability event. Therefore, no significant risk to public safety is presented by the combination of an infrequent event and a common-mode failure, which prevents scram. Thus, attention is focused on those transient situations, which have a relatively high expected frequency of occurrence at a power condition at which serious plant disturbance might result.

GE performed the ATWS analysis for operation at Extended Power Uprate (EPU, at 2957 MWt). The GE analysis shows that a Pressure Regulator Failure Open to maximum steam demand (PRFO) is the limiting ATWS event. Westinghouse demonstrated that PRFO remains limiting with the ATWS analysis performed to support transition to the SVEA-96 Optima2 fuel.

Westinghouse performed a plant-unique ATWS analysis as part of the fuel licensing analysis supporting the introduction of SVEA-96 Optima2 fuel. The analysis was performed for two equilibrium cores (SVEA-96 Optima2 and GE14) using the same cycle design specifications at EPU conditions and for a transition core with 2/3 GE14 fuel and 1/3 SVEA-96 Optima2 fuel. The results of the analysis are reported in Reference 6.

Additional calculations for the long-term ATWS were also performed and reported in Reference 6. Long-term effects refer to the containment response and the general plant response with emphasis on the heat transfer to the suppression pool. The time delay from the symptom to initiating operator action credited for the SVEA-96 Optima2 fuel transition are equal to or conservatively longer than the EPU licensing analysis of long-term ATWS. The results confirm that for the limiting pressure regulator failure open to maximum demand (PRFO) all acceptance criteria are met after the introduction of SVEA-96 Optima2 fuel.

Westinghouse has performed the limiting ATWS analysis for PRFO using the bounding ATWS inputs for Quad Cities. Westinghouse has analyzed the limiting ATWS event for the plant changes including acoustic side branch piping modification and change to 45 a/o enriched B-10 and 40 gpm SLC pump flow rate while considering applicable core design configurations. Westinghouse confirms the peak vessel bottom pressure for PRFO on a cycle-specific basis. Westinghouse ATWS analysis for Quad Cities has addressed all ATWS acceptance criteria. Following specific criteria are satisfied:

- Peak vessel bottom pressure – The Westinghouse calculation in Reference 6 considers the effect of core design changes and has established the licensing basis for the reload of Westinghouse fuel, which replaces the GE ATWS analysis in Reference 4.

- Peak cladding temperature (PCT) - Westinghouse has confirmed in Reference 6 that the PCT for the limiting ATWS event is bounded by the LOCA analysis for the Westinghouse fuel.

- Peak cladding oxidation - Westinghouse has confirmed in Reference 6 that the peak cladding oxidation for the limiting ATWS event is bounded by the LOCA analysis for the Westinghouse fuel.

- Peak suppression pool temperature - The Westinghouse calculation in Reference 6 is cycle independent and provides new bases that represent current plant configuration and replaces the GE ATWS analysis in Reference 4.

- Peak containment pressure - The Westinghouse calculation in Reference 6 is cycle independent and provides new bases that represent current plant configuration and replaces the GE ATWS analysis in Reference 4.

The PRFO would be the most severe postulated event from virtually all aspects when accompanied by a lack of scram. Other significant ATWS events, which are postulated to occur are described in the following subsections. Other transients such as closure of all main steam isolation valves (MSIVs), inadvertent opening of a relief or safety relief valve, and feedwater failure to maximum demand are less severe and are bounded by the PRFO event described in Section 15.8.6. Section 15.8.6 describes the limiting PRFO event in the Westinghouse analysis of Reference 6. Westinghouse has evaluated the impact of installation of the Adjustable Speed Drives (ASDs) for the ATWS analysis (Reference 8), and has determined that there is no adverse impact due to this plant modification.

The following ATWS events have also been analyzed and have been demonstrated to be bounded by the PRFO.

A. Closure of main steam isolation valves

B. Loss of A.C. Power

C. Loss of normal feedwater flow

D. Turbine generator trip

E. Loss of condenser vacuum

The descriptions of the above ATWS events are provided in Sections 15.8.1 through 15.8.5 and are based upon analyses [1,2], which utilized setpoints and intial conditions that differ from those currently in effect for Quad Cities.

The differences include a shorter rod insertion time than specified in the current design of the ATWS mitigation system (described in Section 7.8). The conclusions of these analyses are not completely applicable to the current plant design or fuel cycle. Specific details contained in the descriptions and associated figures should be used only to understand the analysis and its conclusions. These specific details should not be used as sources of current design information. [15.8-3]

15.8.1 Closure of Main Steam Line Isolation Valves

See the introduction to Section 15.8 for information regarding use of details from this analysis description which may not be applicable to the current plant design.

15.8.1.1 Identification of Causes

Closure of all MSIVs is caused by any of a number of plant conditions such as low-low reactor water level, main steam line high flow, main steam low pressure, or main steam tunnel high temperature. [15.8-4]

15.8.1.2 Sequence of Events and Systems Operations

This transient would be initiated with the closure of the MSIVs. Closure of the MSIVs would produce an immediate increase in reactor pressure, which would result in a reduction in moderator voids and a rapid increase in reactor power. In the absence of normal scram, the fuel temperature would rise and the negative Doppler reactivity would limit the power. The opening of relief valves would tend to curtail increase in reactor pressure and power. The reactor pressure would reach the ATWS mitigation system setpoint of 1250 psig (analytical limit) about five seconds after the start of the event. The ATWS mitigation system would initiate trip of the recirculation system pumps (RPT) and would initiate alternate rod insertion (ARI). The RPT and ARI would introduce negative reactivity into the core. The reactor pressure would peak in about 12 seconds and then decrease to a value just above the relief valve setpoints. [15.8-5]

15.8.1.3 Core and System Performance

A. Reactor Shutdown by RPT and ARI

The projected vessel pressure as a function of time for this event is shown in Figure 15.8-1. In this case, the reactor pressure would rise to the ATWS setpoint which would trip the recirculation pumps. This would cause a rapid reduction of core flow and a corresponding increase in core moderator voids which would reduce core power. The resulting neutron flux behavior is shown in Figure 15.8-2. The ATWS signal would also initiate opening of valves on the scram air header which would result in insertion of the control rods. This transient would result in a peak reactor pressure of 1476 psig, which would satisfy the ATWS analysis guideline overpressure limit of 1500 psig[2].

B. Reactor Shutdown by RPT and SBLC (No ARI)

In the event that control rod insertion (via ARI) is unavailable for shutdown of the reactor, the standby liquid control (SBLC) system would be used as an alternative method of achieving reactor shutdown. The SBLC system would be actuated manually, and would inject an aqueous solution of sodium pentaborate into the reactor vessel. When the boron concentration in the reactor coolant reaches approximately 600 ppm, sufficient negative reactivity would be available to bring the core to hot shutdown. The SBLC pumps would continue injection until sufficient boron is in the core to achieve shutdown. [15.8-6]

15.8.1.3.1 Reactor Water Level Response

The projected reactor water level response to the event with utilization of ARI is shown in Figure 15.8-3. The reactor water level would remain at near-normal level from event initiation until hot shutdown. Thereafter, adequate water inventory would be maintained either by the feedwater system or by the high-pressure coolant injection (HPCI) system. [15.8-7]

15.8.1.3.2 Containment and Suppression Pool Response

A. Reactor Shutdown by RPT and ARI

After the MSIVs are closed, the reactor power would be dissipated by the relief valve discharge of steam into the suppression pool. The steam discharged into the suppression pool would heat the suppression pool. The reactor would be shut down by ARI and steam would continue to flow into the pool due to decay heat. The operator would place the residual heat removal (RHR) system in the suppression pool cooling mode. The suppression pool temperature would continue to increase until the decay heat input decreases below the heat removal capacity of the RHR heat exchangers.

B. Reactor Shutdown by RPT and SBLC (No ARI)

Achievement of reactor shutdown using SBLC rather than ARI would result in a peak containment pressure less than the design pressure. Containment pressures and suppression pool temperatures will be higher using SBLC rather than ARI due to the greater amount of relief valve steam flow entering the suppression pool.

15.8.1.3.3 Long Term Response

For ATWS considerations, the reactor condition of concern would be hot shutdown rather than cold shutdown, because the key factor would be stopping thermal power generation during the event. The power generated prior to reaching hot shutdown has the most significant potential impact on the plant. Consequently, the time required to achieve hot shutdown would be the important parameter for ATWS. After hot shutdown is achieved, further action would be required to bring the reactor to cold shutdown conditions. [15.8-8]

15.8.1.3.4 Operator Actions

In case of an apparent ATWS, certain manual actions would be required to be performed by the operator if automatic features do not function as designed. Possible operator actions would include trip of the recirculation pump, manual initiation of ARI, and actuation of SBLC. [15.8-9]

Certain alarms and indications would be provided to the operator to support performance of the required manual actions within the time limits. Annunciator windows would alarm when the reactor water level or reactor pressure would reach the ATWS setpoints. At the beginning of the ATWS event, the recirculation pumps would be signalled to trip and the ARI would be automatically initiated, and the operator would be informed that an ATWS has occurred. The operator would then have sufficient time to perform the required actions. Operator actions that would be required in the event of an ATWS are set forth in plant procedures. Plant procedures specify that upon receipt of an automatic scram signal, if the reactor has not achieved shutdown using the control rods and reactor power is above a specified point, the operator is to actuate ARI. This action would insert the control rods. Manual RPT would be achieved by tripping the recirculation pumps.

Control room annunciators would inform the operator of ATWS trouble, ATWS channel A or B manual push button armed, and ATWS channel A or B tripped conditions.

Manual initiation of ARI and RPT is described in greater detail in Section 7.8.3.

In the event that control rod insertion is unavailable for shutting the reactor down, the SBLC system would be manually actuated to inject an aqueous solution of sodium pentaborate into the reactor vessel. [15.8-10]

Operator actions would involve actuation of the residual heat removal system to cool the suppression pool. Operator actions are also required to bring the reactor from hot shutdown to cold shutdown.

15.8.1.4 Barrier Response

During the MSIV closure transient without scram the reactor fuel would experience a rapid power spike. Since heat removal through the fuel surface would follow the relatively slow dynamics of the fuel, a significant rise in fuel enthalpy would be encountered.

The analysis of the event shows the amount of cladding oxidation (<1% by volume) would be far less than the 17% guideline (per NEDE-25026^[1]^), and peak fuel enthalpy would be less than the Regulatory Guide 1.77 limit of 280 cal/g. Few fuel rod perforations would be experienced.

15.8.1.5 Radiological Consequences

The radiological consequences would be minimal due to the small (if any) number of fuel rod perforations.

15.8.2 Loss of Normal AC Power

See the introduction to Section 15.8 for information regarding use of details from this analysis description which may not be applicable to the current plant design.

15.8.2.1 Identification of Causes

The loss of normal ac power would generally be caused by large grid disturbances which in turn would de-energize buses that supply power to auxiliary equipment such as the recirculation pumps, condensate pumps, and circulating water pumps. [15.8-11]

15.8.2.2 Sequence of Events and Systems Operations

When auxiliary power is lost, the circulating water pumps, feedwater pumps, and recirculation pumps would begin coasting down immediately. The reduction in core flow would begin to reduce the reactor power. A turbine-generator trip would occur at the start of the event due to a general grid disturbance and would contribute to the pressurization of the reactor. The safety/relief valves would open momentarily which would limit the pressure rise in the vessel. The peak vessel pressure experienced in this event would be less than in the MSIV closure event. The short-term response would be much less severe than the MSIV ATWS event for the following reasons:

A. The recirculation pumps would trip at time zero which would result in lower core flow rather than tripping when the reactor pressure reaches the ATWS mitigation system setpoint.

B. The feedwater pumps would trip at time zero which would result in reduced core inlet subcooling, and hence in a lower reactor power.

15.8.2.3 Core and System Performance

A. Reactor Shutdown by RPT and ARI

The reactor would achieve hot shutdown by utilizing ARI. The peak fuel enthalpy reached would be less than 280 cal/g. Combined flow of the reactor core isolation cooling (RCIC) and HPCI systems would restore reactor water level to the normal range. Cold shutdown would be reached by performing the normal manual actions.

B. Reactor Shutdown by RPT and SBLC (No ARI)

In the event that insertion of the control rods via ARI is not achievable, the SBLC system would be utilized as an alternative method of effecting neutronic power shutdown. The vessel water level would be restored by HPCI and RCIC. Containment pressures and suppression pool temperatures will be higher using SBLC rather than ARI due to the greater amount of steam flow entering the suppression pool.

The operator actions associated with this event would be similar to those described in Section 15.8.1.3.4.

15.8.2.4 Barrier Performance

As in the MSIV closure event, cladding oxidation would be less than 1% by volume. Peak fuel enthalpy would be less than 280 cal/g. Very few if any fuel rod perforations would be experienced.

15.8.2.5 Radiological Consequences

Radiological consequences would be minimal due to the small (if any) number of fuel rod perforations.

15.8.3 Loss of Normal Feedwater Flow

See the introduction to Section 15.8 for information regarding use of details from this analysis description which may not be applicable to the current plant design.

15.8.3.1 Identification of Causes

Inadvertent trip of all feedwater pumps or water level controller failure (zero demand) would be a potential cause for loss of all normal feedwater flow to the vessel. Loss of auxiliary power would also be a potential cause of this event. [15.8-12]

15.8.3.2 Sequence of Events and Systems Operations

The short-term effects of this event would be less severe than those of the MSIV closure event.

Reactor core flow would be reduced when the feedwater flow reduction occurs, which would drop power gradually until a low water level scram is initiated. Gradual vessel water inventory reduction would occur until vessel isolation is initiated. The RCIC/HPCI systems would be initiated automatically to maintain proper water level until the event is terminated. When the reactor water level reaches the low-low level, the ATWS logic would initiate ARI and RPT.

15.8.3.3 Core and System Performance

A. Reactor Shutdown by RPT and ARI

Hot shutdown would be achieved by using ARI. The recirculation pumps would also trip. The peak temperature reached in the pool would be slightly less than the temperature for the MSIV closure event. The peak vessel pressure experienced in this event would be less than in the MSIV closure event.

B. Reactor Shutdown by RPT and SBLC (No ARI)

In the event that insertion of the control rods via ARI is not achievable, the SBLC system would be utilized as an alternative method of effecting neutronic power shutdown.

The cold shutdown condition would be achieved by normal manual actions similar to those performed during the MSIV closure event. Without the ARI function, the plant long term response to the transient would be similar to the loss of normal ac power transient (SBLC without ARI).

The operator actions associated with this event would be similar to those described in Section 15.8.1.3.4.

15.8.3.4 Barrier Performance

This event would result in cladding oxidation of less than 1% by volume. Peak fuel enthalpy would be less than 280 cal/g. Very few if any fuel rod perforations would be experienced.

15.8.3.5 Radiological Consequences

Radiological consequences would be minimal due to the small (if any) number of fuel rod perforations.

15.8.4 Turbine-Generator Trip/Load Rejection

See the introduction to Section 15.8 for information regarding use of details from this analysis description which may not be applicable to the current plant design.

15.8.4.1 Identification of Causes

Loss of generator electrical load would initiate fast closure of the turbine control valves to provide overspeed protection for the unit. A variety of equipment protection signals would lead to trip of the turbine stop valves directly. Both the turbine control valve fast closure and turbine stop valve trip would have a similar effect on the reactor. Normally, a scram would be initiated almost simultaneously with the start of the control valve fast closure or with the stop valves starting to close. However, the scram is postulated not to occur. [15.8-13]

15.8.4.2 Sequence of Events and Systems Operations

The fast closure of the valves would cause an abrupt reactor pressure rise which would be limited to well below design pressures by the action of the bypass and the safety/relief valves.

When the dome pressure reaches the ATWS setpoint, a recirculation pump trip and alternate rod insertion would be initiated.

15.8.4.3 Core and System Performance

A. Reactor Shutdown by RPT and ARI

The neutron flux, vessel pressure, and fuel transient peaks experienced in this event would be less than those in the MSIV closure event.

The long-term response of the plant is not analyzed, as it would be similar to the MSIV closure ATWS event. However, because of the availability of turbine bypass to the condenser, the steam flow into the suppression pool would be considerably less than the MSIV closure event. Long-term heat removal would be provided by the steam bypass to main condenser. Reactor coolant inventory would be maintained using the feedwater system.

B. Reactor Shutdown by RPT and SBLC (No ARI)

In the event that insertion of the control rods via ARI is not achievable, the SBLC system would be utilized as an alternative method of effecting neutronic power shutdown. The peak suppression pool temperature will be less than the temperature reached in the MSIV closure event (SBLC without ARI).

The operator actions associated with this event would be similar to those described in Section 15.8.1.3.4.

15.8.4.4 Barrier Performance

This event would result in cladding oxidation of less than 1% by volume. Peak fuel rod enthalpy would be less than 280 cal/g. Very few if any fuel rod perforations would be experienced.

15.8.4.5 Radiological Consequences

Radiological consequences are minimal due to the small (if any) number of fuel rod perforations.

15.8.5 Loss of Condenser Vacuum

See the introduction to Section 15.8 for information regarding use of details from this analysis description which may not be applicable to the current plant design.

15.8.5.1 Identification of Causes

The reduction or loss of vacuum in the main condenser can be caused by loss of circulating water pumps or ineffectual operation of the vacuum support equipment. The long-term results of the event would be similar to a condenser isolation unless enough vacuum can be maintained to preserve bypass flow. Preserving the bypass flow to the condenser would permit decay heat removal through the condenser instead of relying upon the suppression pool and the shutdown cooling systems. [15.8-14]

15.8.5.2 Sequence of Events and Systems Operations

Loss of condenser vacuum would trip the turbine stop valves closed (which would normally scram the reactor). If the event is severe enough the steam bypass valves would be closed. These actions would occur normally over a period of several minutes or at worst, 20-30 seconds. The initial sequence of events would be the same as a turbine-generator trip since all systems would function in the same way.

15.8.5.3 Core and System Performance

A. Reactor Shutdown by RPT and ARI

The loss of condenser vacuum event would result in short-term peak values that would be less severe than in the MSIV closure event. All ATWS logic would be rapidly activated by the high pressure transient. The longer term nature of this event (assuming vacuum continues to deteriorate) would be converted to a nearly normal condenser isolation by action of the ARI which would achieve hot shutdown. The long-term response would be similar to the response for the MSIV closure event. The peak vessel pressure experienced in this event would be less than in the MSIV closure event.

B. Reactor Shutdown by RPT and SBLC (No ARI)

In the event that insertion of the control rods via ARI is not achievable, the SBLC system would be utilized as an alternative method of effecting neutronic power shutdown.

The operator actions associated with this event would be similar to those described in Section 15.8.1.3.4

15.8.5.4 Barrier Performance

This event would result in cladding oxidation of less than 1% by volume. Peak fuel rod enthalpy would be less than 280 cal/g. Very few if any fuel rod perforations would be experienced.

15.8.5.5 Radiological Consequences

The radiological consequences would be minimal due to the small (if any) number of fuel rod perforations.

15.8.6 Pressure Regulator Failure – Open to Maximum Demand

See the introduction to Section 15.8 for information regarding the use of details from the analysis description for applicability to the current plant design.

15.8.6.1 Identification of Causes

Pressure regulator failure to the maximum demand is caused by the malfunction of the normal pressure regulator. The feedback pressure regulator responds only to the low demand signal and does not intervene when the demand signal is high.

15.8.6.2 Sequence of Events and Systems Operations

This transient would be initiated by the failure of the pressure regulator, which generates a maximum demand signal for the turbine-generator. This signal yields the opening of all the turbine bypass valves and forces the turbine control valves to the fully open position. This causes the reactor vessel to depressuirze and to void the core, which in turn further reduces the reactor power and the vessel pressure. The MSIVs start to close once the low steamline pressure is detected at about 13 seconds. The reactor pressure would reach at ATWS mitigation system setpoint of 1250 psig approximately 21 seconds after the start of the event (Reference 4). The ATWS mitigation system would initiate a recirculation pump trip (RPT) and would initiate alternate rod insertion (ARI). The RPT and ARI would introduce negative reactivity into the core. The reactor pressure would peak in about 27 seconds (Reference 4). The standby liquid control (SBLC) system can be initiated if ARI is unavailable. The reactor water level is reduced and maintained at an elevation consistent with Emergency Operation procedures (EOP). The water level is restored to the normal level once the hot shutdown boron weight (HSBW) is injected into the vessel.

15.8.6.3 Core and System Performance

In the event that control rod insertion (via ARI) is unavailable for shutdown of the reactor, the SBLC system would be used as an alternative method of achieving shutdown. According to the Westinghouse analysis presented in Reference 6, the transient would result in a peak vessel pressure less than the overpressure limit of 1500 psig. Westinghouse checks the limiting ATWS pressure every reload. The peak suppression pool temperature is calculated to be less than the bounding post-accident suppression pool temperature limit of 202 degrees F. Therefore, the ATWS acceptance criteria continue to be satisfied for SVEA-96 Optima2 fuel.

15.8.6.4 Barrier Response

This event would result in cladding oxidation of less than 1% by volume. Peak fuel rod enthalpy would be less than 280 cal/g. Very few (if any) fuel rod perforations would be experienced.

15.8.6.5 Radiological Consequences

The radiological consequences would be minimal due to the small (if any) number of fuel rod perforations.

15.8.7 References

1. "Studies of ATWS for Dresden 2, 3 and Quad Cities 1, 2 Nuclear Power Stations", General Electric Co., NEDE-25026, December 1976,

2. "Main Steam Isolation Valves Closure Event With ATWS/RPT and ARI for Dresden 2, 3 and Quad Cities 1, 2 Nuclear Generating Plants," General Electric Company, NSE-45-0880, August 1980.

3. Deleted.

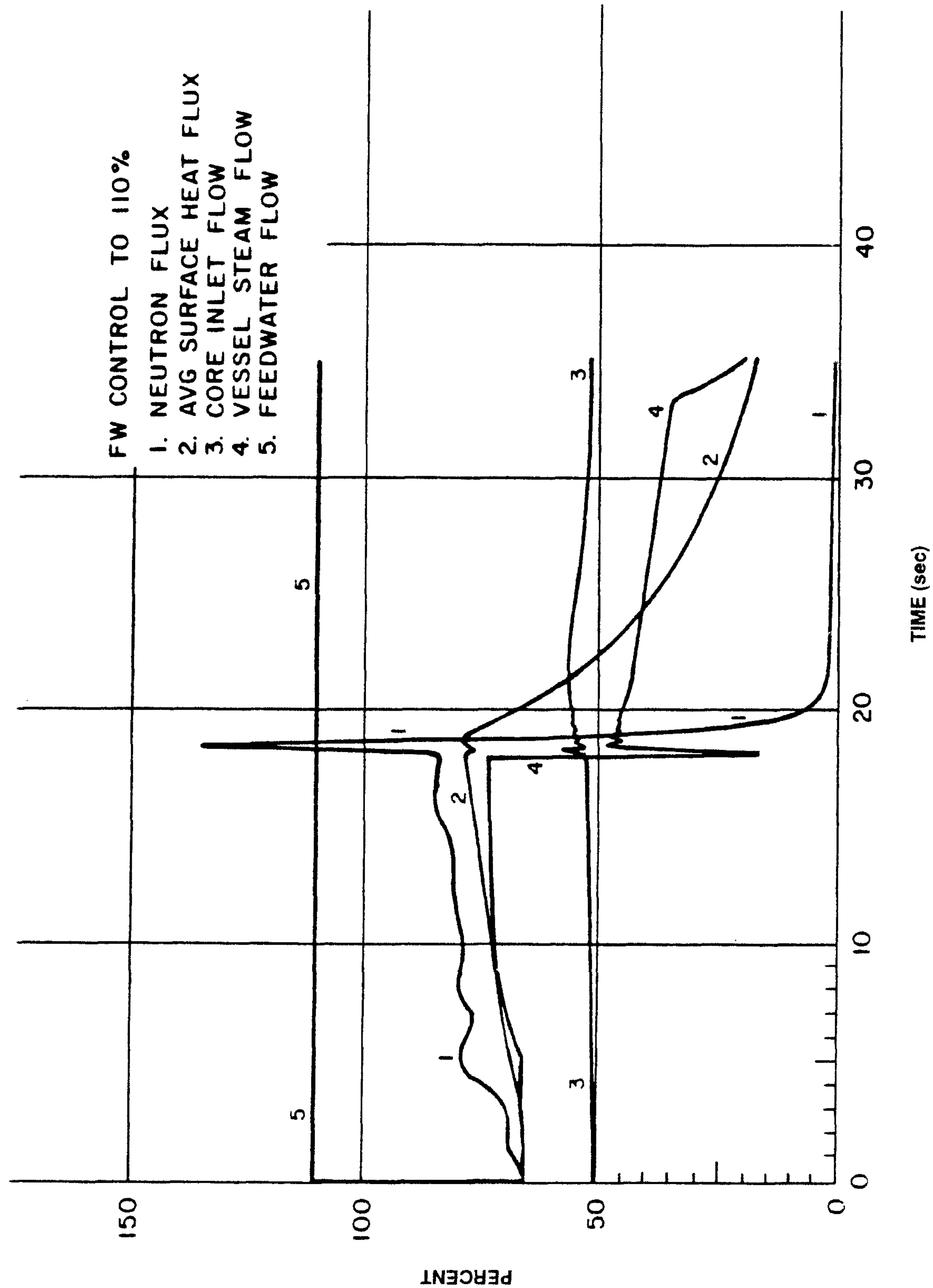
4. "Dresden and Quad Cities Extended Power Uprate, Task T0902: Anticipated Transient Without Scram, " GE-NE-A22-00103-11-01, Revision 2, February 2002.

5. Deleted.

6. OPTIMA2-TR026QC-ATWS, Revision 1, "ATWS Analysis for the Introduction of SVEA-96 Optima2 Fuel at Quad Cities 1 & 2," April 2007.

7. GE-NE-0000-0050-6728-01, Revision 1, "Quad Cities Acoustical Side Branch Modification Evaluation of Current GE Tasks," March 2006.

8. NF-BEX-08-133, Revision 1, "Evaluation of the Planned Implementation of Adjustable Speed Drives in Quad Cities Units 1 and 2," February 2009.

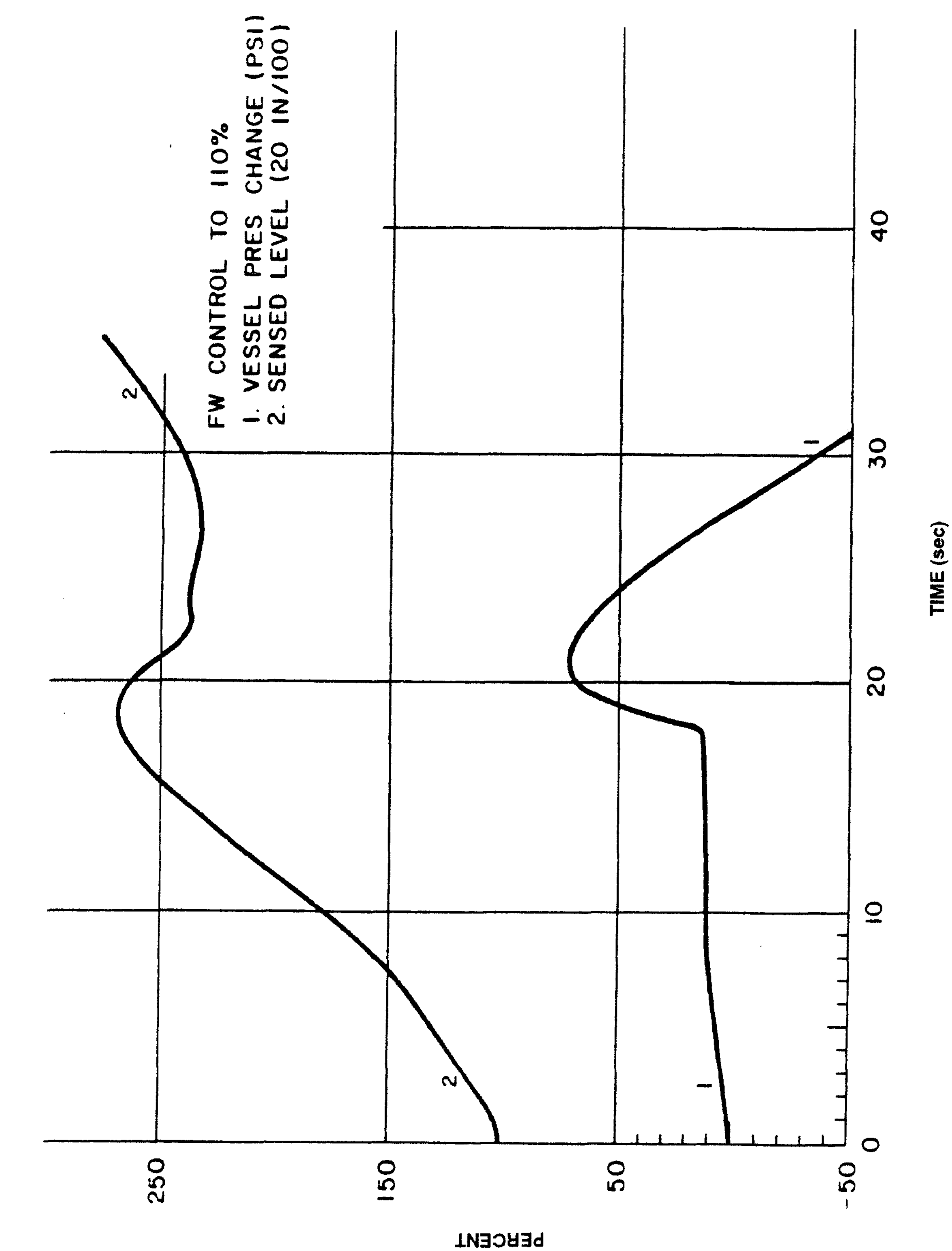


See the introduction to Section 15.1 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION
UNITS 1 & 2

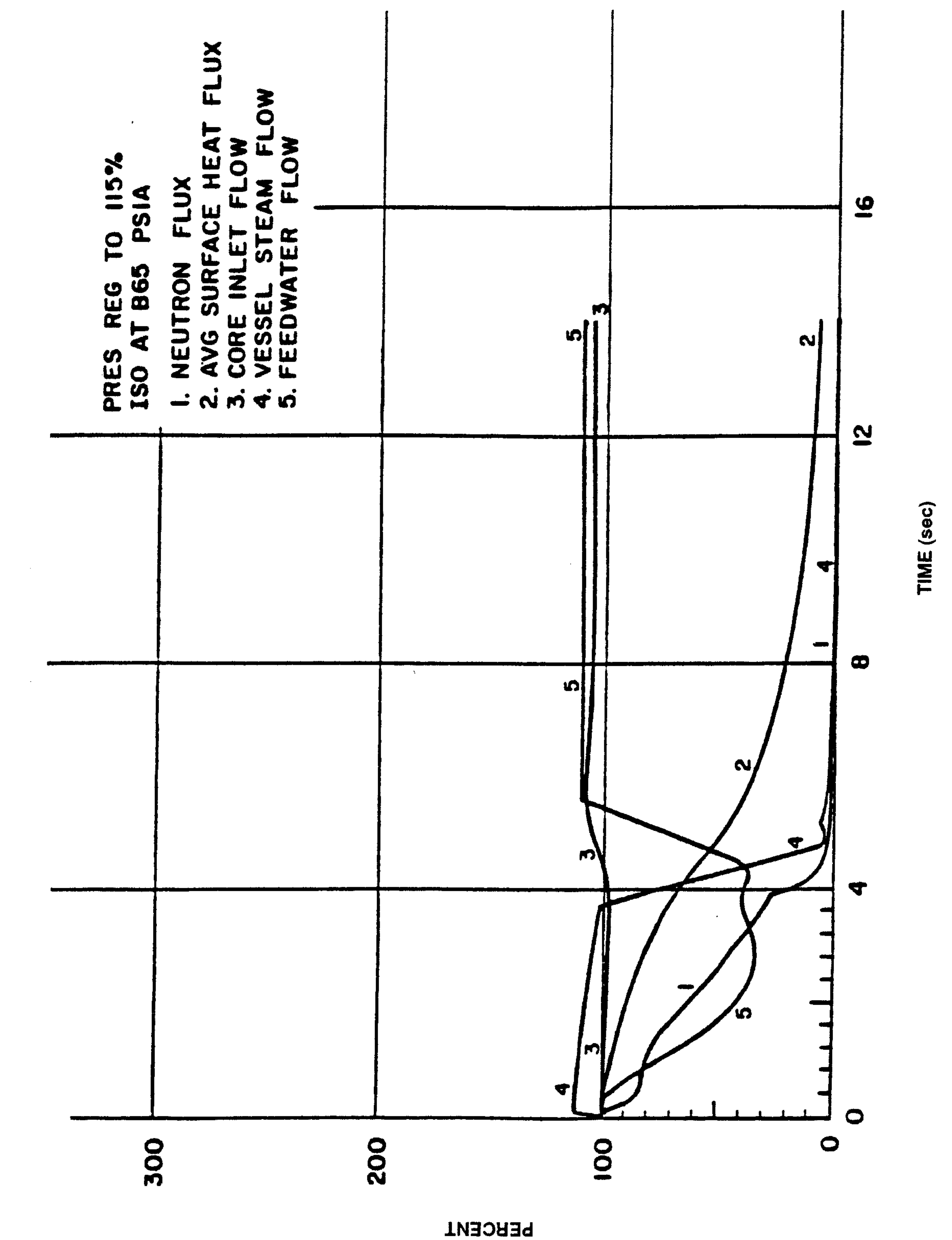
RESULTS OF ANALYSIS
FEEDWATER CONTROL TO 110%

FIGURE 15.1-1



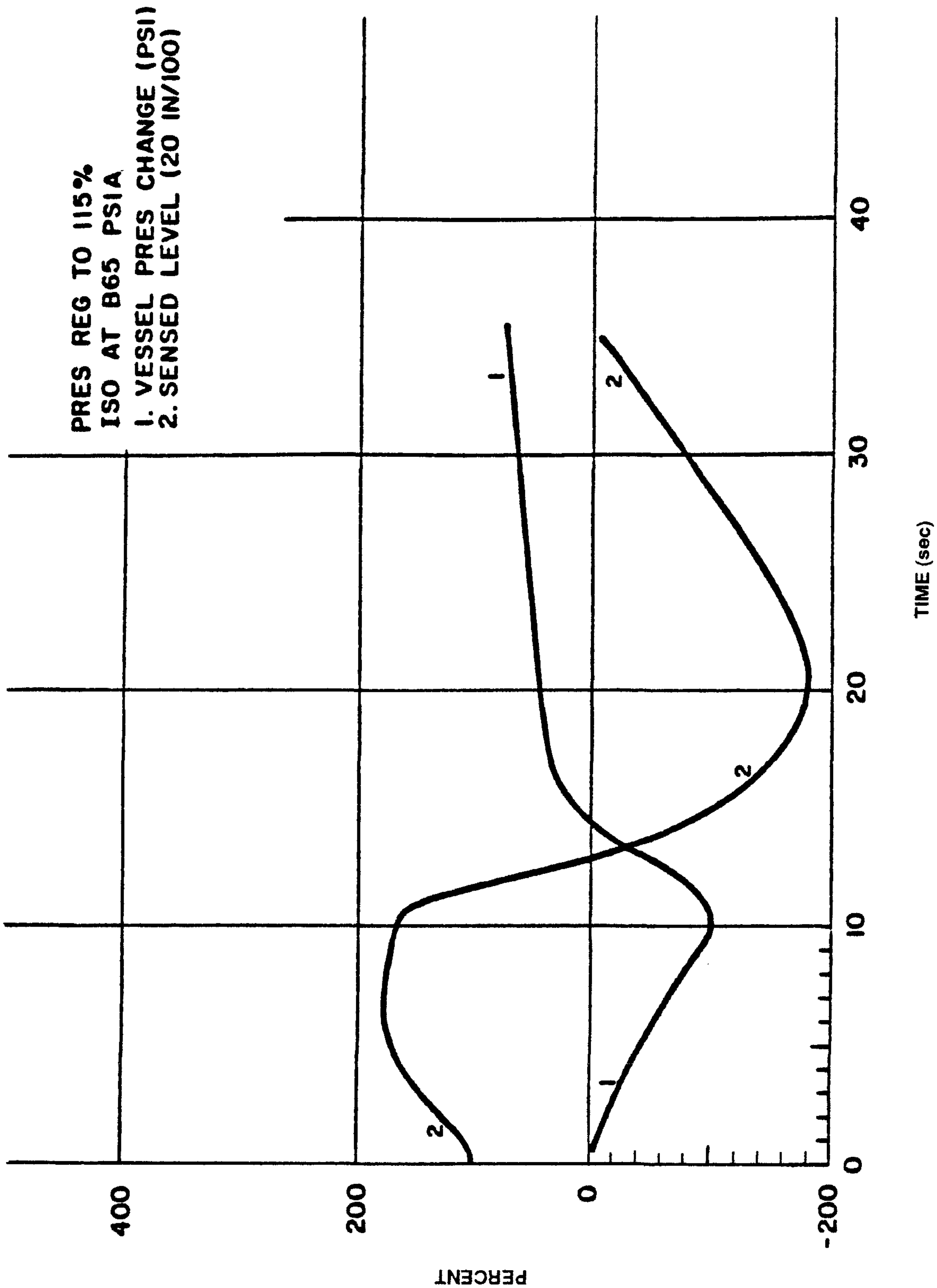
See the introduction to Section 15.1 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION UNITS 1 & 2
RESULTS OF ANALYSIS FEEDWATER CONTROL
FIGURE 15.1-2



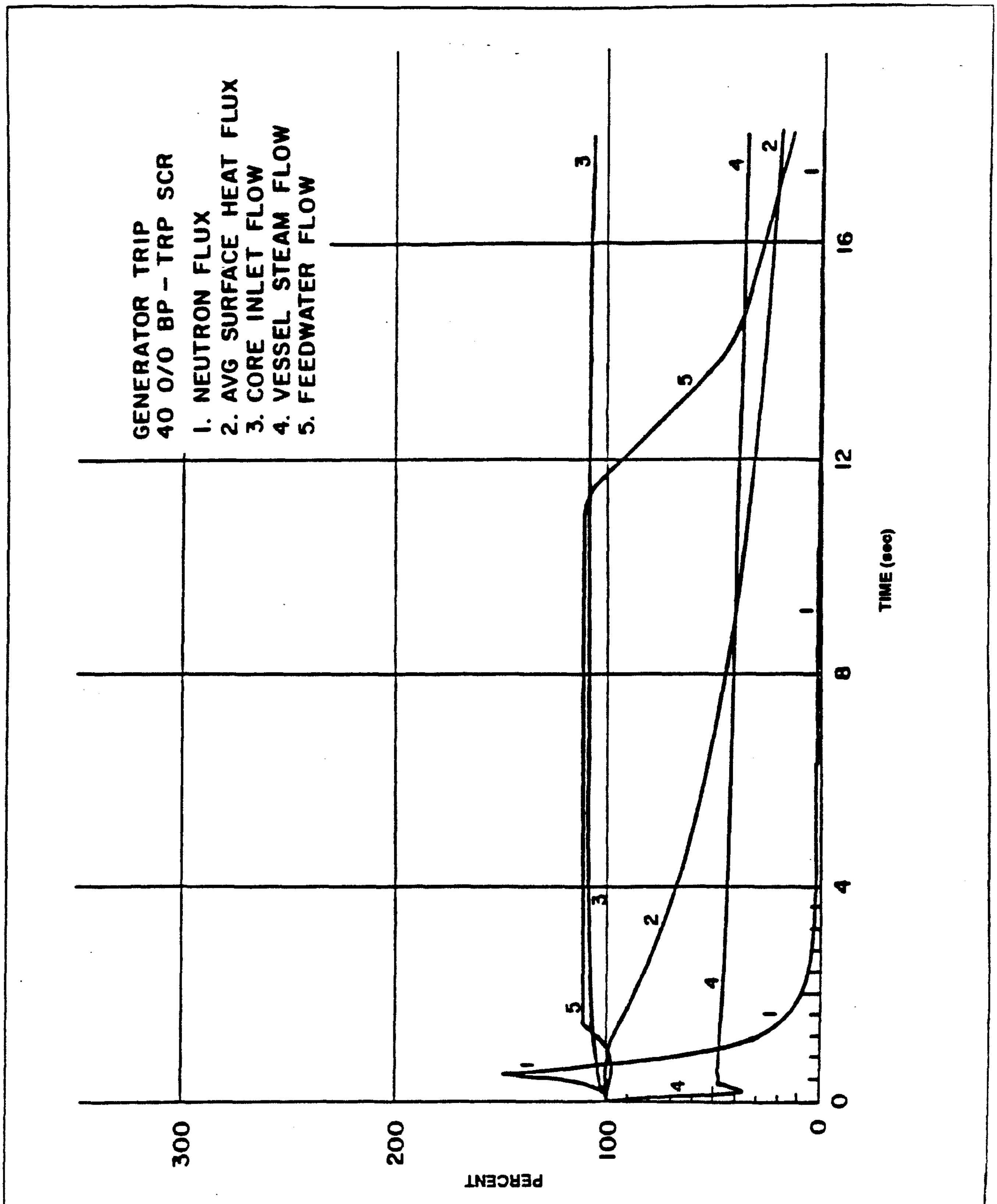
See the introduction to Section 15.1 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION UNITS 1 & 2
RESULTS OF ANALYSIS PRESSURE REGULATOR TO 115%
FIGURE 15.1-3



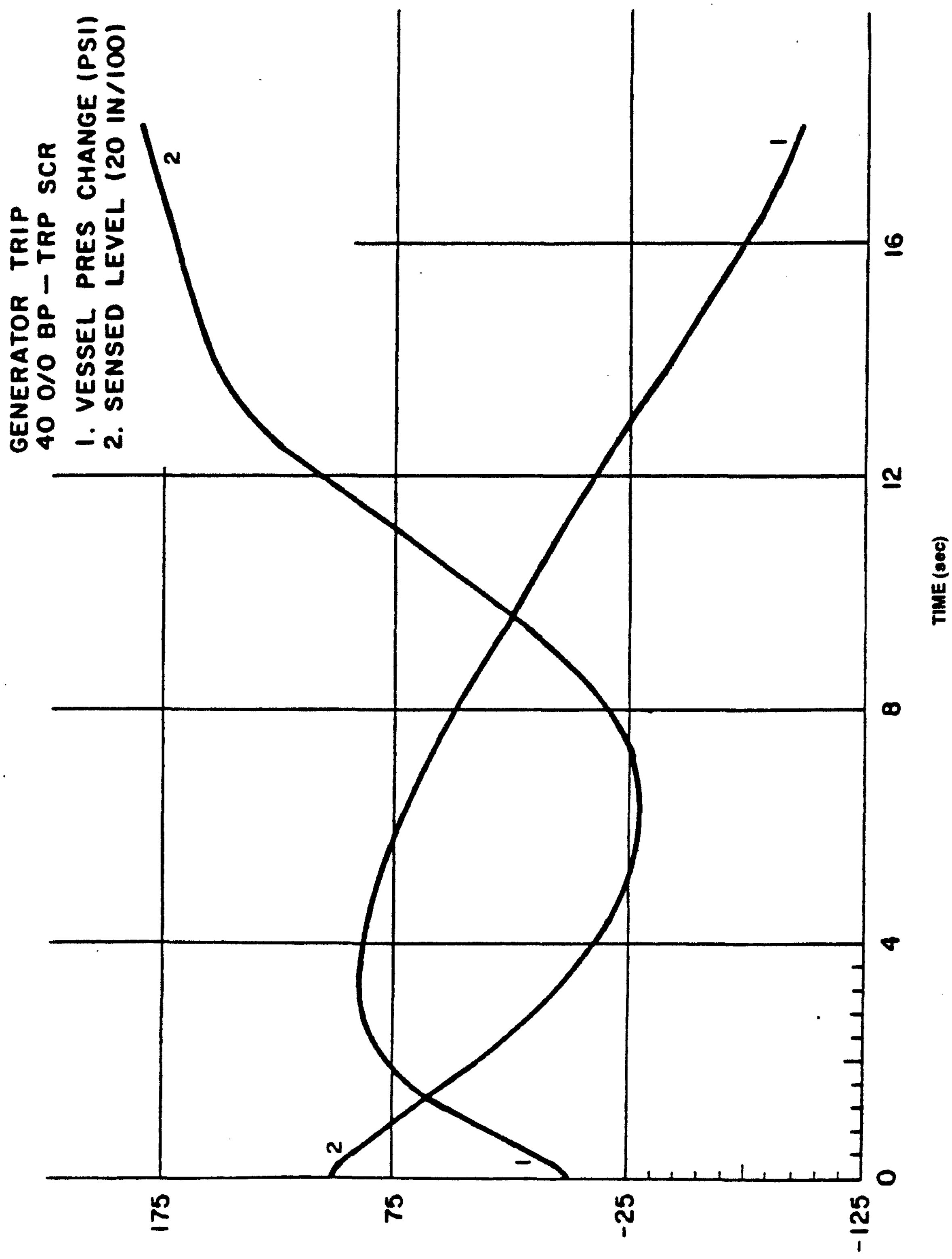
See the introduction to Section 15.1 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION UNITS 1 & 2
RESULTS OF ANALYSIS PRESSURE REGULATOR TO 115%
FIGURE 15.1-4



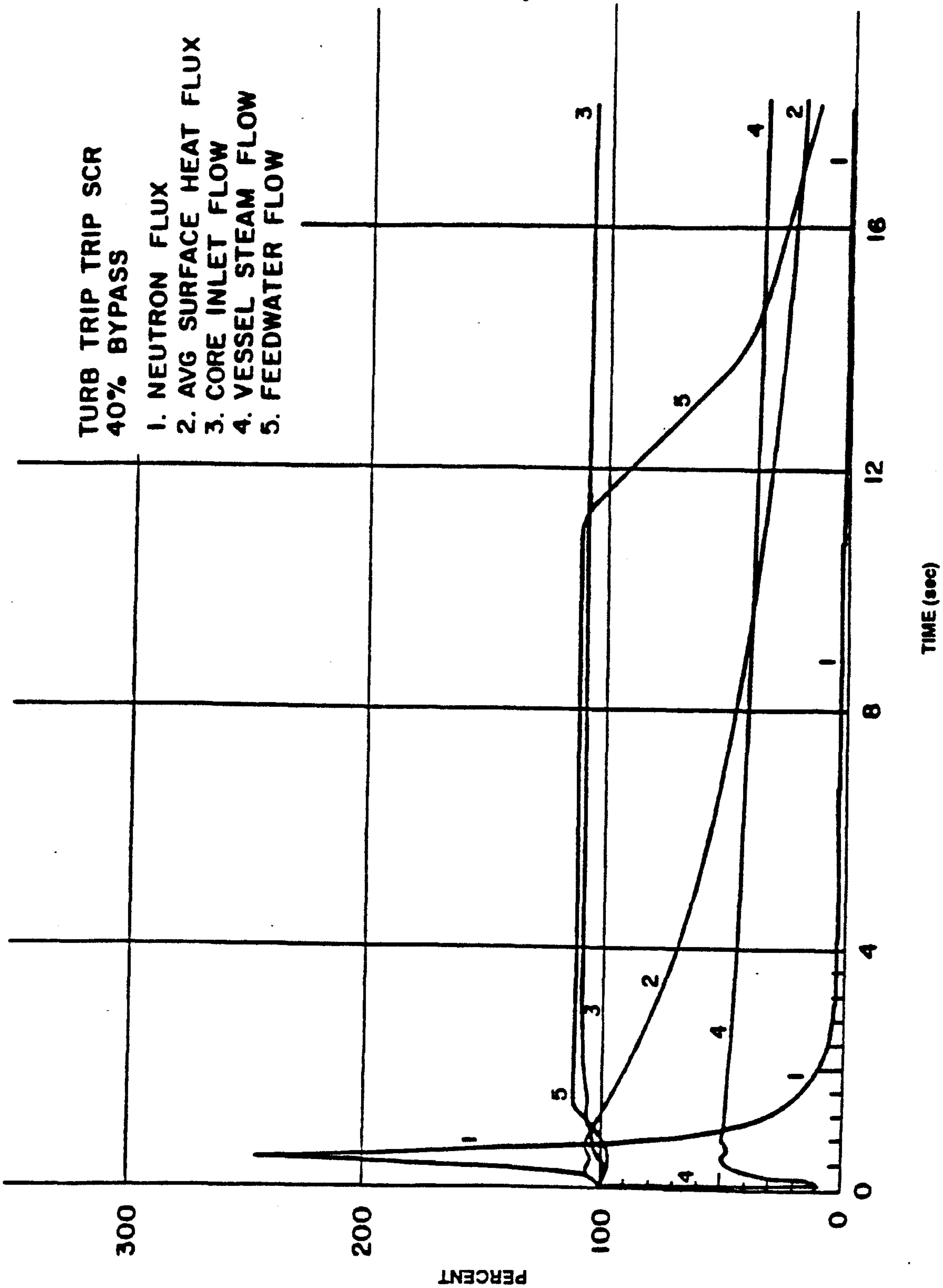
See the introduction to Section 15.2 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION UNITS 1 & 2
(INITIAL CORE) RESULTS OF ANALYSIS GENERATOR TRIP
FIGURE 15.2-1 REVISION 4, APRIL 1997



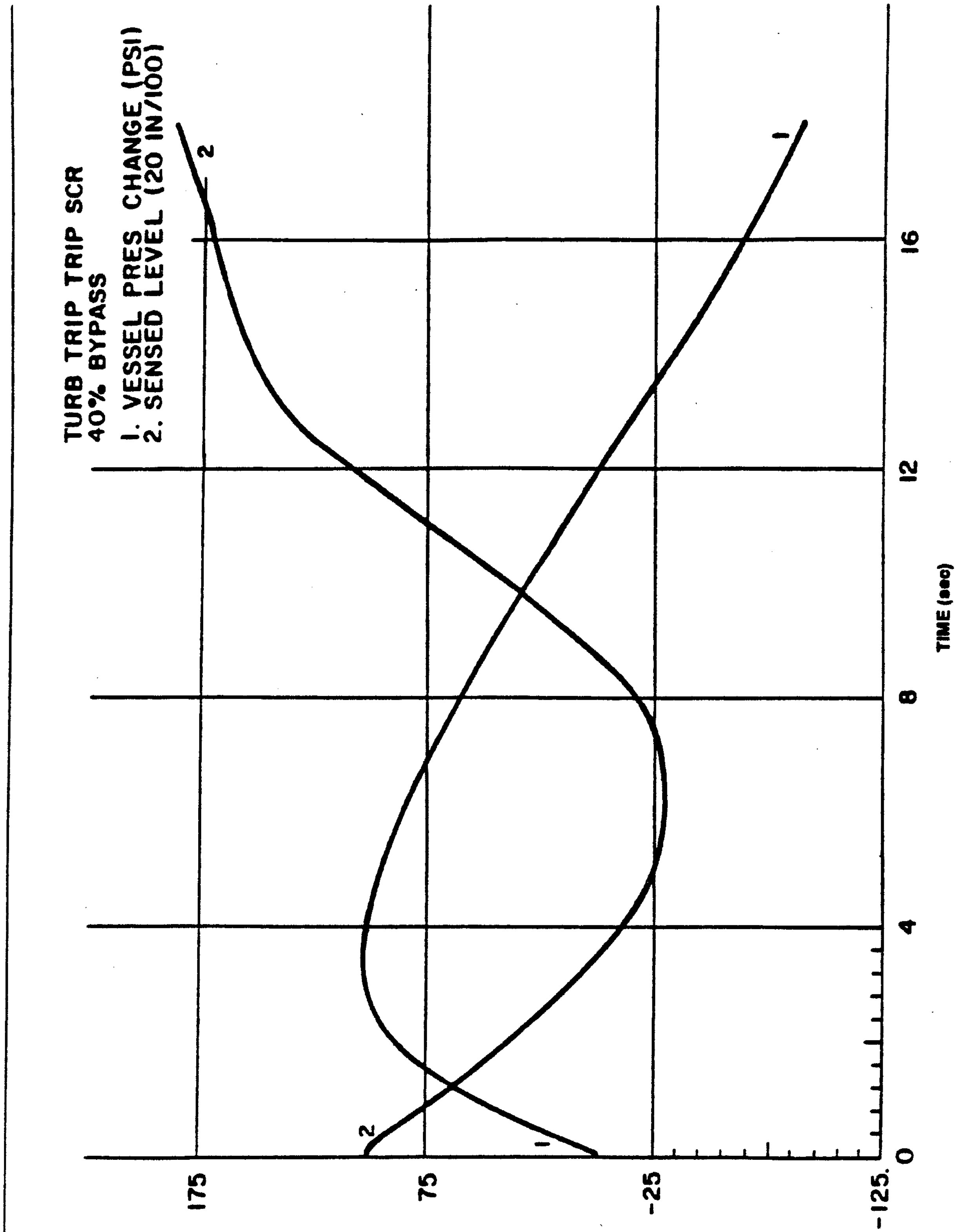
See the introduction to Section 15.2 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION UNITS 1 & 2
(INITIAL CORE) RESULTS OF ANALYSIS GENERATOR TRIP
FIGURE 15.2-2 REVISION 4, APRIL 1997



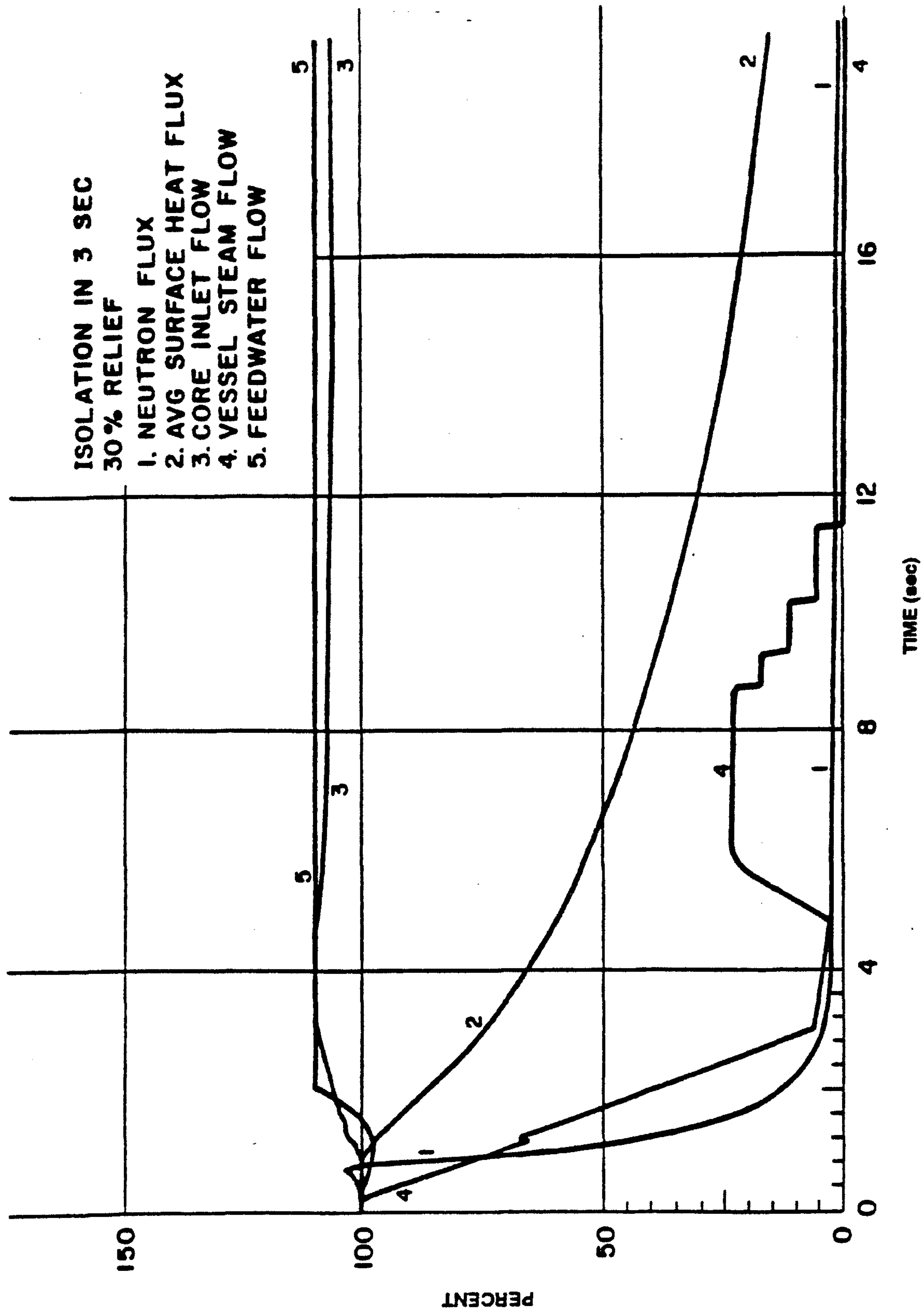
See the introduction to Section 15.2 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION UNITS 1 & 2
(INITIAL CORE) RESULTS OF ANALYSIS TURBINE TRIP (40% BYPASS)
FIGURE 15.2-3
REVISION 4, APRIL 1997



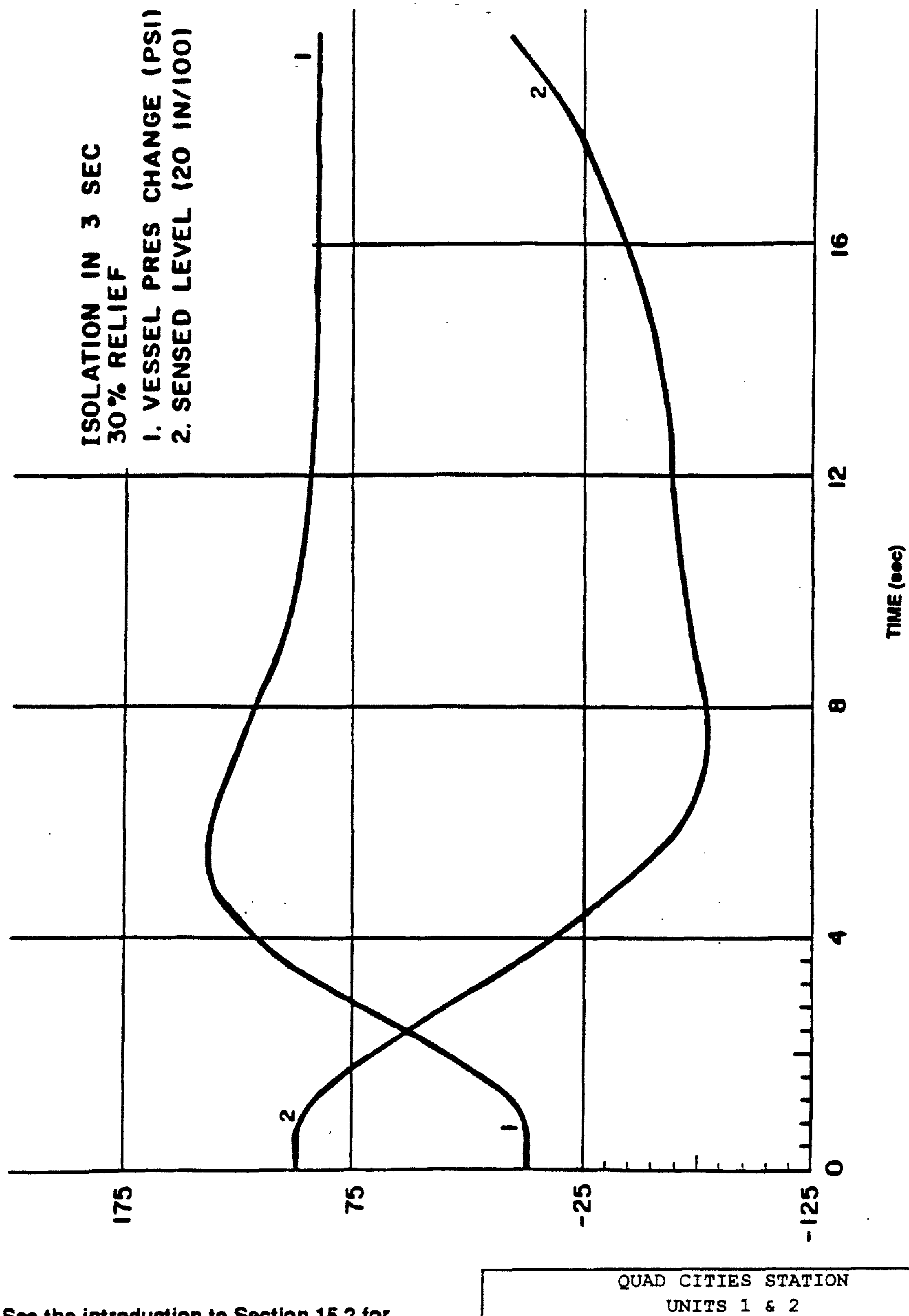
See the introduction to Section 15.2 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION UNITS 1 & 2
(INITIAL CORE) RESULTS OF ANALYSIS TURBINE TRIP (40% BYPASS)
FIGURE 15.2-4 REVISION 4, APRIL 1997



See the introduction to Section 15.2 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION UNITS 1 & 2
(INITIAL CORE) RESULTS OF ANALYSIS MSIV ISOLATION IN 3-SEC (30% RELIEF)
FIGURE 15.2-5 REVISION 4, APRIL 1997



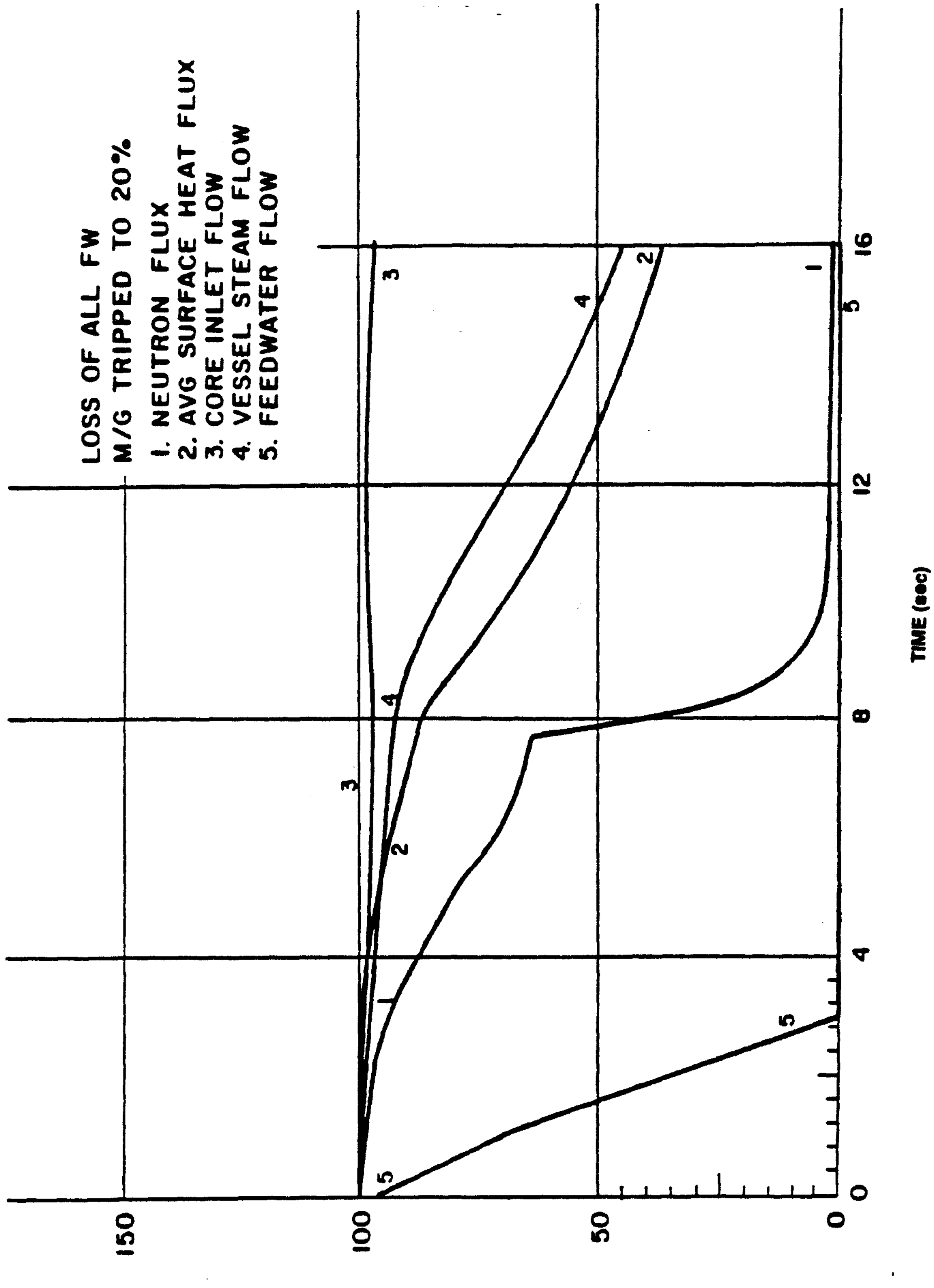
See the introduction to Section 15.2 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION
UNITS 1 & 2

(INITIAL CORE) RESULTS OF ANALYSIS
MSIV ISOLATION IN 3-SEC
(30% RELIEF)

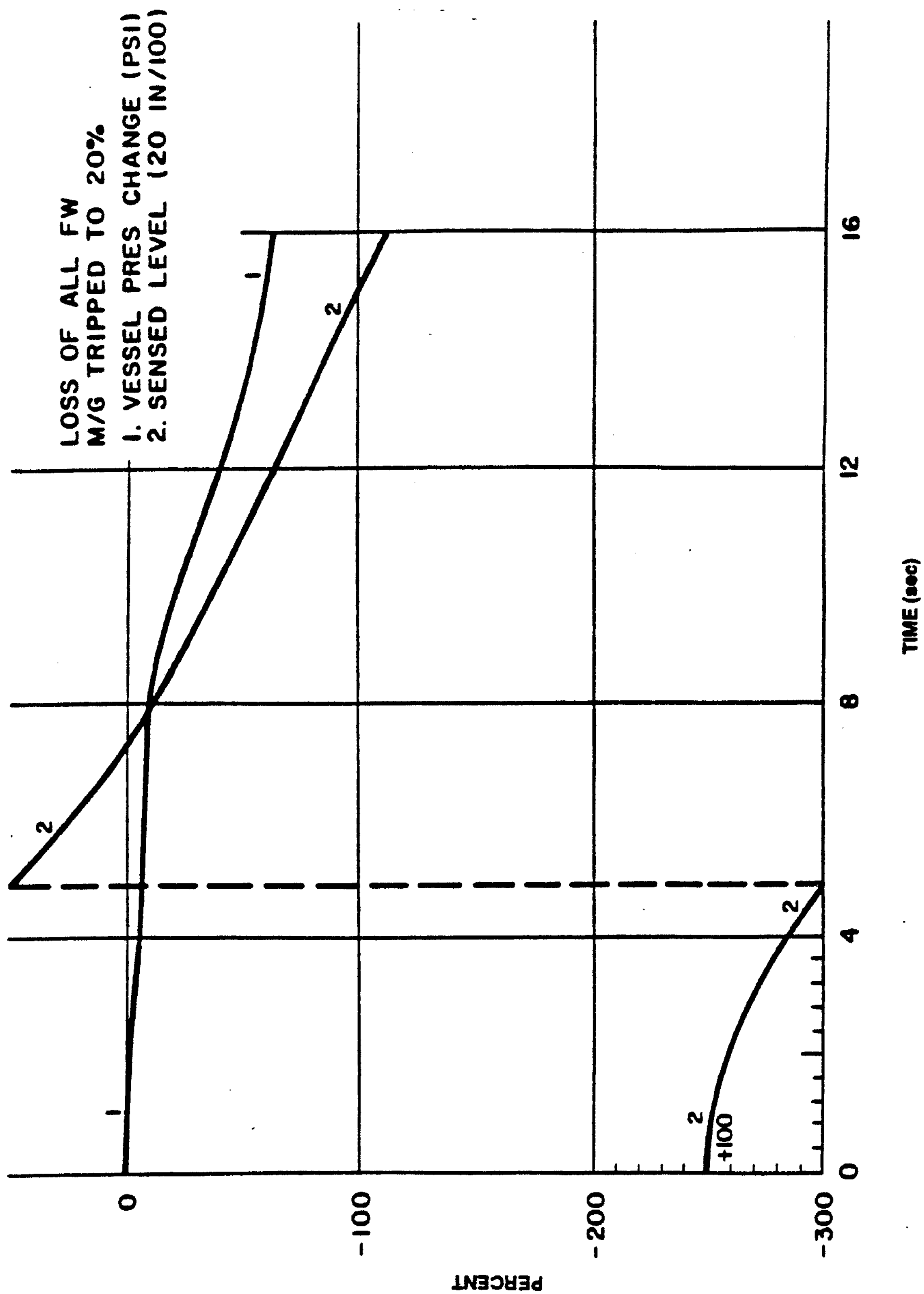
FIGURE 15.2-6

REVISION 4, APRIL 1997



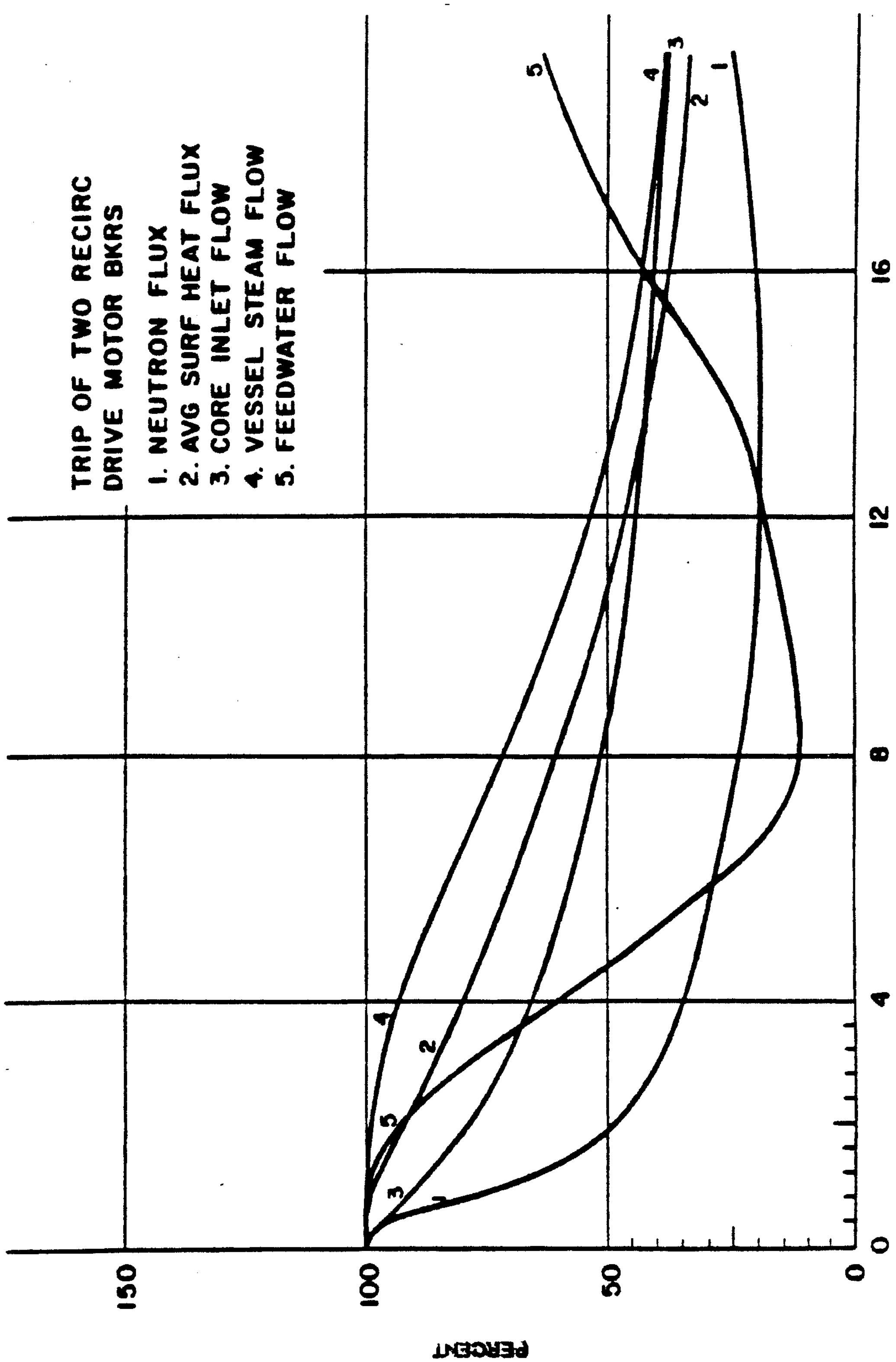
See the introduction to Section 15.2 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION UNITS 1 & 2
(INITIAL CORE) RESULTS OF ANALYSIS LOSS OF ALL FEEDWATER
FIGURE 15.2-7 REVISION 4, APRIL 1997



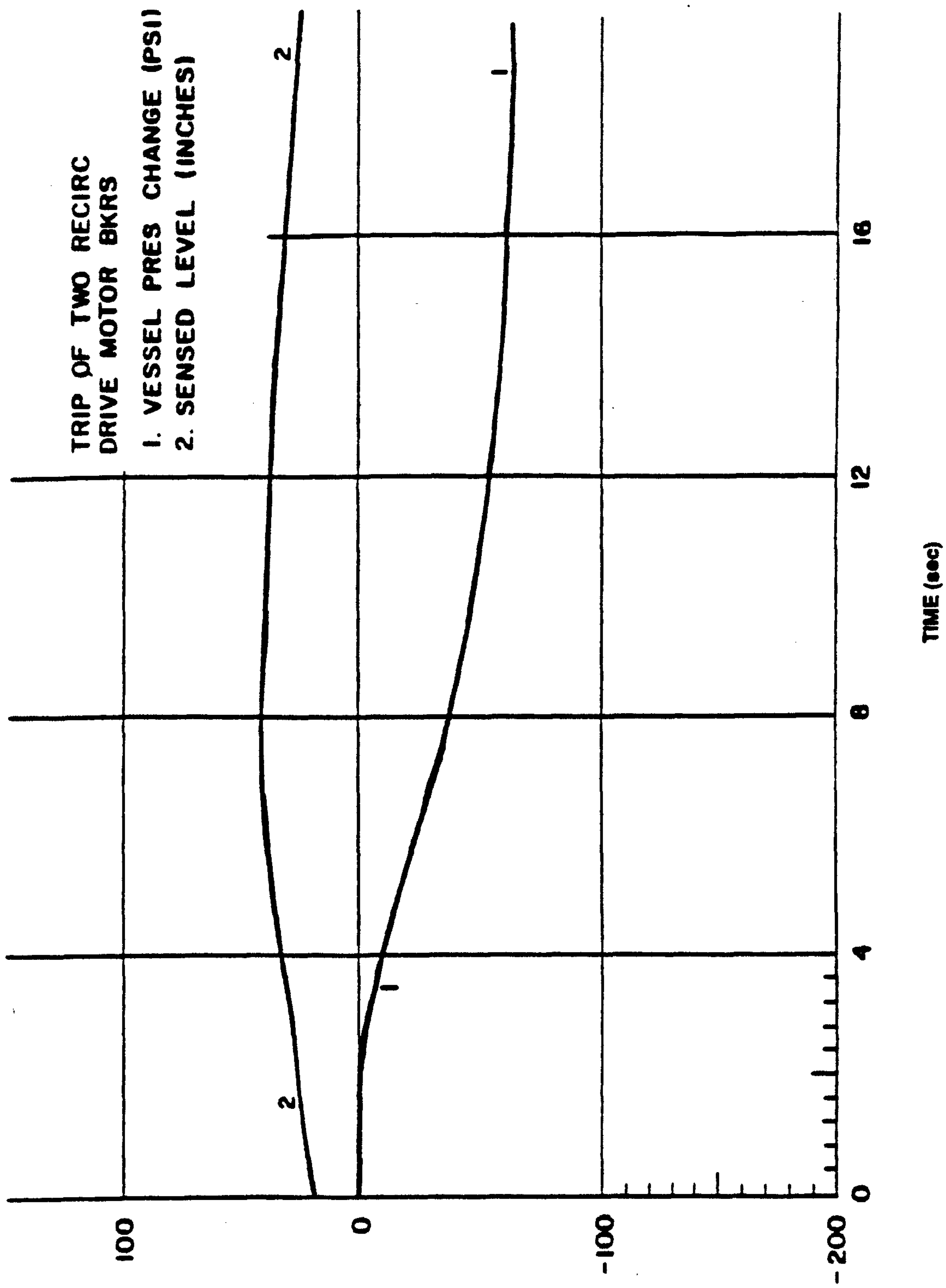
See the introduction to Section 15.2 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION UNITS 1 & 2
(INITIAL CORE) RESULTS OF ANALYSIS LOSS OF ALL FEEDWATER
FIGURE 15.2-8 REVISION 4, APRIL 1997



See the introduction to Section 15.3 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION UNITS 1 & 2
(INITIAL CORE) TRIP OF TWO RECIRCULATION DRIVE MOTOR BREAKERS TRANSIENT ANALYSIS
FIGURE 15.3-1 REVISION 4, APRIL 1997



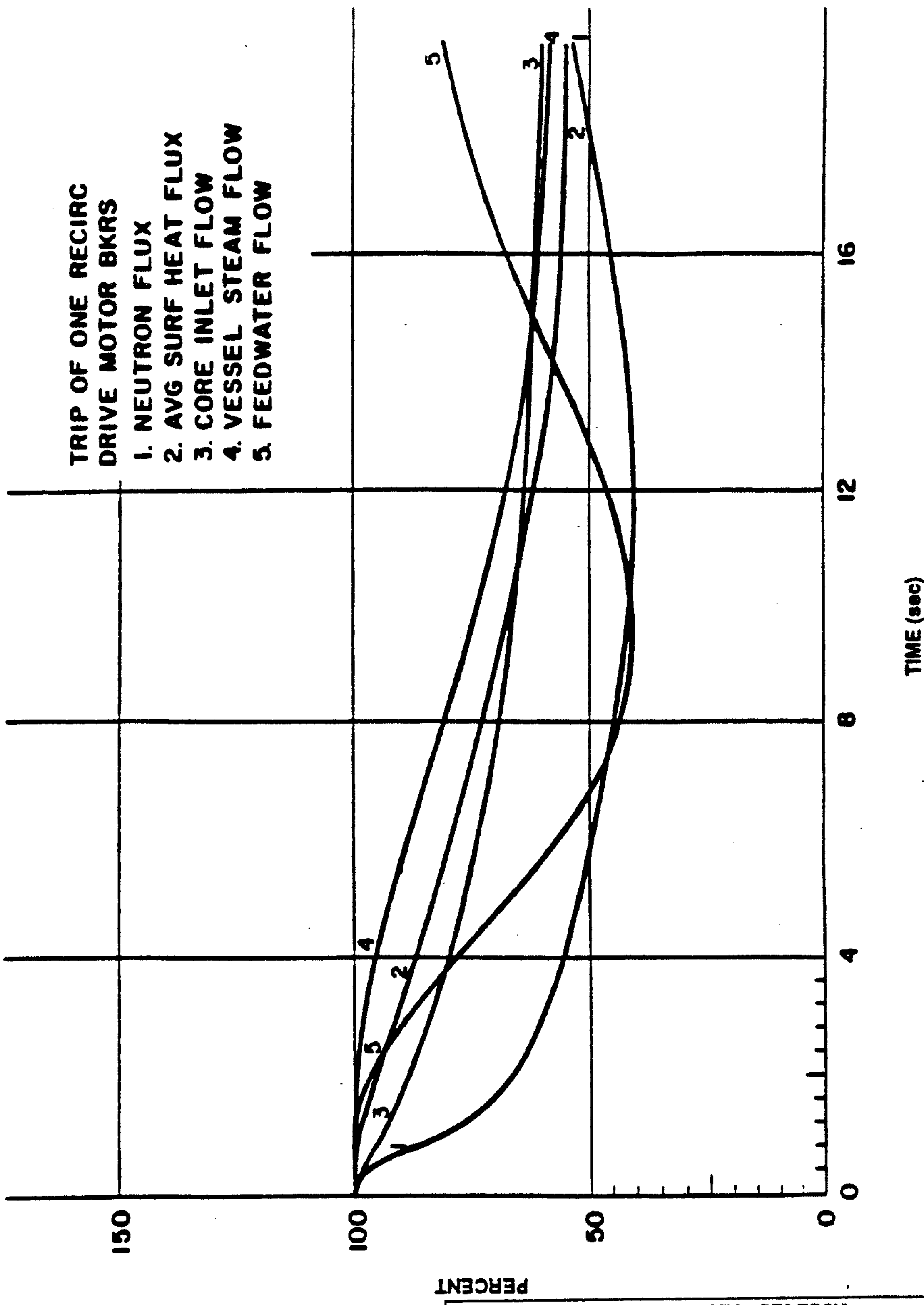
See the introduction to Section 15.3 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION
UNITS 1 & 2

(INITIAL CORE) TRIP OF TWO RECIRCULATION DRIVE MOTOR BREAKERS TRANSIENT ANALYSIS

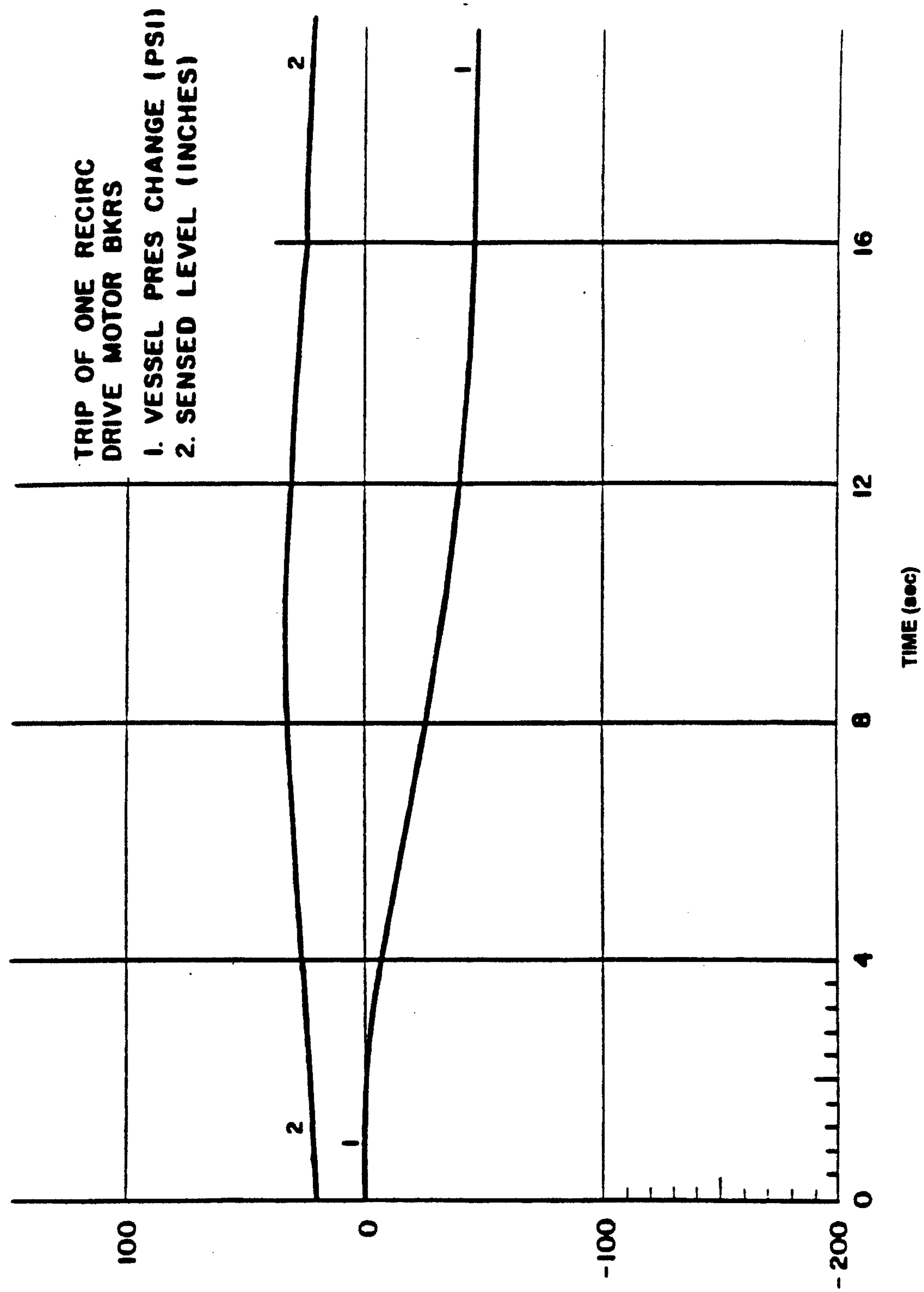
FIGURE 15.3-2

REVISION 4, APRIL 1997



See the introduction to Section 15.3 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION UNITS 1 & 2
(INITIAL CORE) TRIP OF ONE RECIRCULATION DRIVE MOTOR BREAKER - TRANSIENT ANALYSIS
FIGURE 15.3-3 REVISION 4, APRIL 1997



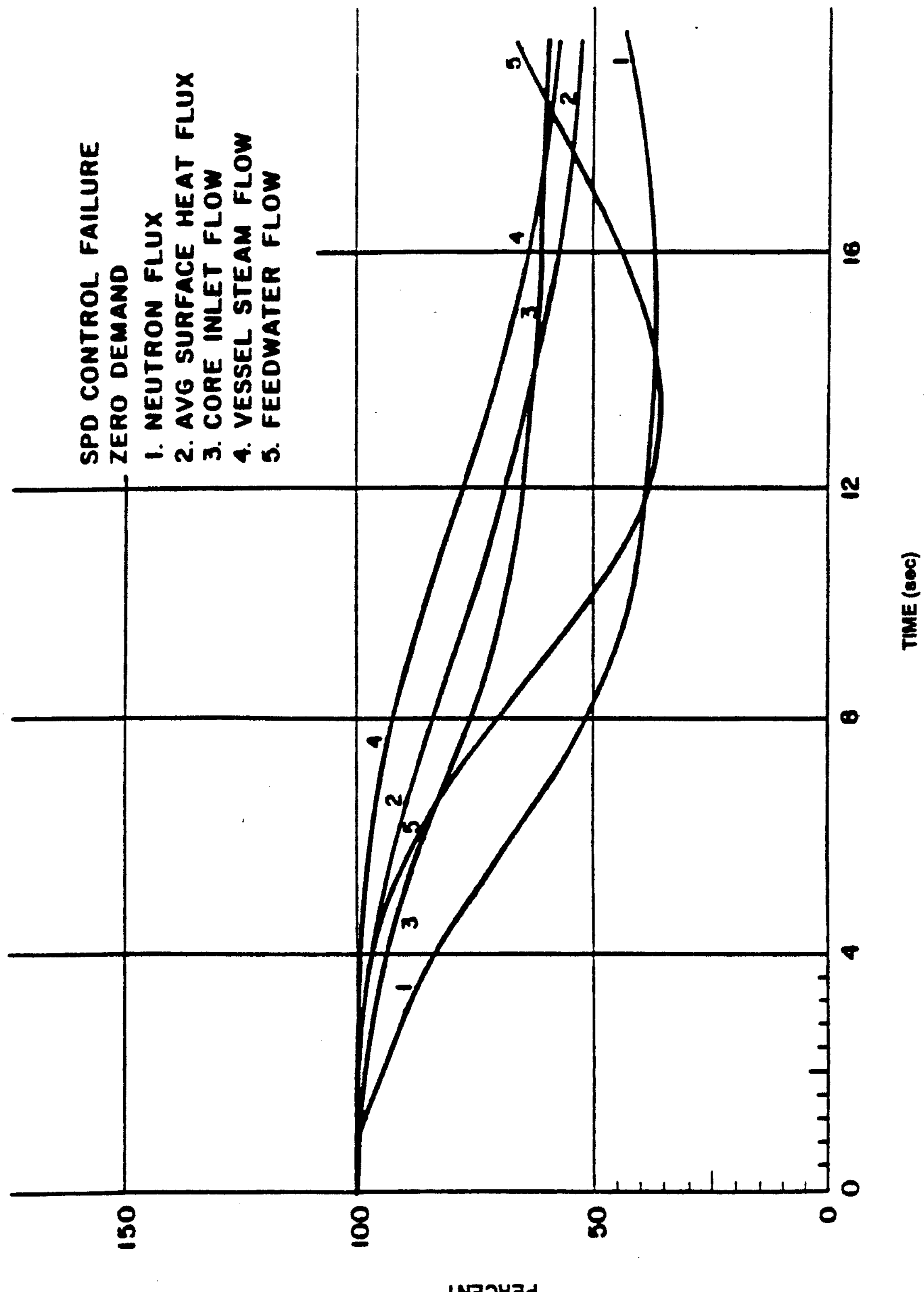
See the introduction to Section 15.3 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION
UNITS 1 & 2

(INITIAL CORE) TRIP OF ONE RECIRCULATION DRIVE MOTOR BREAKER - TRANSIENT ANALYSIS

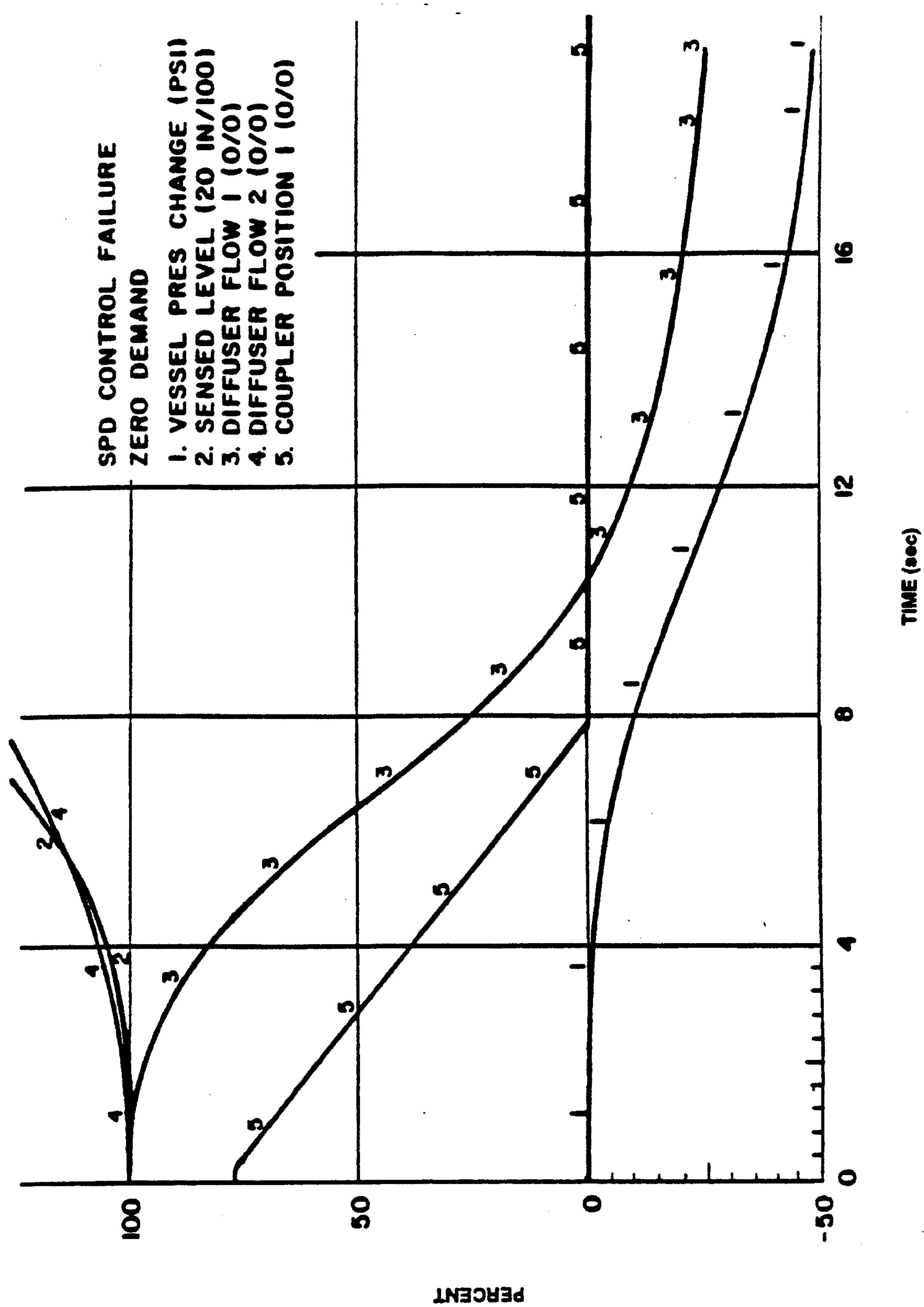
FIGURE 15.3-4

REVISION 4, APRIL 1997



See the introduction to Section 15.3 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION UNITS 1 & 2
(INITIAL CORE) SPEED CONTROL FAILURE, ZERO DEMAND TRANSIENT ANALYSIS
FIGURE 15.3-5 REVISION 4, APRIL 1997



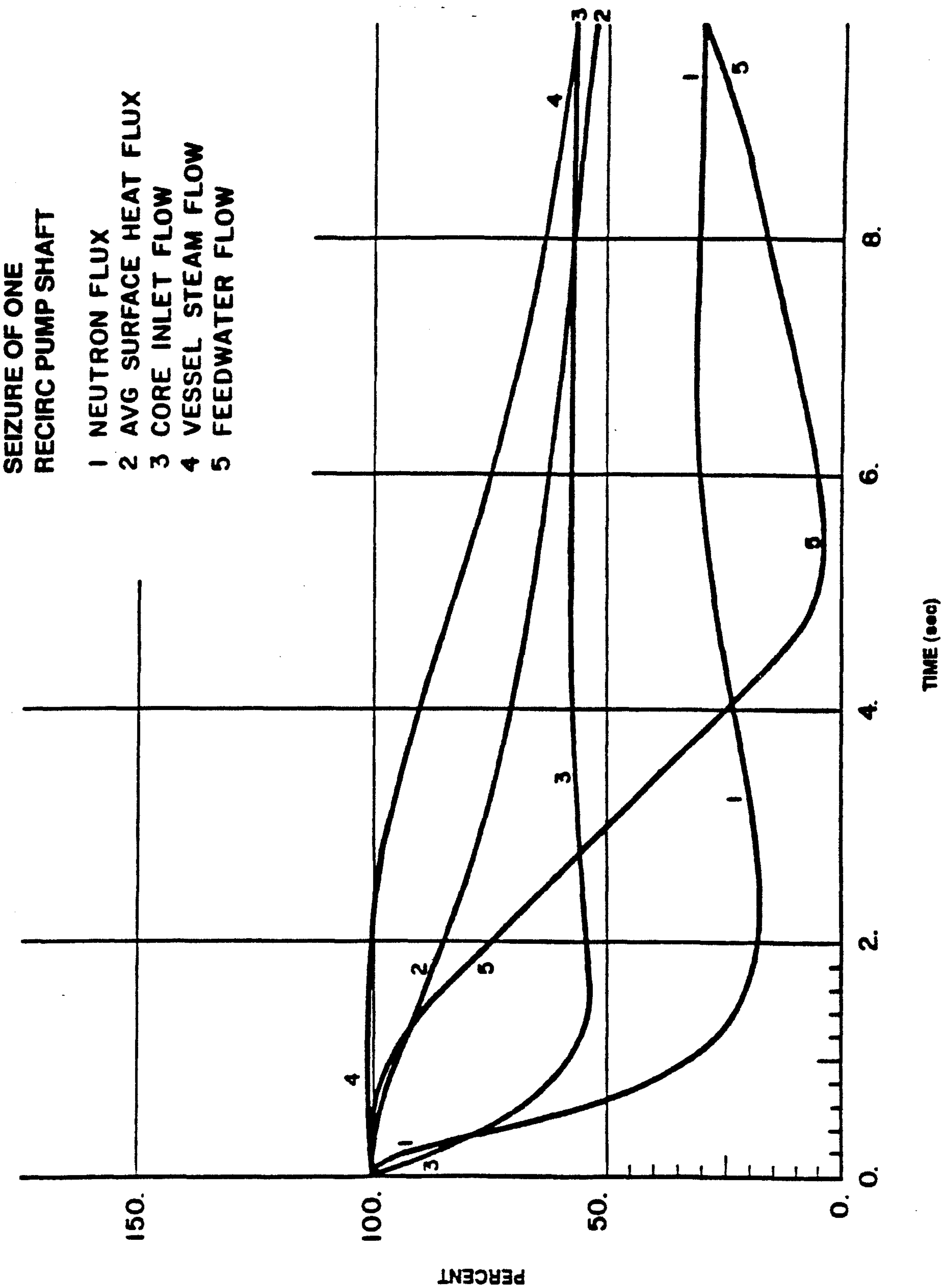
See the introduction to Section 15.3 for information regarding use of details from this analysis which may not he applicable to the current fuel cycle.

QUAD CITIES STATION
UNITS 1 & 2

(INITIAL CORE) SPEED CONTROL FAILURE, ZERO DEMAND TRANSIENT ANALYSIS

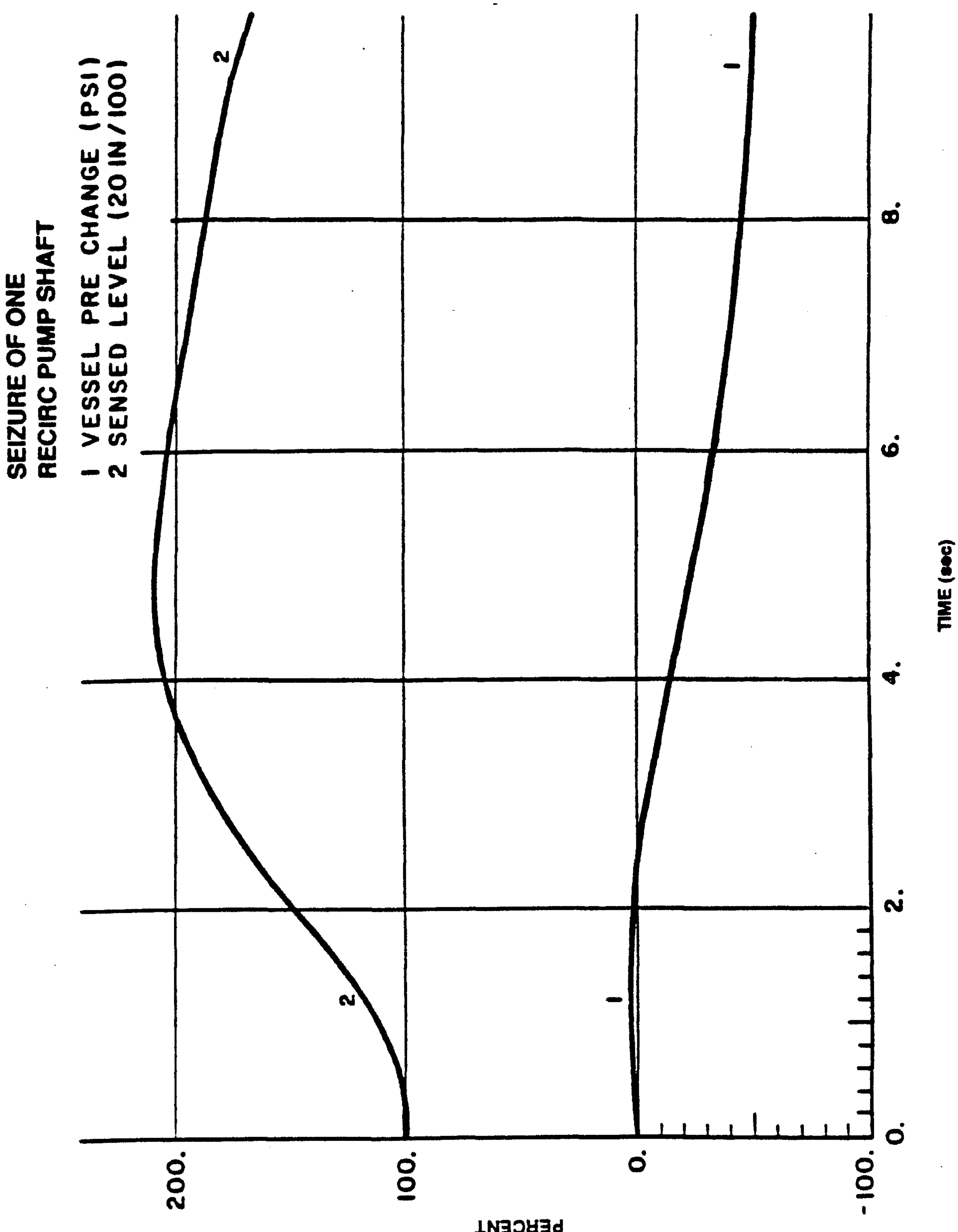
FIGURE 15.3-6

REVISION 4, APRIL 1997



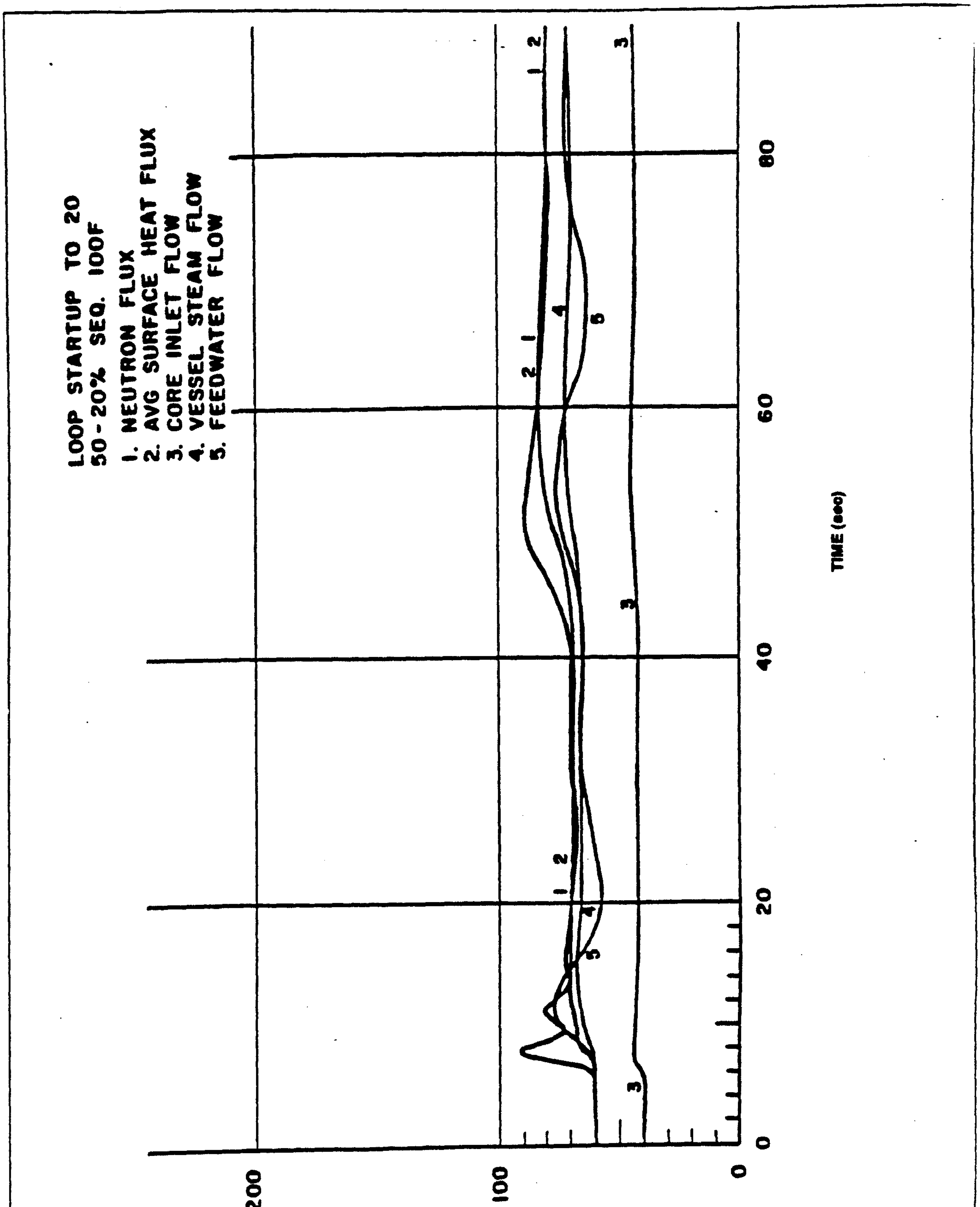
See the introduction to Section 15.3 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION UNITS 1 & 2
(INITIAL CORE) SEIZURE OF ONE RECIRCULATION PUMP SHAFT TRANSIENT ANALYSIS
FIGURE 15.3-7 REVISION 4, APRIL 1997



See the introduction to Section 15.3 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION UNITS 1 & 2
(INITIAL CORE) SEIZURE OF ONE RECIRCULATION PUMP SHAFT TRANSIENT ANALYSIS
FIGURE 15.3-8 REVISION 4, APRIL 1997

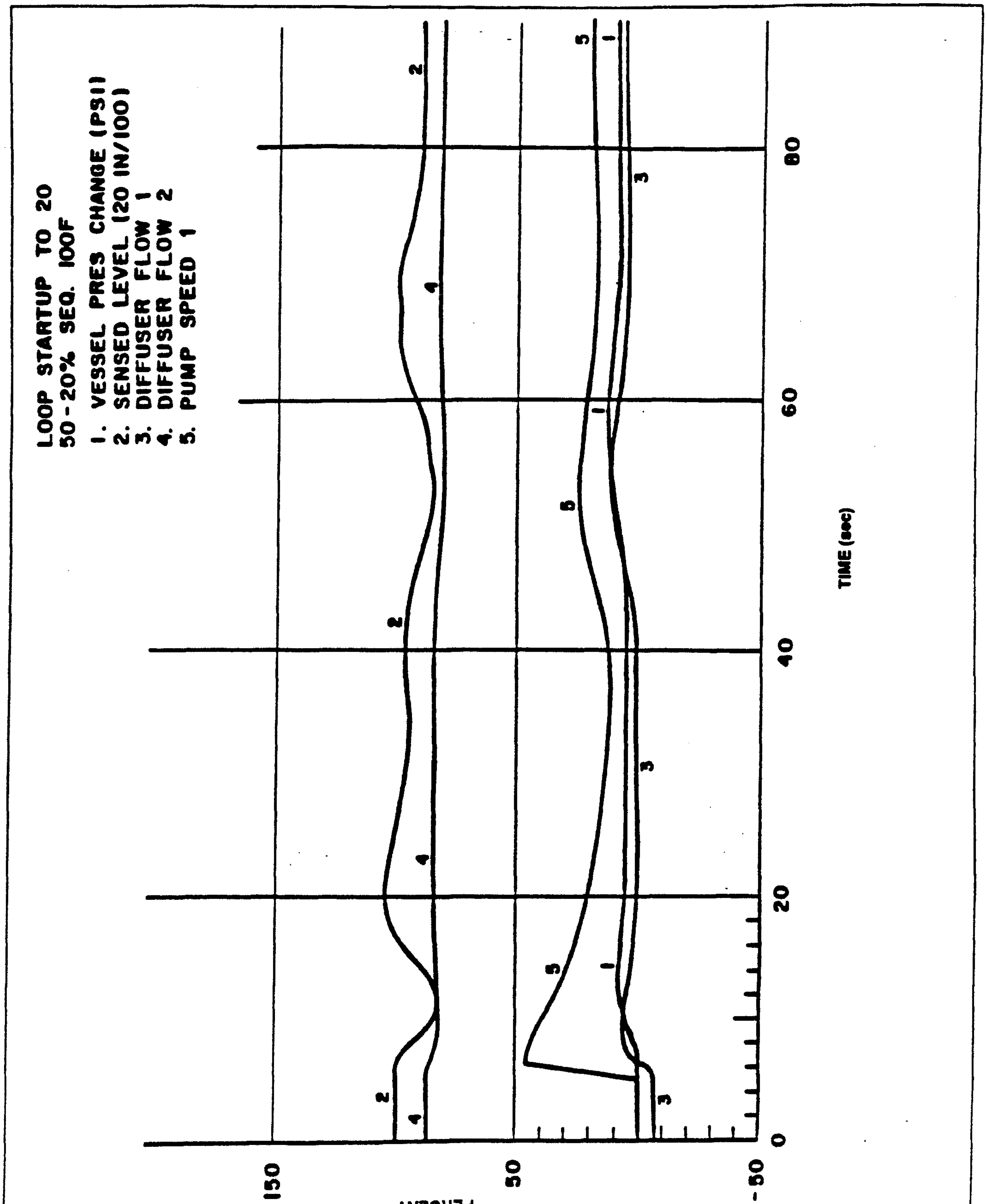


QUAD CITIES STATION
UNITS 1 & 2

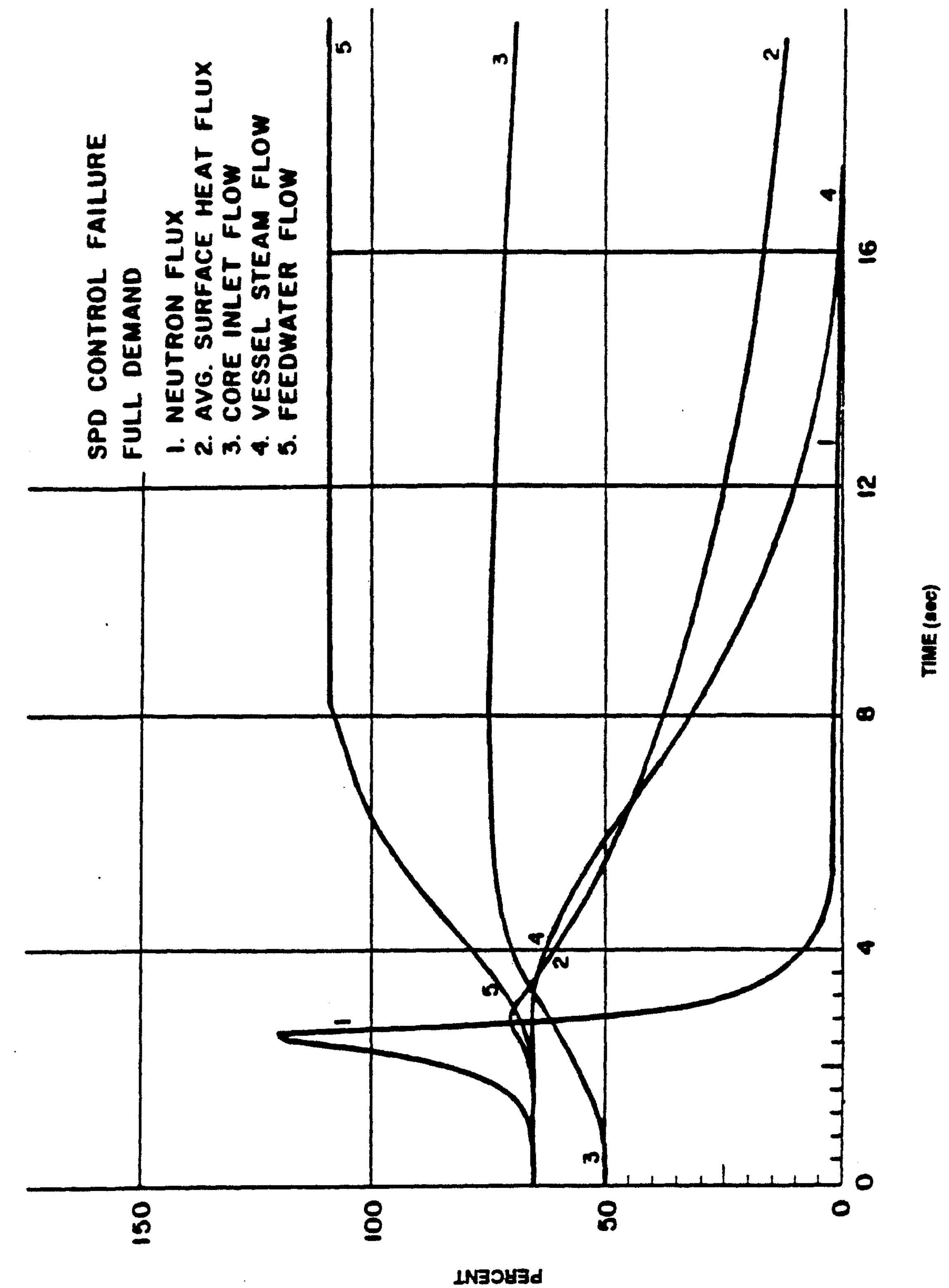
(INITIAL CORE) COLD LOOP STARTUP
TRANSIENT ANALYSIS

FIGURE 15.4-1

REVISION 4, APRIL 1997



QUAD CITIES STATION UNITS 1 & 2
(INITIAL CORE) COLD LOOP STARTUP TRANSIENT ANALYSIS
FIGURE 15.4-2 REVISION 4, APRIL 1997

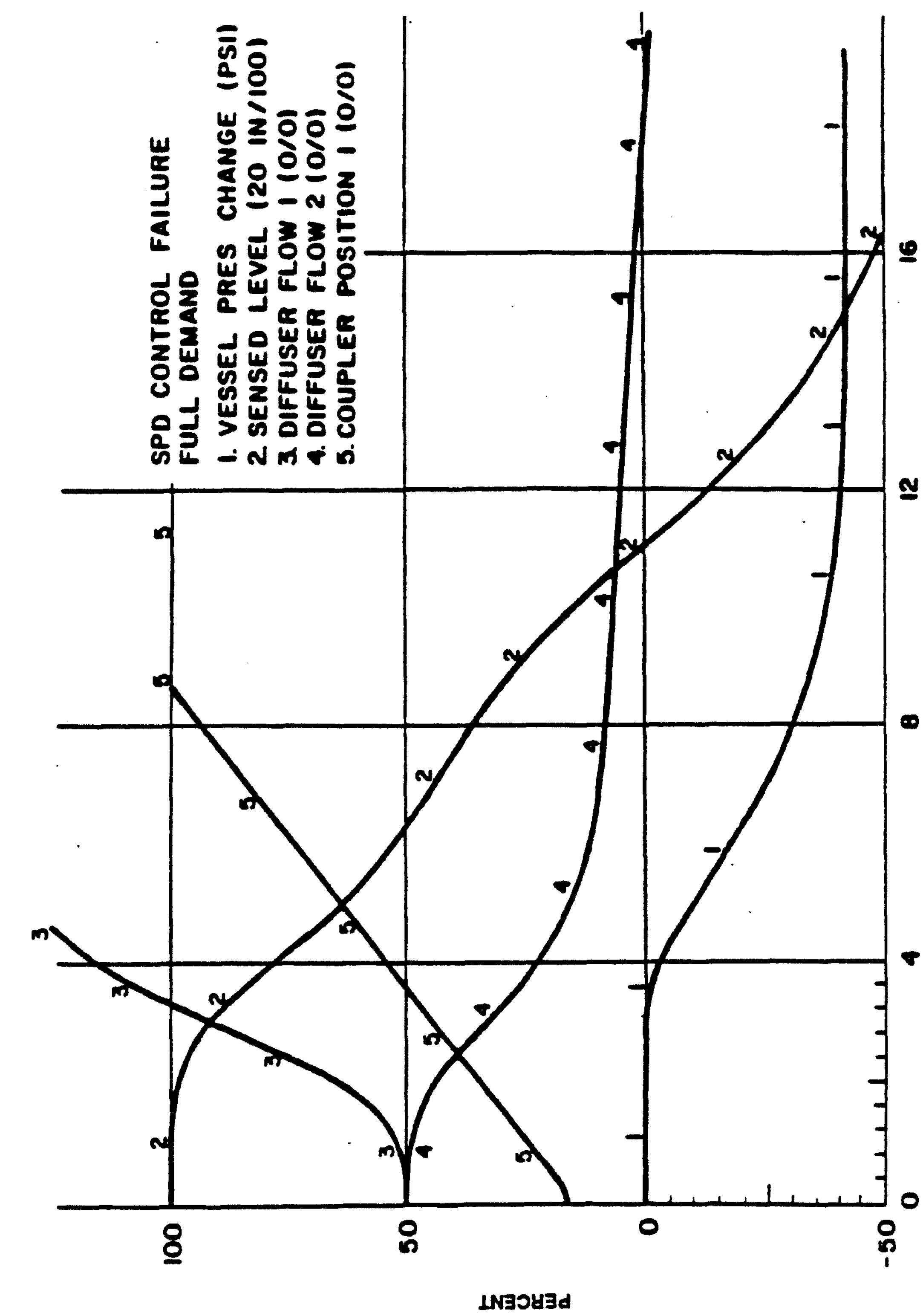


QUAD CITIES STATION
UNITS 1 & 2

(INITIAL CORE) SPEED CONTROL
FAILURE, FULL DEMAND
TRANSIENT ANALYSIS

FIGURE 15.4-3

REVISION 4, APRIL 1997

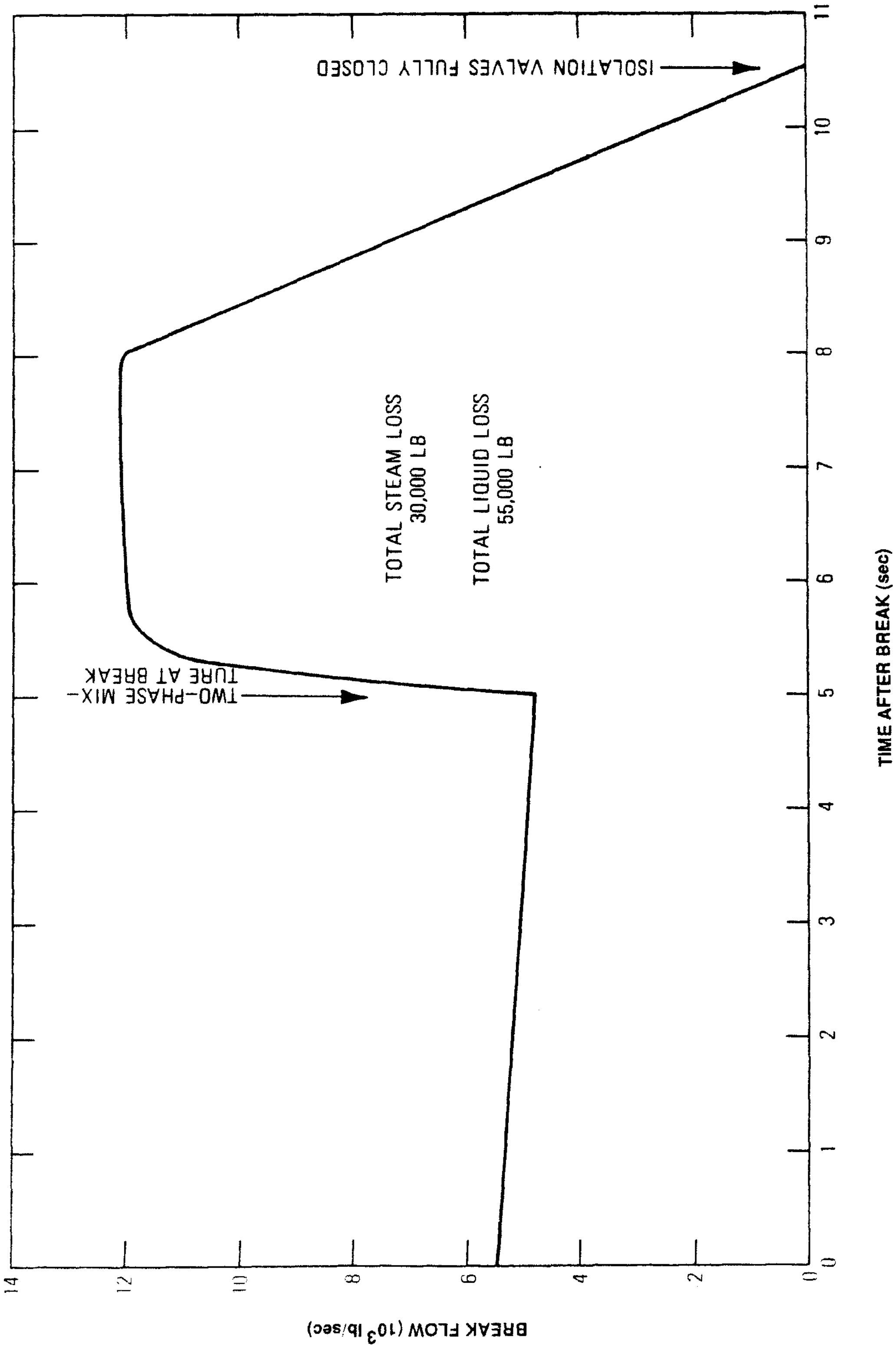


QUAD CITIES STATION
UNITS 1 & 2

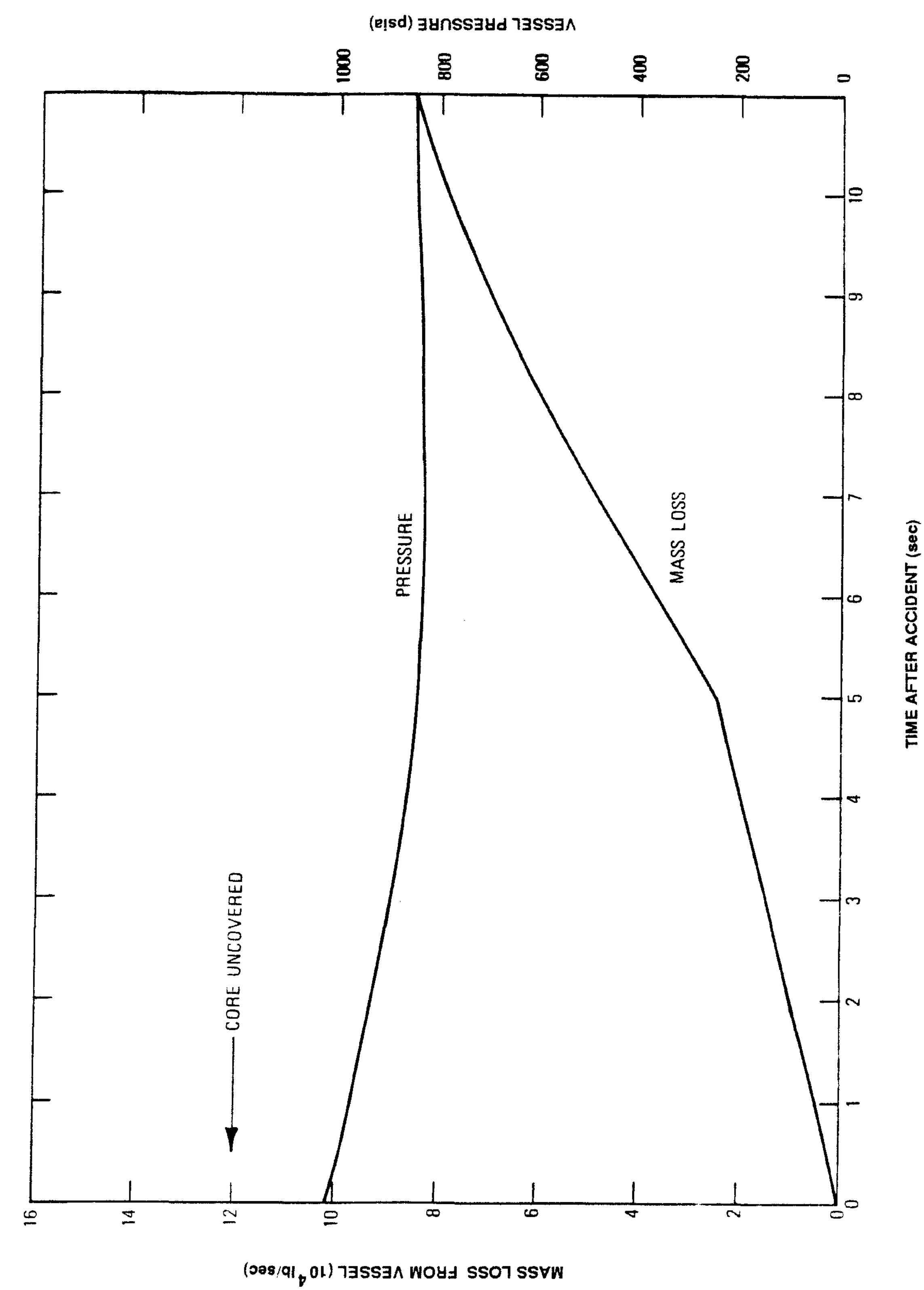
(INITIAL CORE) SPEED CONTROL
FAILURE, FULL DEMAND
TRANSIENT ANALYSIS

FIGURE 15.4-4

REVISION 4, APRIL 1997



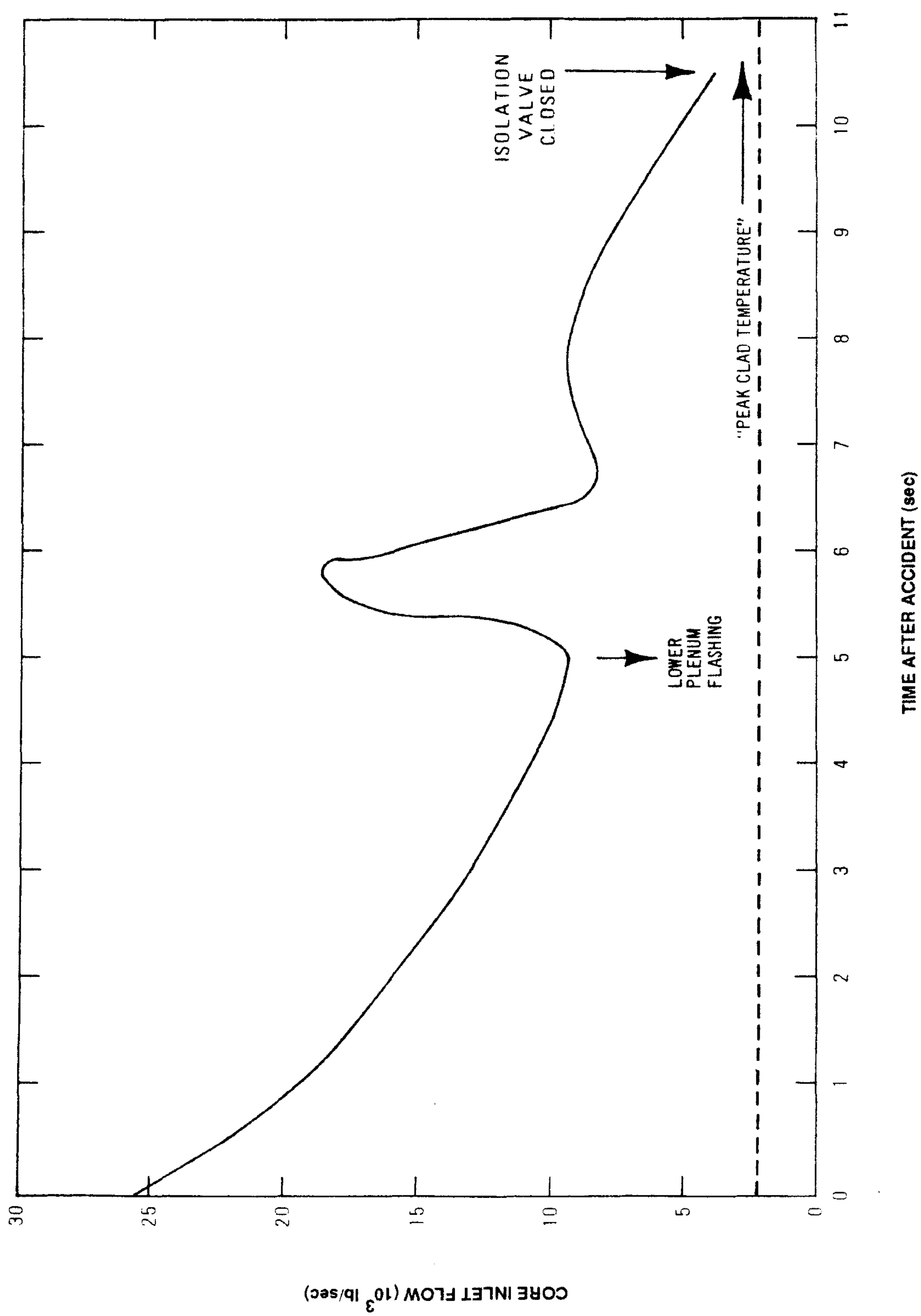
QUAD CITIES STATION UNITS 1 & 2
STEAM LINE BREAK–COOLANT LOSS
FIGURE 15.6-1



QUAD CITIES STATION
UNITS 1 & 2

STEAM LINE BREAK
REACTOR VESSEL PRESSURE AND MASS CHANGES

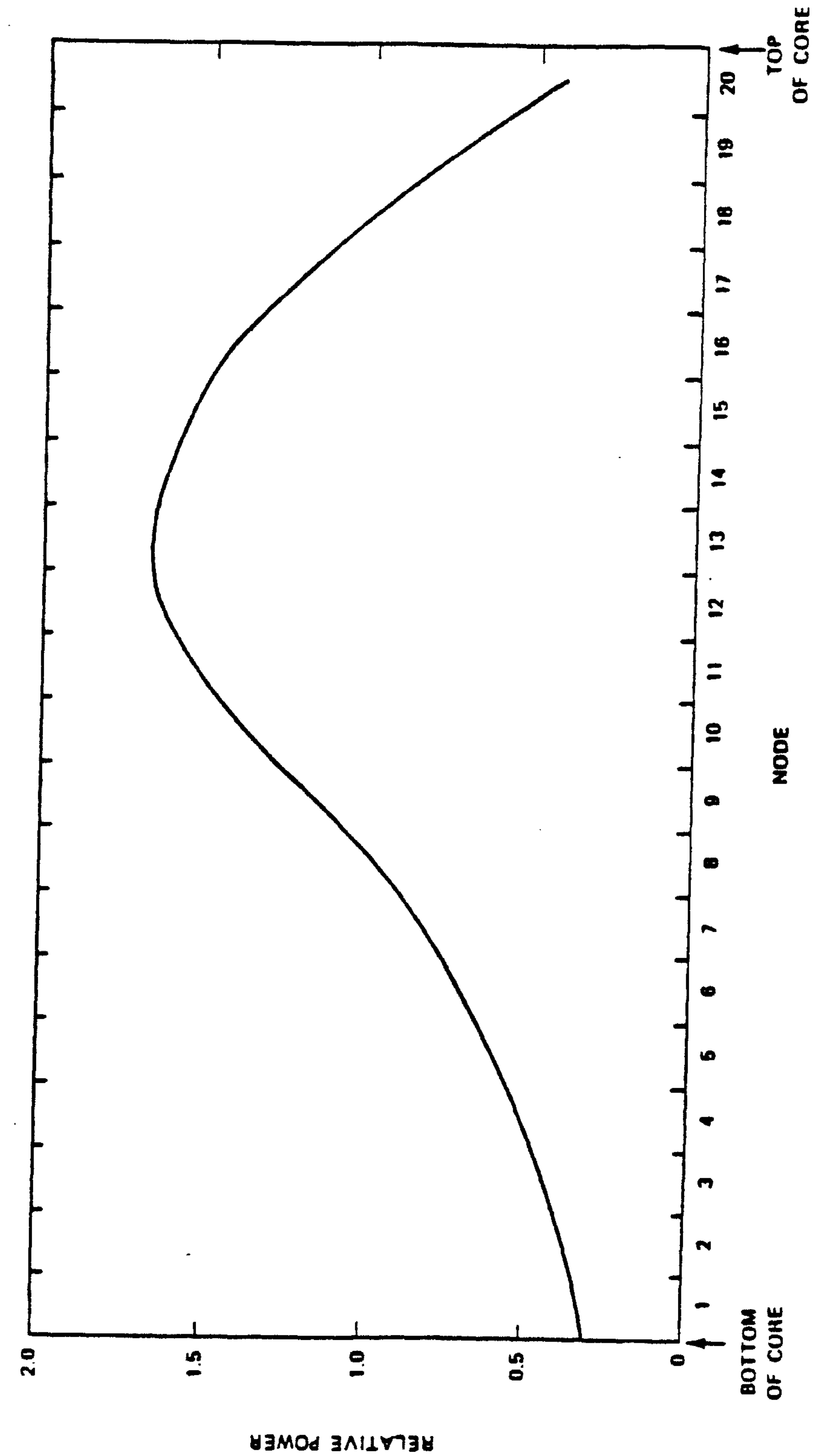
FIGURE 15.6-2



QUAD CITIES STATION
UNITS 1 & 2

STEAM LINE BREAK—CORE INLET COOLANT FLOW

FIGURE 15.6-3



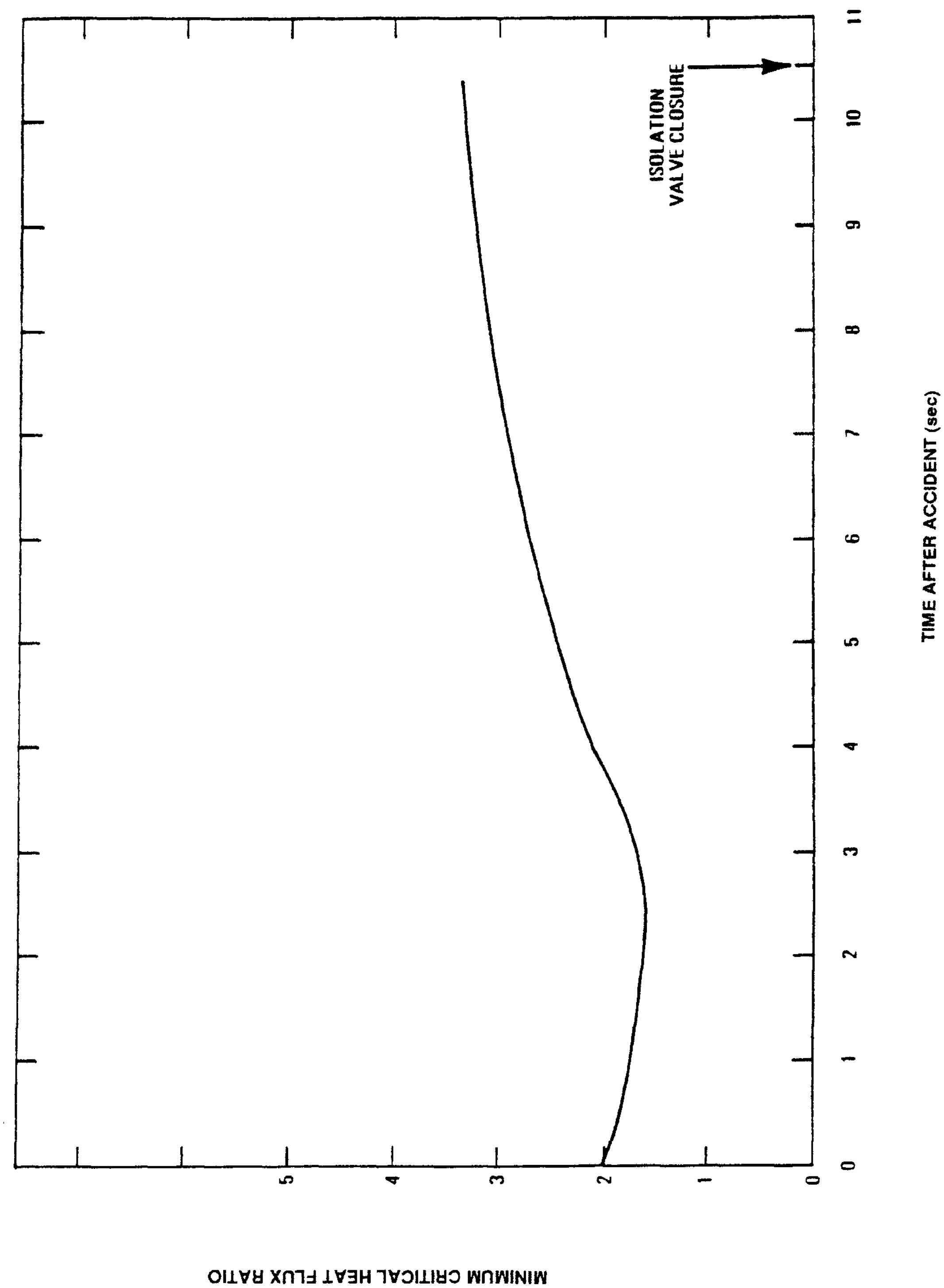
QUAD CITIES STATION

UNITS 1 & 2

(INITIAL CORE) STEAM LINE BREAK ANALYSIS, CORE AXIAL POWER DISTRIBUTION

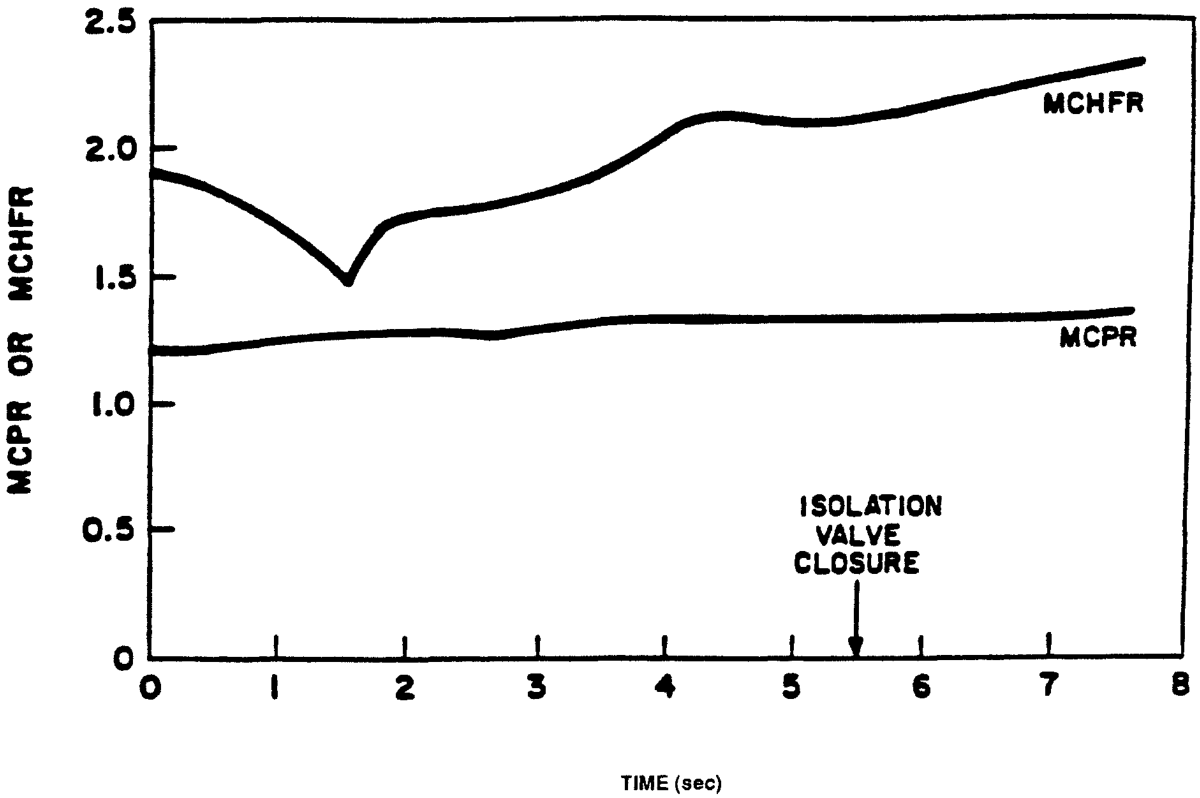
FIGURE 15.6-4

REVISION 4, APRIL 1997



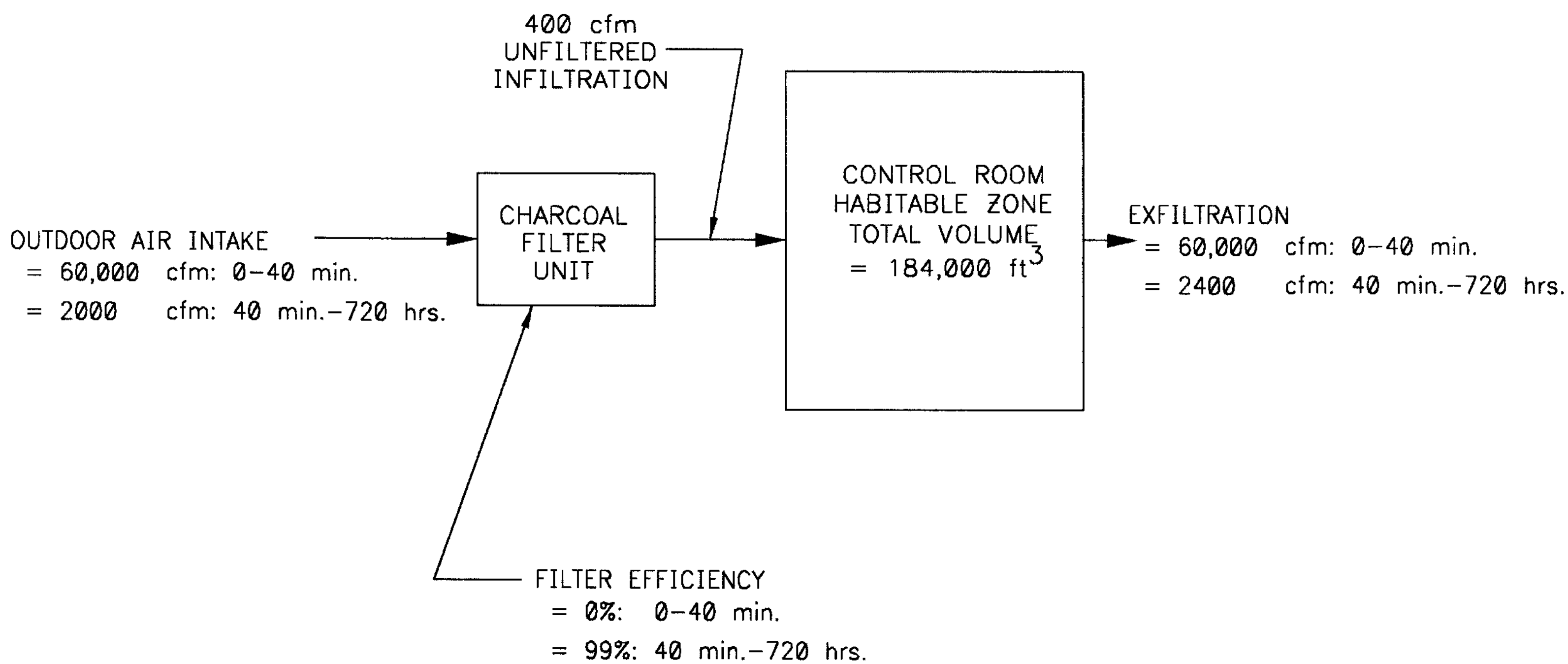
See the introduction to Section 15.6 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION UNITS 1 & 2
STEAM LINE BREAK–MCHFR RESPONSE
FIGURE 15.6-5



MINIMUM CRITICAL HEAT FLUX RATIO AND
MINIMUM CRITICAL POWER RATIO VERSUS
TIME (OUTSIDE STEAM LINE BREAK)
TYPICAL BWR/4

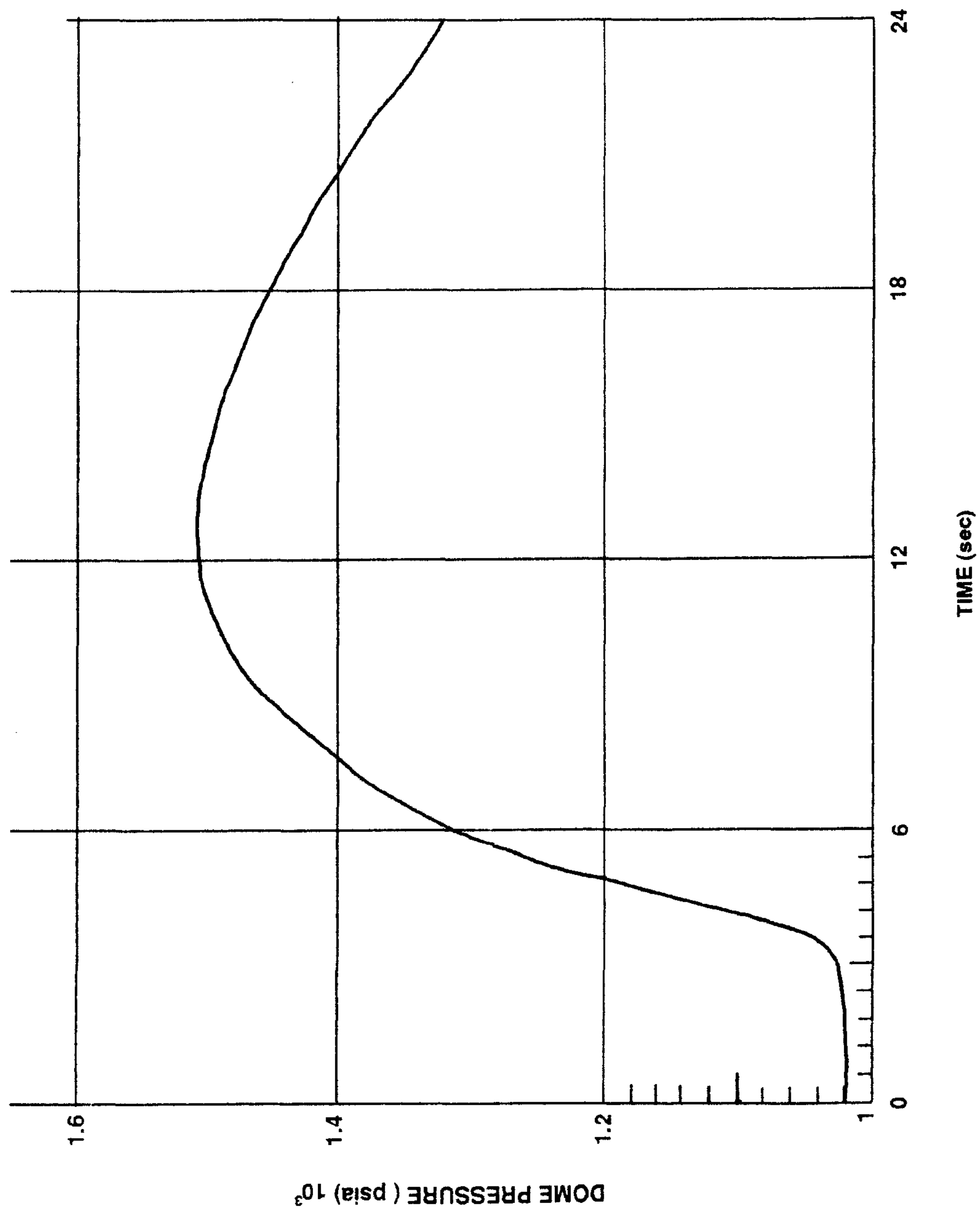
QUAD CITIES STATION UNITS 1 & 2
MINIMUM CRITICAL HEAT FLUX RATIO AND MINIMUM CRITICAL POWER RATIO VS. TIME (OUTSIDE STEAM LINE BREAK) TYPICAL BWR/4
FIGURE 15.6-6



QUAD CITIES STATION
UNITS 1 & 2

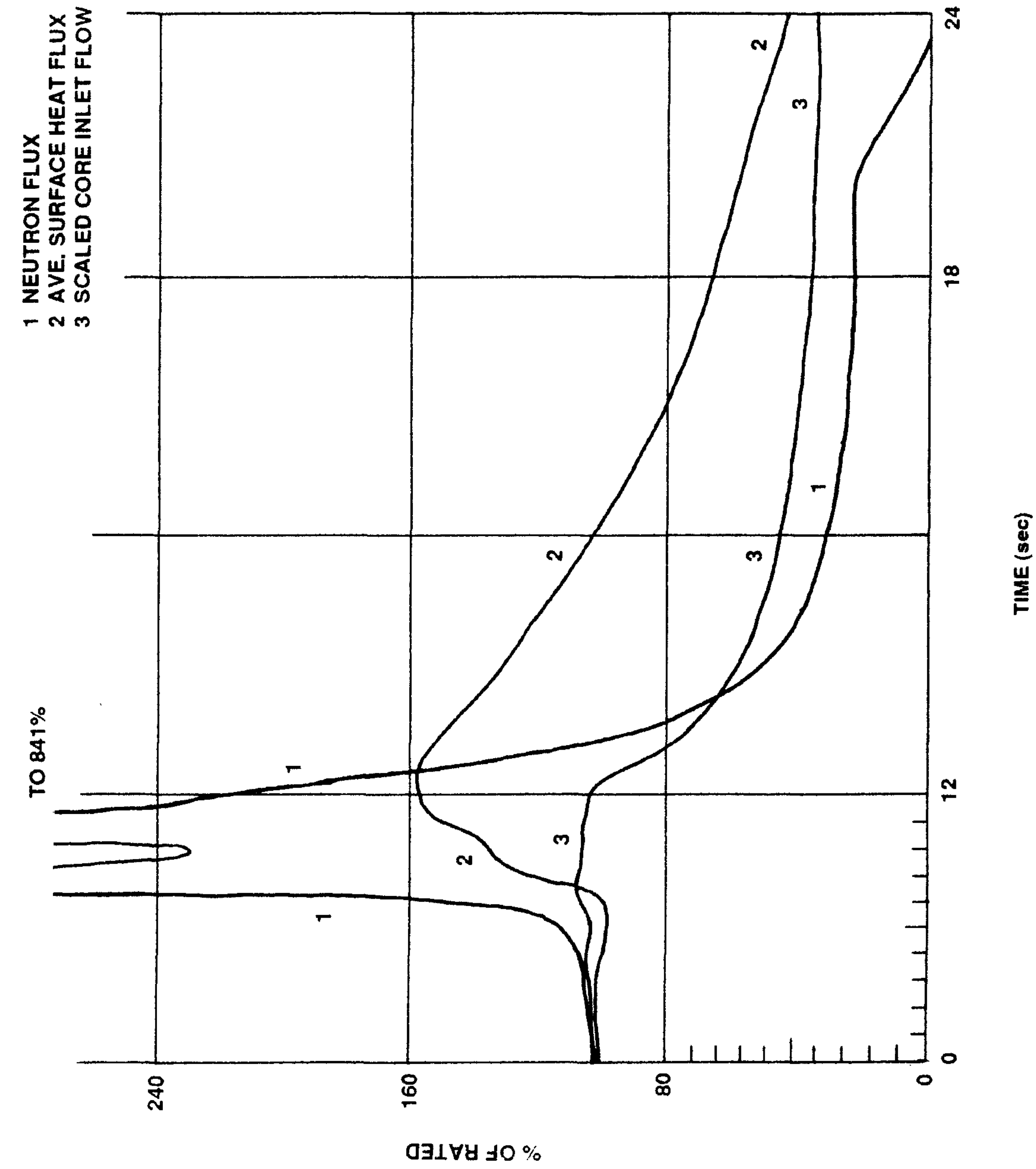
CONTROL ROOM DOSE ANALYSIS MODEL

FIGURE 15.6-7
REVISION 9, OCTOBER 2007



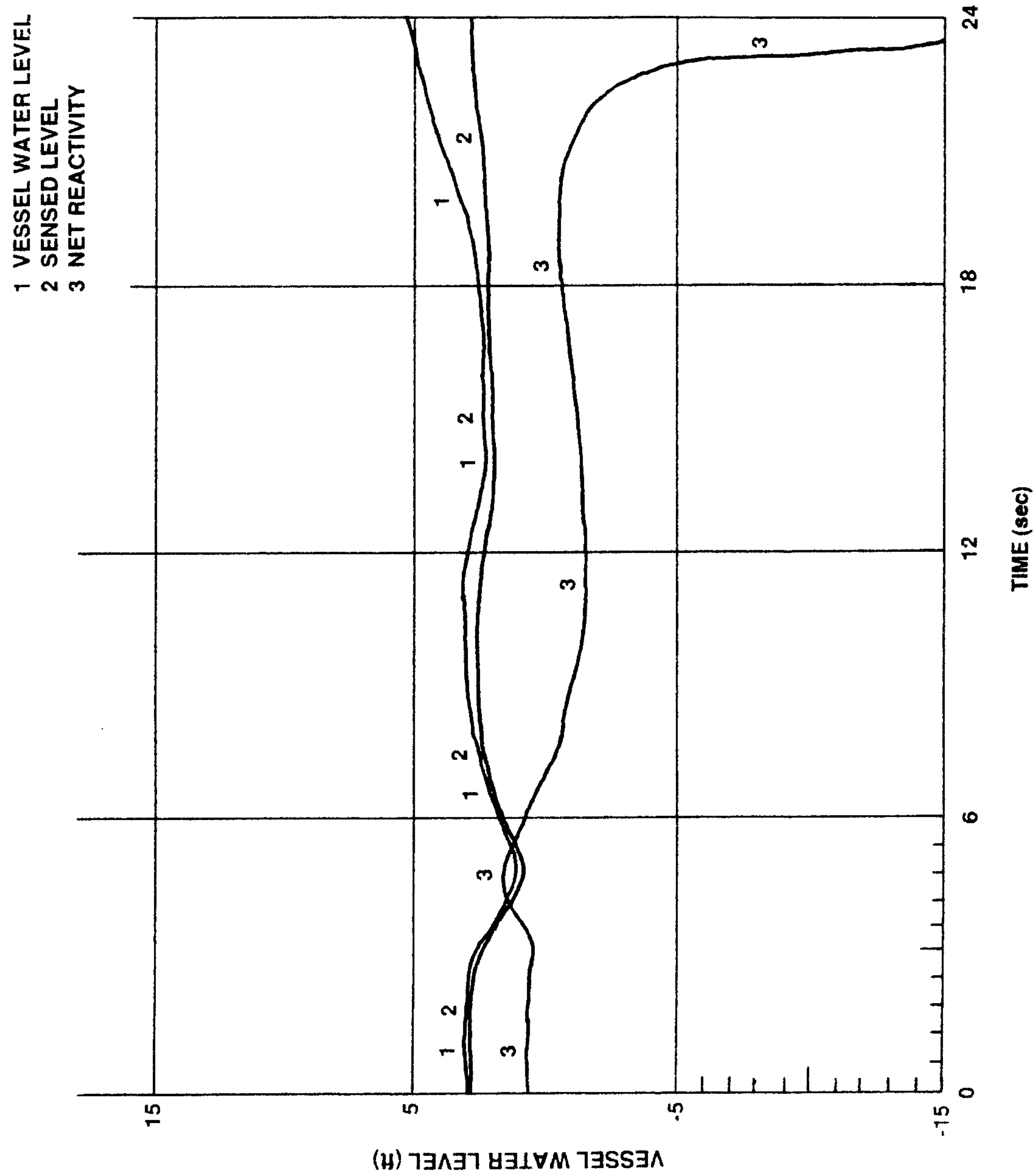
See the introduction to Section 15.8 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION UNITS 1 & 2
ATWS MSIV CLOSURE TRANSIENT: REACTOR PRESSURE RESPONSE
FIGURE 15.8-1



See the introduction to Section 15.8 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION UNITS 1 & 2
ATWS - MSIV CLOSURE TRANSIENT: NEUTRON FLUX RESPONSE
FIGURE 15.8-2



See the introduction to Section 15.8 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION UNITS 1 & 2
ATWS - MSIV CLOSURE TRANSIENT REACTOR LEVEL RESPONSE
FIGURE 15.8-3