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Integrated Systems Analysis of the PIUS Reactor

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ABSTRACT

Results are presented of a systems failure analysis of the PIUS plant systems that are used during normal reactor operation and postulated accidents. This study was performed to provide the NRC with an understanding of the behavior of the plant. The study applied two diverse failure identification methods, Failure Modes Effects & Criticality Analysis (FMECA) and Hazards & Operability (HAZOP) to the plant systems, supported by several deterministic analyses. Conventional PRA methods were also used along with a scheme for classifying events by initiator frequency and combinations of failures. Principal results of this study are: (a) an extensive listing of potential event sequences, grouped in categories that can be used by the NRC, (b) identification of support systems that are important to safety, and (c) identification of key operator actions.

CONTENTS

	Page
Abstract	iii
List of Figures	ix
List of Tables	xv
Executive Summary	xix
Acknowledgements	xx
Table of Definitions	xxi
 1.0 INTRODUCTION	 1-1
 2.0 REACTOR OVERVIEW	 2-1
2.1 General Description	2-1
2.2 Key PIUS 600 Characteristics	2-1
2.3 Simplified PIUS Operations	2-3
2.3.1 Hot/Cold Shutdown	2-3
2.3.2 Startup	2-3
2.3.3 Power Ascent	2-3
2.3.4 Controlled Shutdown	2-5
2.3.5 Pressurization of the PIUS Primary Loop	2-5
2.3.6 Reactivity Control of PIUS	2-5
2.3.7 Active Scram of PIUS	2-5
2.3.8 Passive Scram of PIUS	2-5
2.4 PIUS Upset Scenarios	2-5
2.4.1 Upset Scenario 1: Overpower	2-7
2.4.2 Upset Scenario 2: Excess Heat Removal	2-7
2.4.3 Upset Scenario 3: Hot Leg LOCA	2-7
2.4.4 Upset Scenario 4: Cold Leg LOCA	2-7
2.4.5 Upset Scenario 5: Station Blackout	2-7
2.4.6 Upset Scenario 6: Anticipated Transient Without Active Scram	2-7
2.4.7 Upset Scenario 7: Loss of Heat Sink	2-7
2.5 Fuel Handling Equipment	2-7
2.6 Reactor Parameters	2-8
2.7 System Identifications and Relationships	2-8
 3.0 METHODOLOGY	 3-1
3.1 Methodology Overview	3-1
3.2 Description of Failure Modes Effects and Criticality Analysis (FMECA)	3-3
3.2.1 Qualifications and Information Needs	3-3
3.2.2 Contents of a FMECA	3-3
3.2.3 Depth of a FMECA	3-3
3.2.4 FMECA Preparation	3-4
3.2.5 Practical Completeness	3-5
3.3 Description of Hazards and Operability Analysis (HAZOP)	3-6
3.3.1 Overview of HAZOP	3-6
3.3.2 Composition of the HAZOP Team	3-7
3.3.3 Recording the Session	3-8
3.3.4 Identification of the Operating Mode	3-8

CONTENTS (Continued)

	Page
3.3.5	Process Variable Selection 3-9
3.3.6	System (Node) Selection 3-9
3.3.7	Design Intent 3-9
3.3.8	Guide Words and Causes 3-10
3.3.9	Consequence Identification 3-11
3.3.10	Resolution 3-12
3.3.11	Review 3-12
3.3.12	Record Keeping 3-12
3.4	Overview of Deterministic Analyses 3-12
3.4.1	Reactor Physics/Thermal Hydraulics 3-12
3.4.2	Materials Properties and Chemistry 3-13
3.4.3	Structural Analysis 3-13
3.5	Approach to Assembly of Results 3-14
3.6	Approach to System Dependency Analysis 3-15
3.7	NRC Event Condition Analysis 3-16
3.8	References 3-16
4.0	FAILURE MODES EFFECTS AND CRITICALITY ANALYSIS 4-1
4.1	Procedures for the PIUS FMECA 4-1
4.2	Major Observations from the PIUS FMECA 4-1
5.0	DESCRIPTION OF THE HAZOP APPLICATION 5-1
5.1	Adaptations of HAZOP Methodology to the Study of the PIUS System 5-1
5.1.1	Technical Adaptations 5-1
5.1.2	Programmatic Adaptations 5-1
5.2	HAZOP of PIUS Plant During Power Operation 5-2
5.3	HAZOP of Startup Procedure 5-2
5.3.1	Application of HAZOP to Analysis of Procedures 5-2
5.4	Overview of Results 5-3
5.4.1	The Q-List Problem 5-3
5.4.2	Depressurization Transients 5-3
5.4.3	Behavior of Boron Concentration 5-4
5.4.4	Structural Excitations 5-4
5.4.5	Communication Between Control Volumes 5-4
5.4.6	French Scenario at Start-Up 5-4
5.5	References 5-4
6.0	SUPPORTING ANALYSES 6-1
6.1	Thermal-Fluid Flow Analyses 6-1
6.2	Evaluation of Design Basis External Events 6-2
6.2.1	Site Characteristics 6-2
6.2.2	Seismic Event 6-3
6.2.3	High Wind/Tornado 6-14
6.2.4	Water Level/Flooding 6-14
6.2.5	Fire Protection 6-14
6.2.6	External Missiles 6-16

CONTENTS (Continued)

	Page
6.2.7	Summary of External Event Evaluations 6-17
6.3	Materials Evaluations 6-17
6.3.1	Stainless Steel Pool Liner 6-18
6.3.2	Carbon Steel Membrane 6-19
6.3.3	Gas Lock Cover 6-21
6.3.4	Wet Thermal Insulation 6-21
6.3.5	Reactor Internals (Core Region) 6-23
6.3.6	PCRVR Tendons and Rebars 6-25
6.3.7	Secondary Side Piping 6-27
6.3.8	Primary Side Carbon- and Low-Alloy Steel Components 6-29
6.3.9	Summary of Material Evaluations 6-29
6.3.10	References 6-32
7.	EVENT TREE ANALYSIS 7-1
7.1	PIUS Intermediate Table 7-1
7.2	Assembly of the Initiators and Mitigators 7-11
7.3	The NRC Event Category Method 7-14
7.4	Large Primary Pipe Break (LPPB) Event Tree 7-16
7.5	Small Primary Pipe Break Event Tree 7-21
7.6	Loss of Off-Site Power 7-24
7.7	Turbine Trip 7-26
7.8	Steamline Break (SB) 7-30
7.8.1	Steamline Break Inside Containment (SBIC) 7-30
7.8.2	Steamline Break Outside Containment (SBOC) 7-34
7.9	Intentionally Left Blank
7.10	Feedwater Transients 7-37
7.11	Computer Malfunction (CM) 7-39
7.12	Severe Seismic Event 7-43
7.13	Refueling Accident (RA) 7-47
7.14	Section Summary 7-52
8.0	PIUS SYSTEM DEPENDENCIES AND SAFETY CLASSIFICATION 8-1
8.1	Introduction 8-1
8.2	System Safety Classification 8-1
8.3	System Dependencies 8-6
9.0	RESULTS 9-1
9.1	Prioritized Tables of Accident Sequences Based on NRC Event Classification 9-1
9.2	Prioritized Tables of Accident Sequences Based on Consequence Categories 9-7
9.3	Prioritized Table of Accident Initiators 9-10
9.4	Systems Important to Safety 9-12
9.5	Key Operator Actions 9-13
9.6	Summary of Results from FMECA, HAZOP, and Supporting Analyses 9-15
9.7	Maintenance, Test, and Inspection Observation 9-18
9.7.1	Maintenance Observations 9-18
9.7.2	Test and Inspection Observations 9-19

CONTENTS (Continued)

	Page
9.8 Lessons-Learned in the Study and Considerations for Tests	9-21
9.9 Limitations Imposed by Lack of Engineering Detail	9-22
APPENDIX A: SYSTEMS DESCRIPTIONS AND FMECA ANALYSES	A-1
A.1 Pressurizer Systems 211 and 328	A.1-1
A.2 Reactor Internals, Reactor Fuel and Instruments Systems 213, 215, 261, 216, 217, 291	A.2-1
A.3 Reactivity Control System 312	A.3-1
A.4 Primary Recirculation System (System 313)	A.4-1
A.5 Pressure Relief and Condensation Systems	A.5-1
A.6 Containment	A.6-1
A.7 Control and Monitoring Systems	A.7-1
A.8 Plant and Turbine Control Systems	A.8-1
A.9 Electric Power	A.9-1
A.10 Cooling	A.10-1
APPENDIX B: HAZOP SESSION LOGSHEETS	B-1
APPENDIX C: SYSTEM INTER-RELATIONSHIPS AND SAFETY CLASSIFICATIONS	C-1
APPENDIX D: SCRAM	D-1
APPENDIX E: THERMAL-FLUID FLOW ANALYSIS, USING THE PIPA CODE	E-1

LIST OF FIGURES

	Page
Figure 2-1	Principal Features of PIUS 600 2-2
Figure 2-2	Hot/Cold Shutdown of PIUS 2-4
Figure 2-3	Startup 2-6
Figure 2-4	Controlled Shutdown 2-6
Figure 3-1	Diagram of BNL Methodology Applied to PIUS-600 3-2
Figure 3-2	Tabular Form of the FMECA 3-4
Figure 3-3	Initiator Matrix 3-6
Figure 3-4	The HAZOP Process 3-9
Figure 3-5	Assembling Intermediate Results 3-15
Figure 3-6	Illustrative Event Tree for an Accident Initiator 3-15
Figure 6-3-1	Inner Venting System of Concrete Vessel (System 1591) 6-20
Figure 6-3-2	Reactor Internals (System 213) 6-22
Figure 6-3-3	Wet Thermal Insulation (ABB, 1991) 6-24
Figure 6-3-4	PCRv Prestressing Tendons (ABB, 1991) 6-26
Figure 7-1	NRC Event Condition Methodology 7-14
Figure 7-2	Event Sequence Diagram Symbols 7-16
Figure 7-3	Large Primary Pipe Break Event Sequence Diagram 7-18
Figure 7-4	Event Tree for the Large Primary Pipe Break 7-20
Figure 7-5	Event Sequence Diagram for the Small Pipe Break Scenarios 7-23
Figure 7-6	Small Primary Pipe Break Event Tree 7-24
Figure 7-7	Loss of Off-Site Power Event Sequences 7-25
Figure 7-8	Loss of Off-Site Power Event Tree 7-26
Figure 7-9	Turbine Trip Event Sequence Diagram 7-28
Figure 7-10	Turbine Trip Event Tree 7-29

LIST OF FIGURES (Continued)

	Page
Figure 7-11 Steamline Break inside of Containment Event Sequence Diagram	7-30
Figure 7-12 Event Tree for Steam Line Break Inside Containment	7-33
Figure 7-13 Event Sequence Diagram for Steam Line Break Outside of Containment	7-34
Figure 7-14 Steamline Break Outside of Containment Event Tree	7-36
Figure 7-15 Feedwater Transient Event Sequence Diagram	7-37
Figure 7-16 Event Tree for Feedwater Transient	7-38
Figure 7-17 Computer Malfunction Event Sequence Diagram	7-39
Figure 7-18 Computer Malfunction Event Tree	7-41
Figure 7-19 Postulated Seismic Motion	7-43
Figure 7-20 Severe Seismic-Event Event Sequence Diagram	7-44
Figure 7-21 Severe Seismic Event Tree	7-45
Figure 7-22 PIUS Refueling Structures	7-47
Figure 7-23 Refueling Flow Blockage Event Sequence Diagram	7-48
Figure 7-24 Flow Blockage during Refueling Event Tree	7-51
Figure 8-1 PIUS 600 Major System Dependencies	8-8
Figure A.1-1 Reactor Upper Structure	A.1-2
Figure A.1-2 Anchorage Flange	A.1-3
Figure A.2-1 PIUS Reactor Assembly	A.2-2
Figure A.2-2 Core Bottom and Side Support Structures	A.2-4
Figure A.2-3 Upper Reactor Riser	A.2-6
Figure A.2-4 Typical Envelope Flange	A.2-7
Figure A.2-5 Upper Riser Structure	A.2-10
Figure A.2-6 Expansion Bellows	A.2-11

LIST OF FIGURES (Continued)

	Page
Figure A.2-7 Fuel Assembly	A.2-15
Figure A.2-8 Internal Instrument Supports	A.2-18
Figure A.3-1 Filtered Water Supply System	A.3-2
Figure A.3-2 Distilled Water Supply System	A.3-2
Figure A.3-3 Borated Water Supply System	A.3-3
Figure A.3-4 Inlet and Outlet System	A.3-6
Figure A.4-1 Recirculation System	A.4-2
Figure A.5-1 Pressure Relief System	A.5-2
Figure A.5-2 Condensation System	A.5-4
Figure A.6-1 Cross Section Through the Containment	A.6-3
Figure A.6-2 Detailed Cut Through the Reactor Building	A.6-4
Figure A.7-1 Alarm Display System	A.7-3
Figure A.7-2 Main Process Computers Block Diagram	A.7-6
Figure A.7-3 Core Management Computer System Block Diagram	A.7-8
Figure A.7-4 Main Coolant Pump Control System	A.7-13
Figure A.7-5 Reactor Power Control System	A.7-22
Figure A.9-1 Electric Power Systems	A.9-3
Figure A.10-1 Diesel-Backed Service Water System	A.10-1
Figure A.10-2 Non-Diesel Backed Service Water	A.10-3
Figure A.10-3 Diesel-Backed Closed Cooling System for the Reactor Pool	A.10-5
Figure A.10-4 Diesel-Backed Closed Cooling System for Components	A.10-8
Figure A.10-5 Non-Diesel Backed Closed Cooling System	A.10-10
Figure A.10-6 Passive Closed Cooling System for the Reactor Pool	A.10-12

LIST OF FIGURES (Continued)

	Page
Figure A.10-7 Demineralized Water System	A.10-15
Figure A.10-8 Demineralized Water Distribution System	A.10-16
Figure A.10-9 High Pressure Distilled Water System	A.10-18
Figure A.10-10 Containment Ventilation System	A.10-20
Figure A.10-11 Chemical Systems Building Ventilation System	A.10-24
Figure E-1 Plant Schematic of PIUS 600 Design	E-6
Figure E-2 PIPA Representation of PIUS Primary System	E-7
Figure E-3 Average Pump Speed Response to a PIUS Full Scram	E-8
Figure E-4 Core Inlet and Hot Leg Inlet Flow Response to a PIUS Full Scram	E-9
Figure E-5 Lower and Upper Density Lock Flow Response to a PIUS Full Scram	E-10
Figure E-6 Core Average Boron Concentration Response to a PIUS Full Scram	E-11
Figure E-7 Reactor Power Response to a PIUS Full Scram	E-12
Figure E-8 Reactivity Behavior during a PIUS Full Scram	E-13
Figure E-9 PIPA vs. RIGEL (ABB Atom) for Reactor Power During a Full Scram	E-14
Figure E-10 PIPA vs. RIGEL for Boron Concentration During a Full Scram	E-15
Figure E-11 PIPA vs. RIGEL for Core Flow During a Full Scram	E-16
Figure E-12 PIPA vs. RIGEL for Lower Density Lock Flow during a Full Scram	E-17
Figure E-13 Effect of Boron Mixing on Reactor Power During a Full Scram	E-18
Figure E-14 Primary System Pressure Response to a Loss of Heat Sink Event	E-19
Figure E-15 Pump speed Response to a Loss of Heat Sink Event	E-20
Figure E-16 Lower Density Lock Level Response to a Loss of Heat Sink Event	E-21
Figure E-17 Reactor Power Response to a Loss of Heat Sink Event	E-22
Figure E-18 Density Locks Flow Response to a Loss of Heat Sink Event	E-23

LIST OF FIGURES (Continued)

	Page
Figure E-19	Boron Concentration Response to a Loss of Heat Sink Event E-24
Figure E-20	Boron Concentration Response to an Uncontrolled Boron Dilution Event E-25
Figure E-21	Reactor Power Response to an Uncontrolled Boron Dilution Event E-26
Figure E-22	Reactivity Behavior During an Uncontrolled Boron Dilution E-27
Figure E-23	Pump Speed Response to an Uncontrolled Boron Dilution E-28
Figure E-24	Primary System Pressure Response to an Uncontrolled Boron Dilution Event E-29
Figure E-25	Primary System Pressure Response to an IORV Event E-30
Figure E-26	Pump Speed Response to an IORV Event E-31
Figure E-27	Primary Loop Flow Response to an IORV Event E-32
Figure E-28	Density Locks Flow Response to an IORV Event E-33
Figure E-29	Boron Concentration Response to an IORV Event E-34
Figure E-30	Reactor Power Response to an IORV Event E-35
Figure E-31	Reactivity Behavior During an IORV Event E-36

LIST OF TABLES

		Page
Table 2-1	PIUS 600 Parameters with PWR and BWR Comparisons	2-9
Table 2-2	Top-Level PIUS Structures and Systems	2-11
Table 2-3	PIUS System Identification	2-12
Table 3-1	Consequence Categories	3-14
Table 6.2-1	Horizontal Vibration Modes	6-13
Table 6.2-2	Vertical Vibration Mode	6-13
Table 6.3.1	Plants Experiencing Wall Thinning in the Feedwater Condensate System (Wu, 1989)	6-28
Table 6.3.2	Summary of Corrosion of Carbon Steel by Boric Acid	6-30
Table 6.3.3	Summary of Possible Materials Related Safety Issues	6-32
Table 7-1	PIUS Interim Table (PIT)	7-2
Table 7-2A	PIUS Initiators	7-11
Table 2-B	Major PIUS Mitigators	7-12
Table 7-3	NRCEC Mitigator Failure Classes for Use in the NRCEC Methodology	7-13
Table 7-4	NRC Event Categories	7-15
Table 7-5	NRCEC Classification of Sequences by Initiators	7-15
Table 7-6	Large Pipe Break Event Table	7-19
Table 7-7	BETA Results for PIUS Large Pipe Break Scenarios	7-20
Table 7-8	Small Pipe Break Events	7-22
Table 7-9	BETA Results for PIUS Small Pipe Break Scenarios	7-23
Table 7-10	Loss of Offsite Power Event Table	7-25
Table 7-11	BETA Results for PIUS Loss of Off-Site Power	7-25
Table 7-12	Turbine Trip Events	7-27
Table 7-13	BETA Results for PIUS Turbine Trip Event Tree	7-29

LIST OF TABLES (Continued)

	Page
Table 7-14	Event Table for Steamline Break Inside Containment 7-32
Table 7-15	BETA Results for PIUS Steamline Break Inside Containment 7-33
Table 7-16	Event Table for Steamline Break Outside Containment 7-35
Table 7-17	BETA Results for PIUS Steamline Break Outside Containment 7-36
Table 7-18	Feedwater Transients Event Table 7-38
Table 7-19	BETA Results for PIUS Feedwater Transients 7-38
Table 7-20	Computer Malfunction Event Table 7-41
Table 7-21	BETA Results for PIUS Computer Malfunction 7-42
Table 7-22	PIUS Severe Seismic Events 7-44
Table 7-23	BETA Results for PIUS Severe Seismic Event 7-46
Table 7-24	PIUS Flow Blockage during Refueling Event Table 7-51
Table 7-25	BETA Results for Flow Blockage during Refueling 7-52
Table 7-26	Summary Frequency Categories from Event Tree Analysis of PIUS 7-52
Table 8-1	Summary of PIUS Safety Classifications 8-1
Table 8-2	Safety Classification of PIUS Structures, Components and Systems 8-3
Table 8-3	PIUS 600 Major System Dependencies 8-7
Table 9-1	Initiator Identification 9-3
Table 9-2	Top Event Identification 9-4
Table 9-3	Decreasing Frequency Ordering of NRCEC Accident Classifications for EC-3 9-5
Table 9-4	Decreasing Frequency Ordering of NRCEC Accident Classifications for EC-2 9-6
Table 9-5	Decreasing Frequency Ordering of NRCEC Accident Classifications for EC-1 9-6
Table 9-6	Decreasing Frequency Ordering of Accident of Sequences Leading to CC-3 9-8
Table 9-7	Decreasing Frequency Ordering of Accident of Sequences Leading to CC-2 9-9

LIST OF TABLES (Continued)

	Page
Table 9-8	Order of Initiators by Frequency of Occurrence 9-11
Table 9-9	Initiators Ordered by Associated CC-3 Frequency 9-11
Table 9-10	Initiators Ordered by Associated CC-2 Frequency 9-12
Table 9-11	Top Events of ET Analysis Ordered by Importance to Safety (CC-3 Sequences) 9-13
Table 9-12	Operator Response for Performing Mitigation 9-14
Table 9-13	Summary of Major Results from FMECA, HAZOP, and Supporting Deterministic Analyses 9-16
Table 9-14	Table of Inservice Inspection Observations 9-20
Table A.1-1	PIUS 600 Failure Modes Effects and Criticality Analysis A.1-8
Table A.2-1	Fuel Assembly Data A.2-14
Table A.2-2	Comparison of PIUS and PWR Fuel Rod Operating Data A.2-16
Table A.2-3	PIUS 600 Failure Modes and Effects and Criticality Analysis A.2-21
Table A.3-1	PIUS 600 Failure Modes Effects and Criticality Analysis A.3-8
Table A.4-1	PIUS 600 Failure Modes and Effects and Criticality Analysis A.4-7
Table A.5-1	PIUS 600 Failure Modes and Effects and Criticality Analysis A.5-6
Table A.6-1	PIUS 600 Failure Modes and Effects and Criticality Analysis A.6-7
Table A.7-1	PIUS 600 Failure Modes and Effects and Criticality Analysis A.7-29
Table A.8-1	Relating Modes, Conditions and Systems A.8-2
Table A.8-1	PIUS 600 Failure Modes and Effects and Criticality Analysis A.8-5
Table A.9-1	PIUS 600 Failure Modes and Effects and Criticality Analysis A.9-9
Table A.10-1	PIUS 600 Failure Modes and Effects and Criticality Analysis A.10-26
Table C-1	Summary of PIUS Safety Classifications C-1
Table C-2	Safety Classifications of PIUS Structures, Components and Systems C-3
Table C-3	Systems with Identified Dependencies C-6

LIST OF TABLES (Continued)

	Page
Table C-4 Systems with No Identified Dependencies	C-23
Table D-1 Summary of Failure Probabilities Scram	D-3

EXECUTIVE SUMMARY

The PIUS (Process Inherent Ultimate Safety) reactor is an advanced design nuclear power plant that uses passive safety features and basic physical processes to perform safety functions. The PIUS reactor was designed by ABB-Atom over the past decade. ABB has submitted a Preliminary Safety Information Document (PSID) to the NRC to provide the basic information necessary for a preapplication review of the conceptual design of PIUS. Brookhaven National Laboratory (BNL) performed a detailed analysis of the PIUS reactor, as documented in this report, to assist the NRC in their preapplication review of PIUS.

The study of PIUS consists of basically two main portions, a component level analysis and an integrated systems analysis. These two portions were performed by multi-disciplinary teams and were also completed using several methods, in order to obtain more complete results. Also, some scoping level thermal-hydraulic analyses were performed in support of these efforts. For the component level analyses, a Failure Modes, Effects, and Criticality Analysis (FMECA) and a number of deterministic analyses (e.g. structural, and corrosion-related) were completed. For the integrated analyses, BNL performed a Hazards & Operability (HAZOP) Analysis and an event tree analysis. HAZOP is a method often used in the chemical industry for analyzing system performance. The event tree analysis was performed using probabilistic safety assessment techniques. The resulting accident sequences were analyzed using the NRC Event Category (EC) method, which is based on initiator frequency and combinations of failures. In a simplified fashion, EC-1 sequences are Abnormal Operational Occurrences (AOOs), EC-2 sequences are Design Basis Accidents (DBAs), EC-3 sequences are Severe Accidents, and EC-4 sequences are Residual Risk related sequences. Accident sequences were also placed into broad consequence categories based on the amount of radioactive material that could be released, using engineering judgement. The final results of the various analyses were of a qualitative rather than a quantitative nature. The large majority of the sequences were in the Residual Risk (EC-4) category.

Principal results of this study are:

- (a) An extensive listing of potential event sequences, grouped in categories that can be used by the NRC for selecting AOOs, DBAs, and Severe Accidents that must be analyzed for design certification.
- (b) Identification of systems that are important to safety in the mitigation of accident initiators.
- (c) Identification of key operator actions that pertain to mitigation of accident initiators or accident management strategies.

One noteworthy observation to emerge from the various analyses is the importance (and the apparent reliability) of the passive scram function of PIUS. Also, however, the study identified a lack of diversity and redundancy in the scram function. During the time of the BNL review, ABB continued to develop the PIUS design and in fact has (subsequent to the PSID) modified and improved the scram features. Other important observations are contained in the various Tables of the results section of the report.

ACKNOWLEDGEMENTS

This inter-disciplinary effort required and received contributions from numerous persons. The integrated PIUS Study was initially led by Gregory Van Tuyle and later by Peter Kroeger. James Higgins led the preparation of the event trees portion of the analysis.

The authors are grateful to many co-workers, at Brookhaven and at the NRC, who provided invaluable assistance and support in producing this report.

Andrew Szukiewicz, our NRC Project Manager, as well as Zoltan Rosztoczy and Ralph Meyer at the NRC participated in numerous discussions and provided many valuable suggestions.

The HAZOP effort required the participation of not only most of the authors and organizers, but also of several other specialists. Bing Chan, Gregory Slovik, Charles Hofmayer, William Luckas and Cheryl Conrad made major contributions to this effort.

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TABLE OF DEFINITIONS

BETA

Brookhaven Event Tree Analyzer

Consequence Categories:

CC-1	No fuel damage. Cladding does not leak and fuel can be reused. Radioactivity in coolant may be released.
CC-2	Fuel damage occurs such that fission gases in the fuel escape; the fuel element cannot be reused because of cladding degradation or perforation. Coolable geometry is maintained and no melting occurs.
CC-3	Fuel damage occurs involving relocation, fragmentation or melting.

FMECA

Failure Modes, Effects, and Criticality Analysis

FREQUENCY CATEGORIES:

These categories are used in a number of places and sometimes are failures or occurrences per year while at other times they are failures per demand.

I	Frequent: 10 to 1
II	Anticipated: 1 to 0.1
III	Rare: 0.1 to 0.01
IV	Very Rare: 0.01 to 1E-4
V	Extremely Rare: 1E-4 to 1E-7
VI	Incredible: Less than 1E-7

HAZOP

Hazards and Operability Analysis

NRC Event Category
(EC) Methodology:

This is a method of classifying accident sequences based on the frequency of the initiator and the number of failures that occur. See Section 7.3 for the detailed definitions of each Event Category (EC-1,2,3 & 4.)

PCRV

Prestressed Concrete Reactor Vessel

PIPA

PIUS Interactive Plant Analyzer

PIT

PIUS Interim Table

PIUS

Process Inherent Ultimate Safety Reactor

1.0 INTRODUCTION

The U.S. Nuclear Regulatory Commission's (NRC) preapplication review of the Process Inherent Ultimate Safety (PIUS) reactor is being supported by an integrated systems analysis of the reactor design by Brookhaven National Laboratory.

The overall purpose of this study was to perform a systems failure analysis of the PIUS plant systems that are used during normal reactor operation and postulated accidents to provide the NRC with an understanding of the behavior of the plant.

Principal results of this study are:

- (a) An extensive listing of potential event sequences, grouped in categories that can be used by the NRC for selecting Anticipated Operational Occurrences (AOOs), Design-Basis Accidents (DBAs), and Severe Accidents that must be analyzed for design certification.
- (b) Identification of systems that are important to safety in the mitigation of accident initiators.
- (c) Identification of key operator actions that pertain to mitigation of accident initiators or accident management strategies.

A multi-disciplinary team of experts in the diverse fields of reliability analysis, human factors, reactor operation, the material sciences, thermal hydraulics modeling, reactor physics modeling and structural analysis conducted the PIUS integrated analysis. Two failure analysis methods were used, supported by several deterministic analyses.

At the completion of the study, the major results were presented to ABB-CE in a public meeting at NRC headquarters. Comments were received from ABB-CE as a result of this meeting, which included information not in the PSID, as well as information about design changes in PIUS made after the PSID was written. The study was not completely redone to account for these design changes, however Section 4 was revised extensively, considering the new information from ABB-CE. Additionally, Sections 8 and Appendix C were revised based on the vendor's comments.

The report is organized as follows. After a brief overview of the PIUS reactor in Section 2, the methodology is described in detail in Section 3. Sections 4 and 5 summarize the two failure analyses, Failure Modes Effects and Criticality Analysis (FMECA) and Hazards and Operability Analysis (HAZOP). Supporting deterministic modeling of reactor transients, material evaluations, and evaluations of external events are given in Section 6. In Section 7, the interim results from FMECA, HAZOP and the deterministic analyses are integrated, and event trees are developed. The results of the application of the NRC event category methodology are presented. In Section 8, the system dependency analysis is completed and the PIUS safety classification scheme is discussed and background information for various of the working chapters are given in several Appendices.

The results of the study are summarized in Section 9, together with its limitations.

2.0 REACTOR OVERVIEW

2.1 General Description

Figure 2-1 is an overview of the design features of PIUS. The reactor core and other primary system components that direct the coolant flow to the four primary coolant loops are located inside a pre-stressed concrete reactor vessel (PCRVR), that contains reactor pool water borated to a concentration of 2200 ppm. The walls of the PCRVR are approximately seven meters thick and contain pre-stressed steel cables (or tendons) that provide tensile strength for the vessel. The walls are lined with stainless steel and have an imbedded steel membrane to prevent the loss of the pool water. The reactor core is located in the lower region of the PCRVR to provide a mechanism for natural circulation flow between the reactor pool and the heated core region, when required to remove residual decay heat.

A density lock, located below the core region, prevents the flow of borated pool water into the primary system during normal operation. This lower density lock is a vertical grid of cells approximately one meter in length that acts as an interface between the cold pool water and the hot primary coolant. During normal operation, this interface is maintained in the density lock by controlling the main coolant pump speed. An additional density lock is located above the core region. The hot/cold water interface in this upper density lock is controlled by varying the water level in the pressurizer via the reactivity control system. Any transient pressure imbalances between the reactor pool and the core will cause borated pool water to enter the core region through the upper and lower density locks, and shutdown the reactor. In addition, automatic and manual trip of the primary pumps will similarly shutdown the plant. Since the PIUS design does not include control rods, trip of the primary pumps represents the active scram of the plant.

An important feature of the PIUS design is a siphon breaker located downstream of the nozzles in the cold leg piping. This siphon breaker prevents the excessive loss of reactor pool inventory following the rupture of the cold-leg main coolant piping.

The reactor pool is prevented from overheating by active and passive cooling of the pool water. The active pool cooling system is non-safety grade and is similar to a typical Residual Heat Removal (RHR) System in a current light water reactor. The passive pool cooling system transfers pool heat to passive water-air heat exchangers located on the roof of the containment. In addition, the pool has been sized to remove residual decay heat for seven days without either active or passive cooling while maintaining a water level above the top of the core.

2.2 Key PIUS 600 Characteristics

The overall design for the Process Inherent Ultimate Safety reactor (PIUS 600) is characterized by the following features:

- Four-loop pressurized water reactor,
- A PCRVR that contains approximately one million gallons of borated water,
- Moderate Power Output (640 MWe),
- A primary system that is partitioned into a heavily borated (2200 ppm) low temperature, large volume, reactor water pool and a four-loop coolant system that includes the steam generators, pumps, and associated piping systems,

A) Pressurizer steam volume

B) Steam generator (4)

C) Upper density lock

D) Primary pump (4)

E) Riser

F) Core instrumentation

G) Embedded membrane

H) Pool liner

I) Reactor Core

J) Lower Density Lock

K) Reactor pool

L) Crossover flow

M) PCRV

N) Submerged pool cooler

O) Prestressing steel cables
(tendons)

P) Concentric Annulus

Q) Siphon breaker

Figure 2-1 Principal Features of PIUS 600

- No control or shutdown rods.
- Power control that is provided by controlling the boration of the primary coolant, the flow of water in the steam generator loops, and the temperature of the feedwater to the secondary steam system.
- The reactor pool water is prevented from entering the core by balancing the pressure in the pool and the pressure in the core region by controlling of the speed of the main coolant pumps.
- Pressure imbalances between the pool and the primary system will result in borated water entering the core, resulting in shutdown of the reactor.
- A reactor pool that can absorb the residual core decay heat for one week without pool cooling or addition of water.

2.3 Simplified PIUS Operations

The following sections briefly describe the various modes of operation of the PIUS design.

2.3.1 Hot/Cold Shutdown

Figure 2-2 shows a simplified representation of the operation of PIUS in a hot or cold shutdown condition. The reactor pool is part of the primary loop and operates in parallel with the path through the reactor. If the core is fresh, it will generate negligible heat, and there will be no appreciable natural circulation flow through the core or any other path. However, after the reactor has operated, the primary coolant is heated by residual heat reducing the density and resulting in natural circulation flow through the core. Since the primary pumps are idle, there is little flow through the steam generators; hence, this natural flow passes out of the upper density lock into the reactor pool where it is cooled and returns through the lower density lock. In this mode, the PIUS reactor is passively cooled. The decay heat is transferred to the reactor pool and removed either actively or passively through the air-cooled heat exchangers.

2.3.2 Startup

Startup of the reactor is begun by injecting nitrogen into the gas lock to accumulate gas in the region above the lower density lock. As shown functionally in Figure 2-3, this accumulation prevents flow through the lower density lock since the natural circulation flow is insufficient to force the gas bubble out. With the lower density lock blocked, the natural circulation flow due to the core residual heat will pass through the steam generators and pumps at a flow rate determined by the thermal driving head and the flow resistances.

The primary pumps are run at a low speed as the reactivity control system is used to deborate the primary system and bring the reactor to a critical condition. The reactor is now critical at a very low power level and does not generate any significant heat. The temperature of the primary system is approximately 100°C and the nitrogen accumulation in the gas lock is removed to prepare for the power ascent.

2.3.3 Power Ascent

Schematically, the power ascent is similar to the hot/cold shutdown process (Figure 2-3). The reactivity control system processes the primary flow, reducing the boron content of the water. This reduction causes the core reactivity to increase, increasing the power level and the thermal driving head through the reactor. This increase in the reactor flow rate must be balanced by an increase in pumping rate through the steam generators. If these flows are imbalanced in either direction, the difference in the flow rates will be matched by flow through the reactor pool. This flow from the pool is borated and would shutdown the reactor.

Figure 2-2 Hot/Cold Shutdown of PIUS

2.3.4 Controlled Shutdown

A controlled shutdown (Figure 2-4) is essentially the reverse of the ascent to power. Initially, the flow is through the steam generator primary, as shown by dotted lines. The reactivity control system adds boron to the primary coolant and reduces the reactivity. The resulting power decrease reduces the thermal driving head across the core and the pump flow rate must be reduced. This process of adding boron and reducing flow rate continues until the reactor becomes subcritical and the primary pumps are stopped. Stopping the pumps upsets the flow balance and additional borated water enters through the lower density lock making the reactor further subcritical. Core cooling is maintained by natural circulation flow through the core and the reactor pool, with the heat being removed by the active or passive pool cooling systems.

2.3.5 Pressurization of the PIUS Primary Loop

During normal operation, the operating temperature in the primary loop must be raised while maintaining flow through the reactor and directing flow to the steam generators. The temperature rise depends on the power level at which the core is operating. Boiling the coolant is prevented by maintaining system pressure sufficiently high. (PIUS operates about 28°C subcooled.) Because the reactor pool is part of the primary loop, the pool, reactor, steam generator, primary pumps, and primary piping are pressurized to 9 MPa. An electrically heated system located inside the reactor containment performs the pressurization.

2.3.6 Reactivity Control of PIUS

The reactivity control of PIUS 600 is provided by controlling the concentration of boric acid in the primary coolant. There are no control rods. The concentration of boron in the primary loop can be changed rapidly by the reactivity control system and by admitting borated water from the reactor pool.

With the proper pressure balance between the pool and the flow loop, the primary loop remains insulated from the pool water, allowing the primary loop to operate at high temperature for thermodynamic efficiency. Any increase or decrease in primary water flow through the steam generators can upset this balance, causing highly borated pool water to enter the core and shutdown the reactor.

2.3.7 Active Scram of PIUS

The active method of scrambling PIUS is to stop one of the primary pumps. Flow through the steam generators drops, and the flow through the core is balanced by flow out of the upper density lock, down through the borated reactor pool, into the lower density lock, and to the core to scram the reactor. The PIUS protection system has several automatic signals that will trip the primary pumps. In addition, there is a manual trip.

2.3.8 Passive Scram of PIUS

An increase in primary pump speed will cause a pressure imbalance at the density locks and highly borated water from the pool will be drawn into the primary loop and scram the reactor. In addition to the speed of shutdown that may be achieved by forced flow of borated water into the reactor, the effect is regenerative. The more boron that is added, the more the power level is reduced thereby increasing the flow of water through the upper density lock.

2.4 PIUS Upset Scenarios

This section briefly discusses seven scenarios that will result in either an active or passive scram of the reactor.

Figure 2-3 Startup

Figure 2-4 Controlled Shutdown

2.4.1 Upset Scenario 1: Overpower

Overpower can occur when the power from the reactor core is greater than that being removed by the steam generators. This event is an underflow type of flow imbalance. The primary inventory in the riser is heated, increasing loop buoyancy and causing a pressure imbalance at the density locks. Borated water from the pool enters the primary loop, scrambling the reactor.

2.4.2 Upset Scenario 2: Excess Heat Removal

If the heat removal at the steam generator increases, the temperature of the cold leg and of the core inlet decrease, resulting in reduced riser temperatures, causing a reduction in buoyancy. This, again, results in ingress of borated water into the primary loop, scrambling the reactor.

2.4.3 Upset Scenario 3: Hot Leg LOCA

If a large pipe break occurs in the hot leg, the rapid depressurization results in void formation in the core. This void formation greatly increases the buoyancy in the riser, creating a pressure imbalance at the density locks, causing the ingress of borated water, described previously as passive scrambling.

The rapid depressurization may result in exceeding the critical heat flux, at least briefly, and must be examined in detail.

2.4.4 Upset Scenario 4: Cold Leg LOCA

If a large pipe break in the cold leg occurs, the rapid depressurization results in void formation in the core and increases the core flow rate. As in the hot-leg LOCA scenario, borated water enters the core region from the reactor pool. The siphon breaker prevents a large loss of inventory from the vessel due to siphoning through the cold-leg break.

2.4.5 Upset Scenario 5: Station Blackout

Loss of power to the primary pumps results in a pressure imbalance between reactor pool and the core region that forces borated pool water to enter the core region and shutdown the reactor.

2.4.6 Upset Scenario 6: Anticipated Transient Without Active Scram

Assuming a loss of electrical load as the anticipated transient, power will be lost to the primary pumps; this loss of pumping results in a pressure imbalance, causing ingress of the reactor pool water to the core region and scram of the reactor.

2.4.7 Upset Scenario 7: Loss of Heat Sink

Loss of the heat sink results in heatup of the primary flow through the steam generators and heatup of the reactor. This event leads to void formation in the primary system, causing changes in core flow not balanced by the steam generator primary flow. The resulting pressure imbalance causes ingress of reactor pool water and shutdown of the reactor.

2.5 Fuel Handling Equipment

For this study, BNL assumed that the reactor can be sufficiently opened that System 2311 has access to the core. System 2311 is a track-mounted bridge with a trolley consisting of two working platforms, a control desk, two hoists, and a telescope. A grapple for handling fuel assemblies is attached to the bottom end of the telescope.

Coarse positioning of the grapple is accomplished by means of coordinate scales and a level meter. Fine positioning of the reactor core level is accomplished by tilting the telescope. All handling is supervised visually or by means of underwater television equipment. Trolley and bridge motion is by thyristor control of electrical motors.

The telescope consists of one fixed tube and four telescoping tubes capable only of transverse motion. The fixed tube may be rotated manually through 360°. The bottom of the inner tube is equipped with a pneumatic grab, TV camera, and floodlights. The grab cannot be opened when loaded.

When the telescope is fully withdrawn, the grapple is at the height of the bottom edge of the core grid. The four positions of the grapple, and the fact that it is unloaded, are sensed by microswitches and indicated at the control desk. The load on the elevator cable is monitored by a cable load weighing machine, which provides overload protection.

2.6 Reactor Parameters

The major design parameters of the PIUS 600 reactor were summarized and compared with a Swedish PWR and a BWR of comparable power in Table 1.3.A of the PSID. This information is reproduced and included here as Table 2-1 below. This comparison shows that the PIUS 600 parameters are conservative in many regards. For example, the primary coolant flow rate is higher for PIUS 600 and the core inlet and outlet temperatures are lower than for the PWR. The average core power density is 69% of the PWR value, and the system pressure is 58% of the PWR value. The maximum fuel temperature is about 1000°C lower than that for a PWR or BWR. The flux distribution across the core is about the same as for the PWR or BWR. The moderator temperature coefficient is more negative than is the coefficient for the BWR, and the maximum boric acid concentration is generally less than for a PWR.

The fuel enrichment is essentially the same as for an LWR, and the burnup is high, although in the projected upper range for a PWR. The number of fuel assemblies per core is more than in a PWR. A noticeable difference between the PIUS 600 and LWRs is the shortened fuel element length: it is 2.6 m compared with about 4 m. Safety implications of this difference are not obvious; the shortened length may afford greater uniformity in fuel burnup. The weight of a fuel assembly is slightly more than in a PWR and considerably less than in a BWR.

There are four primary coolant loops for PIUS 600, the same as a large PWR; the steam pressure is about 2/3 of that of a PWR and comparable to a BWR. The coolant temperature range is that of an LWR to obtain comparable thermodynamic efficiency. The number of steam lines is the same as in an LWR (4). The rated thermal and electric power is about the same as in a mid-sized LWR. PIUS 600 has two steam-driven feedwater pumps, as do many LWRs.

A major difference between PIUS 600 and the LWRs in the comparison is the reactor pool. PIUS 600 operates submerged in a borated pool of water to essentially eliminate the possibility of core uncover. This pool is very large being sized to remove the decay heat from a core that had been operating at full power. It would take one week before sufficient pool water had boiled off to cause core uncover (assuming no return of the boil off condensate or make-up feed).

2.7 System Identifications and Relationships

PIUS has several innovative design features. The reactor must contain nuclear fuel in a moderating and cooling environment to produce a nuclear reaction and remove the heat produced. The heat that is produced is converted to steam to drive a turbine and generate electricity. A heat sink must be provided, as required for any thermodynamic cycle. An electrical distribution network is required with other support services. Systems that protect against upset conditions, including a containment, must be provided, as well as the automatic and human interface controls and indicators. This section gives an outline of the identifiable systems in the plant.

Table 2-1: PIUS 600 Parameters with PWR and BWR Comparisons ¹			
Parameter	PIUS	PWR (Ringals)	BWR (Oskarsh- am3)
Thermal and Hydraulic Design			
Rated power MWt	2000	2775	3020
Core coolant flow rate, kg/s	13,000	12,865	11,400
Core coolant temperatures,			
Core inlet °C	260	284	278
Core outlet °C	290	322.6	286
System pressure MPa (psi)	9.0 (1305)	15.5(2248)	7.0(1015)
Average power density, kW/l	72	104.5	49
Maximum fuel rod thermal output, kW/m	29.8	41.4	41.5
Average fuel rod thermal output, kW/m	11.9	17.86	18.1
Maximum heat flux, kW/m ²	1000	1390	1080
Average heat flux, kW/m ²	400	598	473
Maximum fuel temperature °C	700	1870	< 1800
Core inlet coolant enthalpy, kJ/kg	1135		1224
Feedwater temperature, °C	210	221	215
Design Power Peaking Factor			
Radial peaking factor (hot channel)	1.55		1.68
Axial peaking factor	1.4	< 1.6	1.38
Total peaking factor	2.17	< 2.22	2.32
Nuclear Design			
Moderator void coef. 1E-5/%	> -200		> -120
Moderator temperature coef. at power 1E-5/°C, BOC ²	-36		> -26
Moderator temperature coef. at power 1E-5/°C, EOC ³	-46		
Fuel temp. Doppler coef., hot,/°C	-3.10E-5		
Equilibrium enrichment, %	3.5	3.27	3.1
Average discharge exposure, MWd/kg U	45		22/32
Equilibrium fuel cycle	12 mo.		
Maximum boron concentration in primary ppm	375		
Core Mechanical Design/Fuel Assembly			
Number per core	213	157	700
Rod array	18x18	17x17	8x8
Array form	square	square	square
Overall width, m	2.26	2.14	1.36
Length, m	2.6	4.06	4.13
Uranium/assembly, kg	377.8	523	178
Assembly weight, kg	546.7		297
Fuel channels	no	no	square

¹ Parameters were obtained from the Preliminary Safety Information Document

² Beginning of Cycle

³ End of Cycle

Table 2-1: PIUS 600 Parameters with PWR and BWR Comparisons (cont.)			
Parameter	PIUS	PWR (Ringals)	BWR (Oskarsh- am3)
Core Mechanical Design/Fuel Rods			
Number per assembly	316	264	63
Outside diameter, m	9.5	9.5	12.25/11.75
Cladding	Zirc-4	Zirc-4	Zirc-2
Rod length, m	2.625		3.993
Fuel pellet material	UO ₂	UO ₂	UO ₂
Core Assembly			
Fuel weight as U in 1E3 kg	80.5	72.4	124.4
Equivalent core diameter, m	3.75	3.035	4.6
Active core height, m	2.5	3.658	3.68
Steam Generation			
Number of steam generators	4	3	4
Steam outlet pressure, MPa	4.0	6.0	4.7
Steam outlet temperature, °C	270	276	260
Feedwater inlet temperature, °C	210	221	187
Main steam lines	4	4	4
Turbine Plant			
Rated power, MWt	2000	2x1390	2060
Rated power, MWe	700	2x490	680
Turbine speed, rpm	3000	3000	1800
Rated steam flow, kg/s	1012	2x760	1047
Admission pressure, MPa	3.87	5.9	4.6
Feedwater System			
Number of feedwater pumps	2	2x3	3
Feedwater pump drive	turbine	electric	electric
Reactor Pool			
Water in reactor pool, m ³ (gal)	3.3E3 (8.7E5)		
Pool dimensions height, width, depth, m(ft)	43(141) 12(40) 38(115)		
Pool wall thickness, m(ft)	7(23)		
Pool liner	1		
Pool membrane	1 -		
Boron concentration, ppm	2200		

The BNL analysis of the PIUS design used the nomenclature given in the PIUS Preliminary Safety Information Document (PSID). Table 2-2 shows the top-level identifications.

Not all of the systems are equally important to safety; thus, the review of the total system requires using judgement as well as experience gained from reviewing the current generation nuclear power plants to select the systems of most significance to safety, and to prioritize the review effort. Using these considerations, BNL selected the major categories 200, 300, 500, and 600 for the investigations described in this report. Table 2-3 shows the system identification to the first three digits. This system identification is very useful to:

- (1) assure against overlooking systems having an impact on safety (completeness),
- (2) to provide a map to the plant organization,
- (3) aid in locating systems whose description and FMECA are presented in Appendix A.

The system identification is given in this section for use as a reference for the remaining sections of this report.

Table 2-2: Top-Level PIUS Structures and Systems	
Number Identification	Entity Identified
100	Buildings and External Systems
200	Reactor and Reactor Auxiliary Equipment
300	Reactor Plant Process Systems
400	Steam and Power Conversion Systems
500	Central Equipment
600	Electrical Equipment
700	Service Systems
800	Other Plant Equipment

Table 2-3 PIUS System Identification

100 Building, External Systems	200 Reactor and RX Auxiliary Equipment	300 Reactor Plant Process Systems
110 General Civil Engineering Structures	210 Reactor systems	310 Primary loop systems
111 Plant site 113 Cooling water channels 114 Culverts and dikes 115 Roads and parking areas	211 Upper parts of pressure vessel 213 Reactor internals 215 Reactor assembly supports and restraints 216 Vessel internal instr. support systems 217 Startup neutron sources 218 Reactor materials test specimens support equipment	312 Reactivity control systems 313 Recirculation system 314 Pressure relief system 316 Condensation system
120 Central building compound	230 Handling equipment	320 Pool loop systems
121 Reactor building 122 Turbine building 123 Aux. cooling systems building 124 Chemical systems building 125 Control systems buildings 126 Electrical system building 127 Diesel building 129 Active workshop building	231 Refueling systems 233 Reactor internals handling equipment 235 Vessel instr. replacement equipment 237 Tools for vessel internals 239 Handling equipment for reactor internals	321 Reactor pool 322 Pool loop inactivations systems 323 Reactor pool cooling system 324 Fuel pool cooling and cleanup system 325 Reactor pool water replacement system 328 Pressurizer systems
130 Separate buildings and structures	240 Storage equipment	330 External systems
134 Transformer buildings 135 Water treatment building 136 Cooling water rinsing systems building 138 Cooling tower pump building 139 Security building	241 Fresh fuel storage equipment 242 Fuel service pool equipment 243 Spent fuel storage pool equipment 244 Reactor service pool equipment	331 Reactor water cleanup systems 333 Chemical additives system 336 Sampling system 338 Controlled drains system

Table 2-3 PIUS System Identification (Cont'd.)

140 Secondary buildings and structures	250 Service equipment	340 Waste systems
141 Meteorological tower 142 High voltage switchyard structures 144 Main plant stack 146 Permanent personnel building 147 Raw water treatment buildings 148 Storage and workshop area 149 Other permanent structures	253 Fuel service equipment 254 Failed fuel locating equipment 257 Loose parts monitoring system 258 Plant in-service inspection equipment	341 Offgas delay system 342 Liquid waste system 343 Solid waste system 344 Decontamination systems 345 Monitored floor drain systems 349 Offgas filter system
150 Concrete vessel	260 Reactor fuel	350 Other reactor process systems
151 Concrete vessel monolith 152 Containment 157 Concrete vessel liner cooling system 159 Concrete vessel liner venting system	261 Fuel assemblies 263 Spare fuel, replacement fuel	355 Containment pool cooling system
160 Cooling towers	290 Thermal insulation systems	
	291 Wet thermal insulation for reactor primary system 293 Other wet thermal insulation inside of the reactor pressure vessel 295 Thermal insulation system outside of the reactor pressure vessel	
400 Steam and power conversion systems	500 Central equipment	600 Electrical equipment
410 Steam line systems	510 Common central equipment	610 Grid voltage system
411 Steam line system 413 Steam bypass system 415 Turbine sealing system 419 Live steam feedwater heating system	511 Main central room 515 Central cables 516 Trip and interlock systems 517 Alarm and display systems 518 Emergency control room	611 Grid voltage system

Table 2-3 PIUS System Identification (Cont'd.)

420 Steam turbine systems	520 Computer systems	620 High voltage switchyard
421 Main steam turbine 422 Steam admission system 423 Moisture separator and steam reheat system 425 Lubrication and jacking system 426 Control and trip oil system 427 Steam extraction system 428 Other turbine auxiliary systems 429 Turbine protection devices	521 Main process computer 523 Control room man-machine interface system 524 Control processors 525 Core management computer system 527 Safety parameter display system	621 High voltage switchyard equipment
430 Turbine generator systems	530 Reactor and pool control and monitoring systems	630 Generator bus system
431 Main plant generator 432 Exciter and voltage control system 433 Seal oil system 435 Generator cooling water system 437 Hydrogen and carbon dioxide systems	531 Neutron flux monitoring systems 533 Water chemistry control system 534 Main coolant pump control system 535 Reactor power control system 536 Reactor, reactor pool and concrete vessel monitoring systems 537 Reactor pool level control system 538 Reactor pressure control system	631 Generator bus system
440 Turbine condenser systems	540 Process control and measurement system	640 Regular auxiliary AC power supply systems
441 Turbine condenser 442 Condenser gas removal system 446 Condenser makeup water system 447 Seal water system	541 Process measurements principles 542 Process control principles 543 Process operation principles 545 Leak detection principles	642 6.0 kV auxiliary AC system 643 660 V auxiliary AC system 648 Pressurizer, heater power supply system 649 Main coolant pump power supply system

Table 2-3 PIUS System Identification (Cont'd.)		
450 Circulating water systems	540 Process control and measurement system	
451 Condenser cooling pump system 453 Cooling tower pump system 456 Circulation water systems makeup water system 457 Circulation water systems water quality control system	546 Pipe rupture monitoring systems 547 Other process monitoring systems	
460 Feedwater system	550 Radiation monitoring systems	650 Diesel plant
461 Condensate system 463 Feedwater system 467 Condensate cleanup system	551 Extended delay offgas system inlet radiation monitor 552 Half-hour delay offgas system inlet radiation monitor 553 Stack radiation monitors 554 Steam generator tube rupture monitoring system 555 Room radiation monitors 556 Portable radiation monitors 557 Environmental radiation monitors	651 Diesel plant
470 Turbine plant service systems	580 Other measurements and monitoring systems	660 Diesel-backed AC power supply systems
472 Turbine plant leakage and drain systems 474 Turbine plant drying system	581 Containment monitoring system 584 Earthquake monitoring instruments 588 Meteorological instruments	661 Diesel-backed 660 VAC power supply system

Table 2-3 PIUS System Identification (Cont'd.)		
	590 Plant and turbine control systems	670 Battery-backed power supply systems
	591 Plant power control system 592 Turbine protection system 593 Turbine plant process control systems 594 Turbine plant supervision systems 595 Turbine automatic operations system 597 Turbine control system 598 Feedwater control system	672 Regular 110 VDC auxiliary power supply system 673 Safety-grade 110 VDC auxiliary power supply system 677 400 V battery-backed auxiliary power AC system
700 Service systems	800 Other plant equipment	
710 Service water systems	810 Transport equipment	
711 Cooling water intake rinsing systems 712 Diesel-backed service water system 713 Non-diesel-backed service water system 714 Cooling towers for cooling the reactor pool	811 Overhead cranes 812 Elevators 815 Vehicles	
720 Closed cooling system	820 Tools and furnishings	
721 Diesel-backed closed cooling system for the reactor pool 722 Diesel-backed closed cooling system for components and pools 723 Non-diesel-backed closed cooling system 724 Passive closed cooling system for the reactor pool 726 Chilled water system 727 Local air coolers system	821 Laboratory equipment 822 Cleaning equipment 823 Mechanical workshop and store equipment 824 shielded transport containers 825 Special tools 826 Furniture etc. 827 Electrical workshop equipment 828 Instrument workshop equipment	

Table 2-3 PIUS System Identification (Cont'd.)

730 Water treatment and distribution systems	830 Lighting systems and local power supplies	
731 Raw water treatment system 732 Water demineralizing system 733 Demineralized water distribution system 734 High pressure distilled water system	831 Indoor normal lighting systems 832 Outdoor lighting systems 835 emergency lighting system 837 Local power supplies	
740 Ventilation systems	840 On-site communications and alarm systems	
745 Other controlled area ventilation systems 746 Control building ventilation system 747 Electrical and diesel building ventilation systems 748 Ventilation systems for other non-controlled area buildings 749 Emergency ventilation system	841 Plant internal communications systems 842 External telephone exchange 843 Alarm systems 844 Paging system 845 Public address system 846 Clock system 847 Area supervision system 848 Entrance and exit supervision system 849 Plant operation television monitoring systems	
750 Compressed gas systems	850 Off-site communications systems	
751 Bottle air system 753 Compressed air system 754 Compressed nitrogen system	851 Carrier frequency system 852 External radio communications system 853 Dispatch center communications system exchange 856 Plant security center	
760 Heating and drain systems		
761 Tap water systems 763 Plant heating system 765 Non-controlled area floor drain systems 767 Ground water drain system 768 Surface water drain system		

3.0 METHODOLOGY

This section presents the methodology used for the PIUS analysis. The description of each of the parts will receive additional coverage in subsequent sections.

3.1 Methodology Overview

Designs of advanced reactors as submitted to NRC often present systems that are significantly different from conventional nuclear power plants, and sometimes lack the design details needed for construction. Moreover, there is little or no operational experience with portions of these designs, and with some of the equipment.

In investigating a methodology suitable for analyzing PIUS-600, BNL drew on experience in applying PRA methods to the design of the Advanced Neutron Source reactor at Oak Ridge National Laboratory. This work began with conventional PRA methods but encountered difficulties due to lack of detail and operating experience, that are associated with the analysis of the advanced nuclear power plants. Over the course of the more than five years of work on ANS, Failure Modes Effects and Criticality Analysis (FMECA)¹ has been demonstrated to be an effective methodology for overcoming the noted limitations, for discovering design weakness, and for working with the designers to achieve improved designs. Based on this experience, FMECA was adopted as one of the primary tools of the BNL methodology for advanced reactor design analysis.

To further enhance the ability to discover potential accidents, the BNL methodology supplements the inductive² FMECA with the deductive Hazards and Operability Analysis (HAZOP) (Lees, 1980). Whereas FMECA examines the effects of component failure on systems, and systems failure on the plant, HAZOP begins by considering the effects at the plant level of perturbations of the process variables on system safety. HAZOP then goes on to determine the system failures that could cause the perturbations.

Both the FMECA and HAZOP are supported and guided by a multi-disciplinary team of specialists performing deterministic³ analyses in reactor physics, thermal hydraulics, materials, stress analysis, chemical effects, and reactor systems. The determinist analysts provide consultation to the system analysts, as well as the independent analyses that are included in this report.

Figure 3-1 shows the overall BNL methodology for analyzing the safety of advanced reactors. The methodology begins with deterministic and probabilistic analyses by the multi-disciplinary PIUS-600 team. The FMECA and HAZOP probabilistic analyses are performed independently of each other to achieve a more thorough screening process than either alone. The team, using PIUS information⁴, with an understanding of regulatory objectives, performs their analyses to produce FMECA tables, HAZOP records, and deterministic studies.

These tables, records, and studies are merged in the PIUS Interim Table (PIT), which classifies each PIT

¹ A FMECA provides a tabular listing of the effects of components, subsystem and/or system failure in a plant.

² Inductive logic is reasoning from cause to effect; deductive logic is reasoning from effect to cause.

³ The distinction between deterministic and probabilistic processes is sometimes not self-evident. Here, we call *probabilistic* the methods used to model reliability: FMECA and HAZOP, and call *deterministic* the conventional engineering methods used to analyze: heat, flow, stress, vibration and corrosion.

⁴ The PIUS information is primarily from the Preliminary Safety Information Document (PSID) with additional information in the form of test loop information, technical articles and ABB responses to questions.

Figure 3-1 Diagram of BNL Methodology Applied to PIUS-600

entry according to whether it is an accident initiator or an accident mitigator. These are the information classes used to prepare accident sequences that are displayed as event trees.

The PIT is principally based on considerations of single failures, but multiple failures of safety-grade and non-safety grade systems are possible and must be considered. Such is done through the development of Event Trees and a System Dependency Analysis.

The next step uses the information contained in the PIT, in the event trees and the system dependency analysis to prepare the following tables for use in guiding and prioritizing the regulation of PIUS-600:

- Summary of the most significant findings from the various analyses performed, i.e., FMECA, HAZOP, and other various supporting deterministic analyses.
- Various table of accident sequences that include the initiator, the failures that contribute to the initiator, the systems important to mitigating the initiator (systems important to safety). Some tables include qualitative estimates of the initiator frequency, mitigating system failure, estimates of sequence frequency, and sequence end state severity.
- Several tables of initiators and systems, important to safety including frequency class and prioritized and sorted in difference ways.
- Key operator response actions, including related event sequence and mitigation function affected.

- A list of test, maintenance, and inservice inspection concerns.

Also provided is a critique of the methodology used to analyze PIUS-600, and a list of information areas that were lacking in the PSID.

With this overview, we will now discuss the parts of the methodology.

3.2 Description of Failure Modes Effects and Criticality Analysis (FMECA)

3.2.1 Qualifications and Information Needs

Preparing a FMECA requires a fundamental understanding of the physical processes involved in the equipment being analyzed - how things work and how they fail. In addition to knowing how the parts work, it also is necessary to understand how the parts and humans work together to form subsystems, systems, and the plant. Moreover, the analyses must be performed for the normal modes (power, startup, hot and cold shutdown, refueling) and upset conditions. Furthermore, the analyst must estimate the likelihood of the off-normal situation and the consequences to the public and worker safety. When this estimate is done, the postulated events may be ranked by their risk significance.

A FMECA may be prepared by a team or the initial FMECA may be prepared an individual to be reviewed by team members who require explanations and justifications for the analyses and conclusions. The former method may be the best way for very complex systems because it requires the knowledge of several people for understanding the system. The latter method, after reviews and response to the reviews, is believed to be more efficient and robust than the former, and is the method used in this analysis of PIUS-600.

3.2.2 Contents of a FMECA

A plant design is reviewed to identify the forces that would start and propagate an accident, locations of hazardous material, and the levels of protection that are afforded. Eventually, all parts of a plant that appear to have accident potential will be studied, but there must be a starting place, which may be a system perceived to have the greatest potential risk to the public and workers. For a reactor, this is the core, composed of fuel, neutron absorbers, coolant, flow channels, and the pressure boundary envelope. Because all important elements of the reactor will be analyzed, the starting component and the order in which the components are analyzed, makes little difference, although completing one system analysis before starting another is advisable for reasons of logic and grouping of information.

Each component of the design is identified and its purpose is explained. A failure mode is identified along with the causes of failure. The effect of this failure on the system and plant is stated, followed by a qualitative risk estimate called criticality, and a qualitative frequency estimate for this failure mode. This frequency estimate may be based on actuarial data, part count, or reliability class. Mitigative factors are stated, followed by observations that are made in performing the analysis.

This process is repeated until all parts of the plant, deemed to be important for public and worker safety, have been analyzed.

3.2.3 Depth of a FMECA

The depth of analysis should be limited to those items needed to envelope the risk. This depth depends on the plant being analyzed, its complexity, and operating experience with the type of component. PIUS-600 uses unique design principles and components. Thus, some components and systems are explored in detail to determine failure modes that are not obvious or that have not been experienced. The depth of analysis must reach a reasonable postulated cause of failure through the assistance of the deterministic analysts. System dependencies also must be

identified.

3.2.4 FMECA Preparation

The failure mode, and effects analysis (FMEA) or its extension, the failure mode, effects and criticality analysis (FMECA), is a systematic, standard (IEEE, 1975) method for reliability analysis with established procedures. After the analyst becomes familiar with the plant and its operation, a FMECA is prepared by selecting a system that is important to operation or safety, selecting a subsystem of the system and a component in the subsystem for analysis. Because there will be a large number of analyses, they must be prepared systematically in a table such as that shown in Figure 3-2. This table is prepared with a personal computer using a word processor⁵.

The FMECA table consists of a major title followed by a line that identifies the system and the subsystem being investigated. The table has nine columns. The left column contains numbers that identify each of the

Table A.xxx PIUS 600 Failure Modes Effects and Criticality Analysis								
Electric Power 600: Power Cable System 690 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date 1/10/92								
No.	Component/ Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks

Figure 3-2 Tabular Form of the FMECA

analyses. In some cases, a number and a letter are used to identify analyses in which different failure modes are analyzed e.g., a valve failing open and the valve failing closed. This column is followed by a description of the item and its function. The third column contains the postulated failure mode e.g., fail open. The fourth column lists the causes for the failure mode, e.g. binding, distortion, corrosion or power failure. This column may contain

⁵ For the PIUS FMECA, the word processor was WordPerfect 5.1 with the table feature.

multiple failures that may be necessary for the failure of this component. The fifth column explains the effect on the system and plant. Criticality, listed in column six, is a qualitative risk estimate, a combination of consequences and frequency of occurrence. The criticality categories used in this work are:

- High - further consideration is recommended
- Medium - possibly significant
- Low - not believed to be risk significant.

The estimated frequency of occurrence of a failure used in column 7 is based on the following categories provided in the PIUS PSID⁶:

- I - frequent: 10/year to 1/year
- II - anticipated: 1/year to 0.1/year
- III - rare: 0.1/year to 0.01/year
- IV - very rare: 0.01/year to 1E-4/year
- V - extremely rare: 1E-4/year to 1E-7/year
- VI - incredible: less than 1E-7/year.

The mitigative factors are presented in the next column. These factors could include equipment providing redundancy, human actions, or considerations mitigating the effects of the failure. The last column shows observations made by the analyst regarding assumed conditions for the analysis, or suggestions for improving the design. In preparing the FMECAs for PIUS that are contained in Appendix A of this report, we found it necessary, in some cases, to give additional descriptions of the assumed conditions than is practical in the space provided in the table. Thus, notes are attached; these may be applicable for more than one item.

When a FMECA on a component of a system is completed, another component is selected until the system is completely analyzed. This process is repeated until all systems important to the plant have been analyzed.

3.2.5 Practical Completeness

Practical completeness means that all significant ways in which a plant may fail and affect the public or workers have been identified, including a very large number of ways, some of which involve highly improbable combinations. Low probability events are small risk contributors, and analysis of the more probable events will encompass most of the risk. Completeness is approached in this study by:

- 1) System classification,
- 2) Experience,
- 3) Peer review,
- 4) Preparing an initiator matrix, and
- 5) Identifying multiple failures

1) System Classification

⁶ Category I is changed from the ABB nomenclature from "normal plant operation" to that shown.

The system classification scheme used by PIUS and given in Section 2, is an inventory of the equipment in the plant organized by system, subsystem, and component. By stepping through the system classification, all plant components are considered. For practical reasons, FMECAs are not prepared for all components of the system classification; the analyst reviews all components and analyzes those considered to be important to safety.

2) Experience

Several hundred years of power reactor operating experience have been amassed in the United States and much more in the world. This experience is valuable for analyzing advanced reactors and it is used in preparing the FMECAs.

3) Peer Review

An experienced analyst, working alone, attains a high level of concentration that leads to indepth FMECAs. Nevertheless, one or more peer analysts working at a similar depth may identify items not in the first FMECA. They may question items for reasons overlooked by the first analyst or overlooked by the peer analysts. Thus, peer review contributes significantly to the FMECA process.

4) Initiator Matrix

An initiator matrix aids completeness by providing a written list of things that might cause failure. Figure 3-3 shows this as a conceptual matrix equation in which failure causes operate sequentially on the item under FMECA analysis. If a thing may cause an item to fail, the analyst includes this as one of the failure causes in the FMECA. For example, a thing may be stress-corrosion cracking, the item under consideration may be pipe, and, stress corrosion may cause pipe failure; hence, it would be included in the FMECA. The initiator matrix is a systematic way to consider the various causes of failure.

Figure 3-3 Initiator Matrix

5) Identification of Multiple Failures

The FMECA considers the independent failure of components; a component may also fail as the result of a support system failure, or from multiple independent failures. While multiple independent failures can be addressed in a FMECA, there are limits to the number of considerations that can be addressed at one time by an analyst. Thus, as explained in Section 3.1, analyses of major multiple failures are deferred, primarily, until the interim results have been assembled. They are then addressed via event tree analysis in Section 7.

3.3 Description of Hazards and Operability Analysis (HAZOP)

3.3.1 Overview of HAZOP

Hazards and Operability (HAZOP) analysis is an accident detection and prevention technique used by the Chemical Process Industry (CPI) (Lees, 1980). Even though the CPI operates in a different regulatory environment from the commercial nuclear power industry, the goals of risk reduction while maintaining productivity are similar. BNL uses HAZOP for analyzing PIUS-600 to provide a deductive method, which is different from, and complementary to the FMECA. Using both methods leads to more complete accident identification than either

alone.

HAZOP is a formal technique for eliciting insights about system behavior from a multi-disciplinary team that collectively has thorough knowledge of the plant and the physical phenomena involved in the plant.

The HAZOP team evaluates the PIUS-600 plant through a process that involves selecting a system, applying guide words to the selected system, and identifying causes, and consequences of the postulated event. Occasionally, it is not possible to quickly resolve a postulated occurrence on the basis of the available information and expertise. In this case, the HAZOP leader may assign the best qualified team member(s), to further investigate and report the results back to the team. The team reviews the item and the proposed resolution. When the team agrees, the analysis is recorded and the item is closed out.

3.3.2 Composition of the HAZOP Team

A key member of the HAZOP is the HAZOP leader whose duties are to:

- Assure that all necessary disciplines are represented by the team and that the team members are familiar with the system being analyzed,
- Select the system to be analyzed,
- Provide supporting documents and diagrams for the system being investigated and the systems with which it interacts
- Assure that all systems and operating modes deemed to be important to safety are eventually addressed,
- Guide the team in addressing multiple failures,
- Resolve misunderstandings,
- Identify the resolution of technical problems, and
- Assure each HAZOP session is documented.

Responsibilities of the HAZOP team members are to:

- Provide expertise in the discipline that they represent on the team,
- Be familiar with the system being analyzed and other systems with which it may interact, and
- Resolve issues which are assigned.

The expertise represented by a HAZOP team depends on the system being analyzed. A typical HAZOP team should have expertise for the PIUS study in the following areas:

- Instrumentation,
- Control,
- Electrical power,
- Reactor physics,
- Reactor systems,
- Accident phenomenology,
- Chemistry,
- Stress analysis,
- Material properties,
- Reliability specialists,
- Human factors, and
- Reactor and plant operations.

Individuals may provide expertise in several of the above disciplines. The size of the team should be large enough to achieve diversity of points of view, but small enough to function as a team focussed on the analysis at hand. The leader must assure that all members contribute and avoid any intimidating influences.

3.3.3 Recording the Session

The recorder records the HAZOP session on a personal computer, which may be a laptop. The program that is used is HAZSEC available from Technica⁷. This record is not a verbatim transcript; it is a recording of intermediate or final results as directed by the team leader.

Appendix B gives the records of typical HAZOP sessions as recorded using HAZSEC.

HAZSEC generates two types of records. The first page is the log sheet with the time, date, revision number, team leader, and team members. This page also contains a section that describes the part of the plant design under investigation, as well as a statement of the design intent, that is, what the equipment is intended to do under both normal and accident conditions. The pages that follow repeat the design intent with dates and page numbers. For each guide word⁸, there are one or more causes of the postulated condition with associated statements of the consequences of this guide word operating on the equipment.

If the item is not resolved, an individual is assigned responsibility for resolving it after the session. Requests may be made for additional information.

3.3.4 Identification of the Operating Mode

Figure 3-4 shows the HAZOP process as consisting of three looping processes. PIUS has several operating modes (normal power, startup, scram, hot shutdown, cold shutdown, refueling) in which the design's systems may operate differently. The first loop sets the operating mode to be considered in the subsequent analyses. The mode selection is performed by the HAZOP team leader.

⁷ Technica Inc., Software Products Division, 40925 County Center Dr., Suite 200, Temecula CA 92591.

⁸ A guide word (or phrase) is a perturbation on a process variable, component or system.

Figure 3-4 The HAZOP Process

3.3.5 Process Variable Selection

Having selected the operating mode, the HAZOP leader selects the process variable to be considered for the selected mode.

3.3.6 System (Node) Selection

The HAZOP leader selects an appropriate location within the plant's process systems (a process node) and the associated systems of the design of the plant, that affect the selected process variable for the selected mode of plant operation. The selection may be made from the plant system classification, or it may be from the nodal analysis for the thermal hydraulics analysis of the plant.

3.3.7 Design Intent

The information assembled for the selected node includes descriptions of the hardware and physical parameters of the operating mode. For example, the information for a fluid system must include physical parameters, such as flow, pressure, temperature, temperature gradients, density, and chemical composition for every

node⁹ at which the description has meaning. This characterization of the system is called the "design intent".

The first step in analyzing a node is to achieve team consensus on the design intent.

3.3.8 Guide Words and Causes

For each system node the HAZOP discussion proceeds by considering the effect of a set of appropriately selected "guide words" on the design and operating parameters of that node. A typical example of a guide word could be "less flow". These guide words come from standard HAZOP practice, but can be customized to a particular design analysis. During the first phase of the discussion the objective is to identify all potential "causes" of the guide word, i.e. all potential functions which could result in a flow reduction.

As an aid to assure completeness, a general set of guide words is maintained in the computer files of the HAZOP effort and are considered, one by one, for each node.

Guide words used in the HAZOP analysis of PIUS are:

Abnormal Operations - prompts the team to investigate whether off-normal conditions would modify analyses performed under the assumption of normal operations.

Contamination - prompts the team to investigate possible causes and quantities of contamination, namely any of the following: radioactive contamination, improper water chemistry, non-condensable gases. There is a reminder at each system (node) to search for compositional anomalies other than boron concentration.

Corrosion / Erosion - prompts the team to examine the system (node)'s susceptibility to chemical action or erosion.

External Events - causes the team to consider external events (seismic, tornado/high wind, flooding) on a system (node).

Flow Tilt - prompts the team to investigate flow imbalances within systems (nodes) e.g., more cooling on one part of the core than another.

Geometry - refers to relative motion of structures that change their spatial relationships.

Circulation - prompts the team to examine circulating flow patterns within systems (nodes) e.g. possible convection flow within the reactor pool. As another example, applying the guide word to the pressurizer suggests spillover from the riser funnel into the pool.

Ignition - fire and/or explosion.

Instrumentation / Control - reminds the team to direct attention from the process variables to the sensor locations, whether there are sufficient sensors, and whether the right parameters are being monitored and controlled (see also sampling).

Lower / Higher Reactivity - applicable to the core channel system (node)s.

Maintenance - suggests that the team investigate maintenance or improper maintenance on equipment operability.

⁹ The term "node" means a location at which something different occurs. It may be a location at which streams converge e.g. an orifice, or a tank. However, at a region along which the physical parameters are constant, only one node would be used to represent the region.

More / Less / Oscillating Pressure - these guide words are self-explanatory.

More / Less / Oscillating Temperature - some PIUS systems (nodes) are large enough that they encompass static temperature gradients. For these, localized temperature variables are used.

More / Less Boron - applicable to the boron concentration in many systems (not just the core channel nodes).

More / Less / Oscillating Level - applicable to nodes in which the coolant is two-phase (liquid and gas) hence relatively compressible.

No / Reverse / More / Less / Oscillating Flow - not all systems (nodes) have flow, and for some that have, they may have flow in more than one region.

Power Tilt - suggests that the team investigates power imbalance within the core region.

Radiation - prompts the team to examine a system (node) for radiation hardening or other radiation-induced materials effects. Radioactive contamination is identified under a different guide word.

Relief - prompts the team to determine whether or not the subject system (node) has access to pressure relief under all relevant conditions.

Safety - suggests that the team consider operational accidents with regard for the operating personnel e.g., injury or radiation exposure during refuelling.

Sampling - is related to the Instrumentation / Control guide words; it prompts the team to examine sampling regarding extensiveness and information yield.

Service Failure - failure of a support system, such as control power.

Spare Equipment - queries the availability of spares.

Stratification - refers to thermal and boron stratification.

Structural Integrity - refers to maintaining the design strength and leak-tightness.

In listing the causes that led to a guide word, engineering judgement must be imposed, to exclude clearly incredible causes. However, it is not prudent to restrict the list to items of very high probability or of obviously significant effects only. The objective remains to obtain insight into the characteristics of the system under various deviations from normal operation. It must also be noted, that the causes can originate anywhere in the system, not just in the specific node, currently under consideration.

3.3.9 Consequence Identification

Upon completion of the list of causes, the team will address the potential consequences from each of the listed causes to any node in the system. That is, the evaluation will consider consequences arising anywhere in the system, due to system-wide causes, which can effect the subject deviation in the subject node. These consequences include potential operating problems and safety concerns.

3.3.10 Resolution

Complex or particularly significant problems regarding a system may not be resolved immediately, and require further action. In such a case, a team member, whose expertise is most closely related to the subject, is assigned the task of obtaining the information and reporting back to the HAZOP team for close-out.

3.3.11 Review

Several systems (nodes) at a time may be in the process of review and resolution. In some cases, the causes and consequences for a given section, system, node, and guide word can be readily assigned and that item closed out. In other cases, close-out may be deferred until the assigned resolution is complete. Even less frequently, items that were closed-out may be reopened in the light of new information.

These are the steps in the review that, in the end, lead to a thorough investigation of safety and operability of PIUS.

3.3.12 Record Keeping

As described, each session of the HAZOP team is recorded using a computer and special software. These records are revised whenever open items are resolved to produce the final record.

Upon recording the information for one process variable, another process variable is selected, as shown in Figure 4-2, and the process repeats until all important process variables have been considered for an operating mode. When this occurs, another operating mode is selected and the whole process repeats.

3.4 Overview of Deterministic Analyses

3.4.1 Reactor Physics/Thermal Hydraulics

The FMECA and HAZOP analyses identify possible effects of failure which only can be resolved by detailed thermal-hydraulic calculations. These calculations are performed by the PIUS Interactive Plant Analyzer (PIPA) code operated interactively in a personal computer. PIPA is a versatile analytical tool that can model many PIUS scenarios. Section 6.1 describes the PIPA code. The following is a summary of the code features that are important to the application of this methodology:

1. PIPA includes models of the reactor core, the primary coolant system with the pressurizer, the coolant pumps and steam generator, and the upper and lower density locks. In addition, the large borated pool is modelled to calculate natural circulation flow between the primary system and the pool.
2. PIUS thermal-hydraulics are modelled with a non-homogeneous, two-phase drift flux model, and an integrated loop-momentum equation. Two-phase conditions are important in the PIUS model because the plant is designed to operate with a system pressure only slightly (2 MPa or 290 psi) higher than the pressure at which bulk boiling occurs.
3. Neutron kinetics are modelled with the point kinetics equations to account for the reactivity effects of boron, fuel temperature, primary coolant temperature, and steam voids.
4. Boron transport modelling is particularly important in the PIUS analysis, therefore a convective model with a flow dependent mixing efficiency has been incorporated in PIPA.
5. Heat transfer between the fuel and the coolant is modelled with both single- and two-phase heat transfer correlations. Fuel temperatures are calculated with a one-dimensional transient heat conduction model.

Information from the PIPA analysis is useful for evaluating the consequences of the event tree scenarios, including peak system pressure, peak fuel/clad temperature, reactor power, and departure from nucleate boiling ratio (DNBR). As discussed in Section 6.1, such calculations augment more approximate calculations, and engineering judgements for determining the consequences of failure or upset. In addition, prioritized lists of scenarios requiring further detailed analysis are compiled.

In the application of this methodology, certain event sequences may be identified that are beyond the capabilities of the code. These extreme events could occur as a result of multiple failures. However, the flexibility of the PIPA method is used to provide a facile, approximate calculation of the plant condition if such extreme events were to occur. These calculations may be supplemented by other calculations and engineering judgement. The extensive experience of the BNL team in current LWR plant transient and severe accident analyses provides further insights into the PIUS sequence analyses.

3.4.2 Materials Properties and Chemistry

In evaluating the degradation and potential failure of PIUS components, we reviewed information in the literature for analogous situations in typical PWR systems. However, there are many PIUS design features and material selections for the primary system that are quite unique. In these cases, it was necessary to draw upon experience with non-nuclear systems if such information was available.

Within the scope of this work, it was not possible to give a detailed evaluation of every component that affected safety. Therefore, we decided to only review the integrity of those whose in-service failure could contribute to a severe accident. Additional analyses will be carried out in the future as required.

The following components have been assessed with respect to their function, and the consequences of their failure for the listed failure modes:

- Failure of the stainless steel pool liner from stress corrosion cracking,
- Failure of the carbon steel membrane (in the PCR/V) from boric acid attack,
- Failure of the gas lock cover system from stress corrosion cracking or neutron irradiation,
- Failure of the core internals failure (excluding fuel) from neutron irradiation,
- Failure of the PCR/V from tendon and rebar degradation,
- Failure of the secondary side carbon steel piping from erosion-corrosion or thermal fatigue, and
- Failure of certain primary circuit, carbon steel components from boric acid attack.

A detailed description of the failure modes, and the effects of the chemical and irradiation environments that are anticipated during service, are given in Section 6.3 together with supporting literature.

3.4.3 Structural Analysis

The review of PIUS for structural adequacy focused on external events, with primary emphasis on seismic events. Section 6.2 documents the results of this assessment.

In addition to the data contained in the PSID, ABB Atom also provided several technical reports documenting preliminary seismic analyses of the Prestressed Concrete Reactor Vessel (PCR/V) and the Reactor Internals; a preliminary stress analysis report for the Steel Vessel Extension Anchor Flange; and written responses to structural-related questions posed by BNL. This supplementary information was reviewed and incorporated in

the final assessment.

To gain a better understanding of the dynamic characteristics of the PCRV, a simplified computer model was developed and used to study the effects of base support flexibility on the natural vibration frequencies and mode shapes. The level of model detail for the PCRV is the same as that used by ABB Atom in its preliminary analyses. The results of this study and the comparison to ABB Atom's predictions for a "soft" site and a "hard" site are presented in Section 6.2.

Technical input on structural-related issues associated with PIUS operation was provided to both the FMECA and HAZOP studies. In the first case, pertinent FMECA Tables were reviewed and comments and recommendations were provided. In the second case, there was direct participation in the HAZOP exercise.

3.5 Approach to Assembly of Results

At this point, the record of the PIUS study consists of: FMECAs contained in Appendix A, HAZOP session records in Appendix B, and specific deterministic studies consisting of reactor physics/thermal hydraulics, seismic response, missile protection, flooding, fire, material problems and chemical effects contained in Section 6. The assembly process consists of coalescing these three information streams into a single, composite information stream, as shown in Figure 3-5.

The PIUS Intermediate Table (PIT) contains items taken from the FMECA, HAZOP and deterministic analyses. Each numbered item briefly describes the item, its location in the FMECAs, HAZOP records, or deterministic analyses. An item includes criticality and frequency estimates, as well as the safety class of equipment involved in the failure and in the support systems associated with the item.

An item is classed as an initiator if it can start an accident sequence, or as a mitigator if it mitigates an accident sequence. The PIT identifies plant accident initiators and the accident mitigating systems. These, and a knowledge of the plant, provide the information for constructing the event trees, as shown in Figure 3-5.

Figure 3-6 is an illustrative functional event tree for an accident initiator; such event trees are prepared for each accident initiator. An event tree depicts, in a quasi-time sequence, the events following the initiator. An upward branch at a node assumes success of the system identified at the column head; a downward branch assumes failure; no branch means that the system has no effect. The consequences of an accident sequence are estimated on the basis of the amount of radioactive material that may be released in the three classes identified in Table 3-1.

Each branch of the event tree is associated with a particular consequence category, determined by the initiating event and the mitigator failures that occur on that branch. However, it should be realized that assignment of sequences of events into these consequence categories was somewhat subjective and was not based on quantitative calculations of the amount of damage or the amount of radioactive material released.

Table 3-1 Consequence Categories	
Consequence Category	Explanation
CC-1	No fuel damage. Cladding does not leak and fuel can be reused. Radioactivity in coolant may be released.
CC-2	Fuel damage occurs such that fission gases in the fuel escape; the fuel element cannot be reused because of cladding degradation or perforation. Coolable geometry is maintained and no melting occurs.
CC-3	Fuel damage occurs involving relocation, fragmentation or melting.

Figure 3-6 Illustrative Event Tree for an Accident Initiator

3.7 NRCEC Event Condition Analysis

In addition to this conventional qualitative application of PRA to PIUS, we also apply the new NRC Event Condition Methodology (NRCEC) to PIUS. The NRCEC methodology identifies accident sequences according to the frequency of the accident initiator and various combinations of failures subsequent to the initiator. The NRCEC is a valuable method for rapidly identifying systems that may be important to safety. By applying the method to PIUS and comparing the results with the conventional (although qualitative) PRA results, greater insight into the safety of PIUS is achieved and the applicability of the new NRCEC methodology is explored. Section 7.3 describes the NRCEC methodology in more detail, and the remainder of Section 7 shows its applicability to the event sequences that are analyzed.

3.8 References

IEEE Guide for General Principles of Reliability Analysis of Nuclear Power Generating Station Protection Systems," ANSI/IEEE Std 352-1975.

F. P. Lees, editor, Loss Prevention in the Process Industries, Butterworths, London, 1980.

4.0 FAILURE MODES EFFECTS AND CRITICALITY ANALYSIS

4.1 Procedures for the PIUS FMECA

Section 3.2 provides an overview of the FMECA methodology that was applied to PIUS-600. Plant familiarization and the development of FMECA were done, to a large extent, in parallel. The major source of PIUS Information is the PSID (Preliminary Safety Information Document). This document is similar to an FSAR (Final Safety Analysis Report) that is required for licensing nuclear power plants in the US, the form of which is specified in USNRC Regulatory Guide 1.70. The PSID contains a very large amount of information that is diffused through five volumes. This information was condensed to a safety-edited system description (SESD)¹, needed for the FMECA. The result is given in Appendix A in ten parts, with the SESD preceding the FMECA in each of these parts.

The FMECA was performed for each important subsystem, selected from the SESD (in the system classification order). Observations were made regarding safety aspects. The results are contained in a table prepared with 9 columns and about 15 rows filled in similarly to the form shown in Figure 3-2, with the correct identification regarding the table number, analyst's name, date, system, and source of information.

Referring to Figure 3-2, a number is assigned to the left column to identify the analysis of the item described in column 2. Also included in the column is a description of the purpose. This description serves two functions: it provides the reviewer with an understanding of the purpose of the item, and, it reflects the analyst's understanding of the item, thereby aiding in detecting and correcting errors. The failure mode is described in column 3 with suggested causes of failing in this mode in column 4. To fill out column 5, the analyst must determine what would happen to the system and the plant if this subsystem fails in the postulated manner. Having done this, it is appropriate to assign a frequency category for this failure mode (column 7). Column 6 is an assessment of the combination of the failure effect and the frequency, hence, the risk posed to the public and workers by this failure; this is called the "criticality." Column 8 is used to describe mitigating factors for this failure mode, and column 9 is used to record comments and observations made by the analyst in performing the FMECA. It may also be used for describing the scenario related to the failure. In several cases, notes are given to provide more scenario details that can be shown practically as a table.

This process was used for preparing FMECAs for 10 major PIUS-600 systems - from the core through containment. Two hundred and eighty separate analyses were prepared from which 37 potentially significant items were identified for possible further review at a later time, using more advanced analytical methods or additional design information.

4.2 Major Observations from the PIUS FMECA

The following major observations have been identified. It may be that with additional discussion with, and information from ABB-Atom, these items are due to misunderstanding. ABB Atom has reviewed this chapter and has provided some comments about the major FMECA findings described in this section (ABB Atom, 1993). They are referred to in the remainder of this chapter as "ABB Atom's comments". These comments have been taken into account and the major FMECA observations have been re-reviewed under their light. Please note, however, that the detailed FMECA study contained in Appendix A has remained unchanged.

The following list of 37 items is not exhaustive; the FMECAs in Appendix A provide other items that were not included, because they are considered to be generic to nuclear power plants and are already being addressed by the NRC, or because of their small impact on safety.

¹ A safety-edited system description is an edited PSID to include only safety-significant information.

1. Computer Malfunction due to Electromagnetic Interference (EMI)

(A.9-11 items 6&7, A.9-15 item 20) The speed of the main PIUS coolant recirculation pumps is varied from 20% to 105% by thyristor control of the motors. Thyristors are a back-to-back silicon controlled rectifier that goes from non-conducting to conducting very rapidly (of the order of microseconds). This transient from switching multi-megawatts of power results in considerable power in high frequency electromagnetic waves that can propagate throughout the plant and cause EMI in sensitive equipment. PIUS is a computer-controlled plant - the operators do not have hard-wired control over the plant. However, it should be noted that this could result in unexpected problems in one or more of the plant computers.

ABB Atom has more than a decade of experience with BWR plants with internal recirculation pumps and thyristor controlled power supplies. They state that they have eliminated EMI-caused problems by means of proper shielding in the apparatus cubicles, an efficient plant earthing system, and installation with proper separation from the control room and other areas with sensitive and important equipment. ABB Atom does not intend to introduce special electrical noise suppression circuits as primary means of suppressing the EMI problem.

Lightning is another possible cause of EMI, but, again, ABB Atom states that they use, as a standard design, a type of lightning protection such that a lightning strike would not be capable of causing a total loss or malfunction of the computer systems throughout the plant.

Another possible cause of a total malfunctioning of the computer systems is a large electromagnetic pulse (EMP) such as may be caused by a nearby nuclear explosion. However, this scenario is not part of the design of the plant and is not addressed in this study. A separate study would be necessary for this scenario.

Despite the design measures and experience of ABB Atom with the elimination of EMI, since the consequences of computer malfunctions can be serious and since PIUS-600 is a new plant with no previous experience, control measures for EMI may be appropriate to consider.

2. Lack of Redundancy in Scram

(A.7-29 items 1-5, A.7-30 items 6-8) In several places, the PSID suggests that PIUS is scrambled by dropping power to one pump. Apparently there is a designated pump used for scram purposes. The reason that multiple pumps are not tripped is believed to be, to limit the primary flow decrease, thereby limiting the thermal shock from the pool activation. However, one pump scrambling may not provide adequate redundancy.

From the current ABB Supplemental Information package to the PSID, and oral presentations, PIUS has now been provided with a "scram valve system" that replaces the "trip of one pump scram" described in the PSID. This new system appears to provide improved redundancy.

3. Seismic Collapse of the Flow Structure

(A.2-21 item 4) The flow structure of PIUS is not a pressure boundary, and therefore, does not have the strength that it would have if it were a pressure boundary. The flow structure is essentially a long cylinder, constrained at each end with the concentrated mass of the core at the lower extremity. No intermediate constraints are shown on the flow structure.

There are two observations with regard to the behavior of the flow structure under the conditions of a seismic event. The first arises from a seismic occurrence of such magnitude that the inertia of the core will tend to displace it from the central axis of the PCR/V, which in turn will also displace the flow structure and displace water from one side to the other side. On the opposite phase, the structures are displaced in the other direction to displace the water back to the other side. The effect is similar to two coupled oscillators. If the phasing is such that the hydraulic motion is in phase with the mechanical motion, it might be possible to break the flow structure.

The comments from ABB indicate that "In the event of an earthquake, the reactor core inertia will tend to displace it horizontally relative to the core envelope structure and the supporting skirt, but the freedom of movement is very limited and the skirt and envelope structure compound so stiff that any major difference in oscillation is improbable". Furthermore, at the defined SSE level, ABB states that, according to some analyses, there is no need for any intermediate constraints for the riser/downcomer structure to accommodate the movements and stresses. In the event of an earthquake beyond the SSE level, the long structure may start to oscillate, as indicated, possibly with rupture of the structure, at the bellows connection.

The second observation is dependent on the outcome of the first; it is possible that the resulting debris could block the flow path to inhibit core cooling to the point of overheating and failing the core support structure.

In its comments, ABB Atom views the outcome of oscillation and rupture of the flow structure as a mild event, resulting in the lower structure breaking loose and tilting, or at worst a decrease of the inflow through the density lock due to some blockage of the surrounding structures. In all cases, the core will be exposed directly to the pool water, with shutdown and ensured decay heat removal as a result.

However, the worst case outcome of oscillation and rupture of the flow structure may be core melt. The core would fall to the bottom of the PCRV and continue to overheat, weakening the concrete with possible collapse of the core and water in the PCRV into the open space below the PCRV. This would constitute a release inside of containment, but may cause problems in maintaining the containment integrity.

ABB Atom's comments state, as a conclusion, that a worse scenario could be the postulated failure of the lower core support structure. These comments, however, state that, "*in most of the cases*", the core would end up in a coolable configuration with water all around it. This sentence implies that in "*some cases*" the core may end up in a non-coolable configuration, suggesting that the seismic-related scenarios may be important.

4. Procedures for Determining the Release of Non-Condensibles

(A.1-10, item 13) Radiolysis of water produces hydrogen and oxygen. Such radiolysis gases collect in the pressurizer dome and are bled periodically as a manual operation. There is the remote possibility of the buildup of an explosive gas mixture in the pressurizer, which forms part of the pressure boundary of the pressure vessel. ABB states that the risk of building up an explosive gas mixture in the pressurizer is small and does not require immediate attention.

5. Failure of Plant Status Information

(A.7-41 items 41-45) The PSID is not clear regarding the independence and redundancy of the plant status indications in the control room. An observation is that a common cause failure could result in the loss of significant indication of important parameters that are needed to know the plant status in a degraded condition and for taking corrective actions.

PIUS does have status information available on different levels: on systems and components level via the process communication equipment and associated man-machine communication equipment, on system and plant function levels via the supervisory computer system and associated man-machine communication equipment, and on safety equipment, systems and functions via the safety parameter display system.

However, there still exists the possibility that a common cause failure such as fire, loss of power or an inhabitable control room, could make several or all of these levels of status information unavailable or inaccessible.

An argument may be made that because of the inherent safety of PIUS-600, it is not necessary for the operators to know its status or take corrective actions.

6. Location of the Emergency Control Room (ECR)

(A.7-39 items 33-36) The purpose of an emergency control room is to be able to shutdown and monitor the status of a plant when the main control room is uninhabitable. The ECR serves the same function as the remote shutdown panel in current reactor plants. The design of PIUS places the emergency control room in the lower part of the reactor building. It is thus physically well protected against external impacts, and it is also earthquake proof.

However, the habitability of this room must be ensured at all times. Given that the main moving force of flooding is gravity, placing this room almost at the bottom of the reactor building seems to make it particularly susceptible to flooding. Important areas for consideration include the access to the emergency control room, as well as all the factors which make this room habitable for relatively long periods of time, such as its temperature.

7. Failure of Emergency Control Room Reactor Status Display/Controls

(A.7-39 items 33-36) PIUS does not appear to have any hard-wired instrumentation and control circuits. This is a major difference from current plant designs. This absence of hard-wired instrumentation and control in PIUS may impact the operability of the status indicators and may effect the ability to control the plant and/or scram the reactor actively both from the MCR and from the emergency control room, under conditions of an unlikely multiple computer inoperability.

8. Gas-Lock Blockage

(A.2-21 item 2) Because the main cooling pumps can only be reduced to 20% of full flow, it is necessary to block the flow of the pool loop. This is done just above the lower density lock by an inverted cap (dome) that can be filled with nitrogen gas. The pressure required to eject the gas is greater than the differential pressure across the blockage. Gas is injected into the cap to cause the flow blockage and removed by a bleed line. When the reactor power has increased to a level that allows the removal of gas cap, the gas supply is stopped and all gas bleeds off.

An observation with regard to the Gas Lock system (3225) is the possibility of a gas accumulation exceeding the gas release rate, due to one or more of the following factors: foreign and loose material blocking the gas release holes, boron accumulation in the holes, failure of the gas injection control (presumably, in power operation, there is no bleed from the gas cap). The gas accumulation could prevent activation of the pool loop, which constitutes the emergency cooling system for the reactor. The gas accumulation may also be due to a nitrogen gas bubble which has not been bled.

This accumulation can also be due to the radiolytic gases that are produced, and under degraded core situations, or to other gases that are released. However, as mentioned in ABB Atom's comments and in Appendix A of this report, it will probably be difficult for these gases to migrate under the cap and be trapped there

Also, according to ABB Atom's comments, the gas lock would not be capable of preventing the pool water ingress in the event of transient during startup, because an activated gas lock represents a total pressure difference of about 3 kPa.

Since unobstructed pool loop flow is imperative for PIUS-600, potential questions are:

- a) What is the maximum pressure difference that can exist in the gas lock that cannot prevent the pool flow from entering into the core?
- b) How was the value of 3 kPa obtained?

- c) Is there a chance for a (nitrogen) gas bubble to produce a larger pressure difference than the quoted 3 kPa?

Furthermore, if the reactor is normally operating under pressure, but a gas bubble has remained trapped in the gas lock cover *after* startup, the pool flow required to scram the reactor may be jeopardized by the obstruction presented by the trapped gas bubble.

Due to these observations, the functioning of the gas lock system may need to be studied under various circumstances to make sure it does not represent an obstacle to the pool flow when this flow is required.

9. Upper Density Lock Blockage

(A.2-23 item 6) Blocking a flow path in any part of the circuit will block the flow. Loose material may enter the flow structure of the reactor above the lower density lock during refueling; material from core degradation may be released. In either case, material entrained by the upward coolant flow may be deposited at the entrance to the upper density lock and either restrict or block the pool loop flow. If such were to result from core degradation, the effect would be especially bad, because at this time, the pool loop is needed the most. (The primary coolant communicates with the pool water in the pressurizer. However, this is not a replacement for the upper density lock, because the density gradients are not correct for natural circulation flow.)

PIUS has loose parts monitors, which should be helpful in detecting foreign objects that may enter at refueling. Furthermore, objects entering at this time will tend to be entrained in the primary loop flow and will be removed from the flow structure. (They may be deposited in the region where the primary exits to the core, but they would not be blocking the lower density lock.)

The design of the current upper density lock presents a flat entry to the density lock tubes on which blocking materials may collect. ABB Atom's comments state that the area at the top of the upper density lock is enormous compared with what is really needed, and for this reason they conclude that the flow path is not very important.

However, the observation is related to the entry to the upper density lock tubes, and not to their top.

10. Core Flow-Distribution Balance

(A.2-24 item 2) The PIUS core has less flow impedance than conventional LWRs; the flux flattening must be done by design and construction; it cannot be modified in operation, since PIUS has no control rods.

Preferential deposition on the fuel elements, fuel swelling or bowing may result in variations in temperature across the core that are regenerative, i.e. heat imbalance causes more variation in local impedance, which causes more heat imbalance. ABB Atom's comments state it is not likely that the PIUS core would be more susceptible to this than the cores in current plants.

However, since the PIUS core has less flow impedance than conventional LWRs, then deposition on fuel elements, fuel swelling or bowing, would have a greater impact on the case of PIUS than on the case of conventional LWRs.

11. Failure of Piping in the Reactivity Control System

(A.3-14 item 2) Check valves V102, V202, and V204 prevent reverse flow in case of breaks in lines in the borated water, filtered water, or distilled water systems. The check valves are needed because the pipes are penetrating the containment. These check valves represent the second isolation valve for the three penetrating lines.

If additional check valves were placed closer to the reactor, they could protect against small breaks in system 312 piping as well. A check valve cannot be located in the letdown line to system 3125.

12. Failure of Letdown Water Heat Exchanger

ABB Atom's comments clarified this item such that it is no longer of interest.

13. Valve Failure and Discharge through the Upper Density Lock

(A.3-17 item 2) Failure of the motor-operated valve in system 3126 will result in transient-caused coolant expansion and discharge through the upper density lock, which is apt to be followed by a scram when the water volume contracts. The effect of this scenario would be to contaminate the pool with primary water and an operational upset.

However, in the modified design described by ABB in the current PSID Supplemental Information package, the need for the temporary letdown system has been much reduced, since the expansion volume for the primary loop water has increased considerably. According to ABB Atom's comments, the letdown system may be found to be totally unnecessary.

14. Large Pipe Break

(A.3-7 item 1) A Large Loss of Coolant Accident (LLOCA) resulting from a guillotine break of the largest primary pipe is a design basis for conventional LWRs. A LLOCA in PIUS would result in coolant being discharged into the containment for collection in the wet and dry wells, the pressure would rapidly depressurize, voids would form in the core, increasing the thermal pumping action, further disrupting the flow balance, and activating the pool loop for core cooling by natural convection.

One possible observation is possible fuel damage from exceeding the critical heat flux (CHF) from the rapid depressurization. The PIUS linear heat flux is half that of conventional PWRs, but its pressure is half of PWR pressures, which places it within about 2 MPa above the critical heat flux. The PSID (page 15.9-12) describes RELAP5 code calculations that were performed by ANSALDO Spa of Genova Italy, suggesting that cladding temperatures in a LLOCA do not exceed normal operating temperatures. However, the comments by ABB Atom state that "As stated during the meetings in Västerås and Washington, D.C., the more recent LOCA calculations show that there will be temporary dryout following a double-ended cold leg break, but the duration of the *overtemperature* (with a maximum value of about 700°C) will only be in the order of 5-10 seconds. ABB Atom, therefore, claims that there will not be any fuel damage due to the exceeding of the critical heat flux".

The event trees of Section 7 have included the possibility of exceeding CHF and thus causing core damage.

15. Failure of Safety Valves

(A.5-6 items 1 and 2) Overpressure protection of PIUS is provided by four relief valves, four safety valves, and ultimately tendon stretching. An observation was made based on the PSID that the relief and safety valves are similar and the complex servo system is controlled.

Common practice uses safety valves operating on diverse principles from the relief valves.

With regard to diversity, ABB Atom's comments state that the safety valves are pressure-actuated and the relief valves are electrically actuated.

With regard to reliability, ABB Atom's comments state that they selected the servo-controlled safety and relief valve design because they have a proven reliability record supported by its successful use in ABB Atom nuclear power plants.

Therefore, according to the last two paragraphs, this area appears to be appropriately addressed.

16. Failure of Relief Valves in Long-Term Cooling

(A.5-6 item 2) Given a total failure of all pool cooling systems, a week is required for the decay heat to boil away the water in the reactor pool. After this time, additional water may be provided by means of a fire hose connection external to the containment. A manually operated valve located inside of containment must be opened to supply water to the reactor pool. BNL observed that if the reactor is under pressure, this valve should not be opened because the pressure would rupture any hose connection, and primary steam/water would blow into the environment. Before this can be done, the relief valves must be opened so the pressure will discharge to the drywell. Under highly disrupted conditions, the operation of the relief valves cannot be assured.

However, ABB stated that in case the normal depressurization procedure with the relief valves fail, the manually operated emergency blowdown valves, with actuation from outside the containment, provide a diverse backup for depressurizing the system. Therefore, the failure of relief valves in long-term cooling is not a particular concern.

17. Containment Pressure Relief

(A.6-7 item 1) The PIUS primary contains 870,000 gallons of water. If this volume were converted to vapor, depending on the containment leak rate, the pressure would likely rupture the containment. PIUS, like conventional LWRs, has no means for relieving the containment pressure before rupture.

Relieving the pressure is better than rupturing the containment even though there would be an atmospheric release, unless some off-gas processing or filtering is used for the release.

18. Seismic-Caused Failure of Reactor Service and Spent Fuel Storage Pools

(A.6-7 item 3) The reactor service and spent fuel storage pools (belonging to system 152, "Containment") are in the upper part of the reactor building. In a seismic event, the accelerations tend to be the highest in the upper part of a structure. There is fuel handling and transport equipment in this area, which may move about under seismic accelerations and damage structure. Lastly, the thickness of these basin walls is considerably less than the PCR (but it is under pressure). Should the spent fuel storage pool drain, the fuel would overheat - depending on the time since the fuel was removed from the reactor.

Table 3.A of the PSID states that the seismic classification of system 152 is "I". Seismic Category I is the highest class with the highest demands. However, the classification given in Table 3.A of the PSID is "tentative" and a final judgment cannot be made following this preliminary information. On the other hand, since the containment serves very important safety functions it is expected that it will definitely comply with Seismic Category I.

19. Failure of Reactor Trips and Containment Isolation

(A.7-29-36 items 1-26) PIUS is designed to actively scram from power and hot shutdown, and isolate the containment. There are many possible causes of failure: human error, miscalibration, set point drift, electronic failure and welded contacts. According to ABB Atom's comments, PIUS is designed with scram and containment isolation functions like any other LWR plant. However, ABB Atom's comments of this item do not address the following points:

- a) All scram modes (active, manual and passive) depend on the borated pool water entering into the core. The suggestions for improving the scram function reliability are redundancy and diversity. In fact, as a comment to another point (item 30), ABB Atom's comments recognize the need for another dedicated scram system, letting borated reactor pool water into the primary loop on the suction side of the RCPs.
- b) With regard to the containment isolation, it is necessary to make sure that this function has the required redundancy and diversity. In particular, current LWR plants do not have the strong dependence PIUS has on the computer systems. Therefore, a hard-wired, manually initiated, containment isolation independent of the computer systems is suggested.

20. Failure of Visual Display Units (VDU)

(A.7-38 items 29-32) There are VDUs for the reactor operators, shift supervisors, plant operator, and for indicating the plant status (SPDS). The VDUs operate by the same principles, hence, they may be subject to common cause failure. Multiple VDU failure might be such a psychological shock to personnel, trained on the advanced PIUS control room, that errors may be committed. Additionally, there is some status information on the display panels, but the information in the absence of the VDUs seems insufficient for plant status monitoring, and perhaps, even for plant shutdown.

ABB Atom's comments have made clear, however, that a common cause failure of the dozens of VDUs in the main control room is highly unlikely. In addition, even in the highly unlikely event of a multiple VDU failure situation, the PSID states that the reactor can manage for many days without supervision because of the large water inventory in the reactor pool and the passive pool cooling system, thereby allowing ample time for the operators and maintenance personnel to get the man-machine communication system back to normal again.

However, even though there is ample time for recovery of the man-machine communication system, it is necessary to monitor at least the safety-critical parameters of the reactor and of the plant, so that their status is known while the recovery actions are performed. Knowledge of the status of the reactor and plant would also decrease the stress on the people performing the recovery actions.

Operator training typically includes procedures to be used in the absence of the VDUs. Diverse (hard-wired) indications of critical plant parameters would be very beneficial to assure safe shutdown and monitoring.

21. Operation on One Central Computer

(A.7-40 item 37) PIUS has redundant central computers, as well as redundant process computers. The PSID is mute regarding operation without one member of a redundant computer pair. Considering the large number of computers, it might be highly restrictive to require that all computers must be operating for the plant to be operable. However, they serve critical functions which is the reason they are redundant.

Since the plant and its processes control, and the gathering and interpretation of information are functions performed by computer-related equipment, failure of one or both of a redundant pair of computer-related equipment must be fully managed from a safety point of view. If a single unit of a pair of computer-related equipment fails, adequate redundancy may be provided by:

- a) bringing the redundant unit into operation to cover the functions of the failed unit,
- b) not interfering with the operation of the "good" unit
- c) annunciating for the plant staff to acknowledge the failure and take corrective actions.

A more clear presentation of failure of various computers in PIUS may be useful.

22. Criteria on Computer Errors that Cause a Switchover to Backup

(A.7-40 item 38) PIUS has redundant central computers, as well as redundant process computers. In this redundant arrangement, the active computer performs calculations, display, and control; the standby computer performs computations, but does not perform display or control functions. If an error is detected in the active computer, the standby computer switches over to providing the display and control functions. Information on the method for detecting errors, and switchover to the standby computer, was not available for this review. Some errors could go undetected and operation could continue on a malfunctioning computer that provides erroneous information or control.

Hard-wired control of critical functions and hard-wired information display, in addition, to computer control and display, would provide valuable diversity to the monitoring and the control of critical functions.

23. Susceptibility of Display Panels and Data Links to Cable Failure

(A.7-41 item 42, A.7-42 item 47) PIUS, like other nuclear power plants, displays status information by display panels. Multiple-conductor cables, which also are used as data links to process computers, send information to these panels. Severance of a cable or connector will fail many display panels and/or disrupt a whole data link.

However, ABB Atom's comments have clarified that inter-computer communications are using serial links (multiplexing) based on the HLDC (High Level Data Control) standard and multiple-conductor cables are used only for a few applications. They regard the HLDC serial links to provide a fully acceptable technical solution.

Nevertheless, diverse methods for transmitting information, such as fiber-optics in conjunction with electrical cables, would provide added diversity and/or redundancy.

24. Operator Control via Computer

(A.7-41 item 44) In PIUS, the operator only indirectly controls the reactor; the operator controls the computer which, in turn, controls the reactor. Indeed, the operator does not have a hard-wired connection that will cause the power to be dropped on a primary pump to cause a scram.

ABB Atom's comments state essentially three arguments against the implementation of a hard-wired scram: 1) the operator will be provided with an advanced data display system to monitor plant operation and he will always be in a position where he can assume control of the operation, 2) the passive nature of PIUS does not require much in the way of operator intervention, and 3) the same type of computer-based systems are presently operating in the waste handling systems of several of their facilities.

Areas to consider are: that the reactor operator be able to control the reactivity independently of the computer and be able to shutdown the reactor by reactivity control rather than thermal shock accompanying a scram; and to provide a hard-wired connection to a breaker that can interrupt the power on a primary pump and produce a manual scram independently of the computer system.

25. Software Error in Functional Processor (FP) or Central Computer

(A.7-42 item 49) PIUS is controlled and operated by computers; computers are controlled by software. It is possible to have an error in software that will go undetected until some combination of operating mode, perhaps associated with high data rates, causes an error. The validation and verification of software is difficult. It is probably not possible to absolutely assure against software problems.

ABB Atom's comments are in agreement with the difficulty to detect errors in a large, complex computer structure. They describe their approach to minimize the software validation and verification problem: the use of simple, standardized micro-processor units for which a reliable validation and verification is possible. In addition they say that, essentially, established technology will be used and that the real new feature is the signal compacting technology.

Furthermore, ABB Atom's comments state that the addition of diversity through analog equipment, in advance to computers, will improve the functional reliability of the control equipment. However, they also state that quite probably this addition would result in diverging control room information on process measurements and status, may place the operator in a difficult position.

However, ABB Atom's comments do not address the following points:

- a) The main concern is the failure of software. The use of several micro-processor units may increase the reliability mainly from a hardware point of view, but has little effect to improve the software reliability.
- b) If several computers or micro-processor units use the same software, and the software experiences a failure, it will very likely affect all the computers or micro-processor units running it. This is a serious problem, since it is a common cause failure. However, it also depends on the severity of the failure, and it is minimized for the case of the micro-processors using ABB Atom's approach, but a strict QA process needs to be followed.
- c) The traditional LWRs do not have such a strong dependency on digital equipment as PIUS does. Since PIUS is, to a large extent, a computer-controlled plant, a formal review of software validation and verification is important.
- d) It is true that the addition of analog equipment could result in diverging control room information on process measurements and status. However, a significant deviation from the indications provided by digital (computer) equipment and analog (hard-wired) equipment would be indicative of problems in any one of this equipments or in both. This discrepancy may place the operator in a difficult position, but it will also make him aware of the situation, and he can take proper action, like running some tests and dispatching a troubleshooting crew. Best of all, the operator will not have to have complete faith on a system (computers) that might apparently be functioning properly, but that in reality is functioning incorrectly. Furthermore, if the operator is properly trained to expect this kind of situations, the discrepancy between digital and analog equipment is not likely to place the operator in a difficult position, but on the contrary, it can be used to troubleshoot these equipments.

Therefore, diversity through hard-wired connections, in addition to computer control, will protect against software failures in the computers. This protection is not in the sense of making the software more reliable (it will not decrease the number of errors in a piece of software), but it will give some diversity and redundancy to the digital equipment of PIUS, which has a central role in the operation and safety of the plant.

26. Error in Boron Concentration Monitor

(A.7-45 items 52-55) The boron concentration in the pool water and in the primary coolant is measured by the neutron attenuation in samples of the solutions. An error in the measurement could result in under-boration of the pool or primary coolant. If the primary coolant is being under-borated, the power level will be increasing and higher than expected. If the pool were significantly under-borated, it would not perform its passive scram function.

ABB Atom's comments state that there is another diverse, automatic boron-measuring system operating at intervals using chemical means. In addition, it states that sampling and laboratory analyses represent a backup means for controlling the automatic measurement systems. These systems may address the above concern.

27. Function Processor Computers Control the Reactor Power

(A.7-50 item 70) Reactivity control in PIUS is performed by doubly-redundant computers. These are susceptible to software and hardware failures (see 21, 24 and 25 above). A computer problem in reactivity control could cause a reactivity excursion before causing a scram. Furthermore, it is not clear that a failed function processor (FP) would place the reactivity control in the safest configuration.

ABB Atom's comments essentially state that the worst possible outcome of this situation would be a normal scram. In addition, the monitoring of reactor power, riser temperature and pressurizer pressure is handled in an entirely different function processor, and they state that it cannot be "reasonably" assumed to have failed simultaneously with the other one.

Nevertheless, we suggest the following observations be addressed:

- a) Even though the situation postulated in the first paragraph of this item ends with a normal scram, we still suggest examining the design to assure that FP failure will: 1) be announced and not remain hidden until it causes an scram, which could be confusing to the operators, 2) not result in boron dilution during operation, 3) at least not interfere with the scram signal to isolate the injection line for deborated water, and 4) not allow reactivity excursions to occur due to the failure of the FP.
- b) As stated in Item 25 above, a software failure can fail several processors, even if they are separated.

28. Mode Error

(A.7-50 item 71) Specifying the operating mode of PIUS determines many of the computer-controlled operations that follow. The mode is specified by operator input to the computer. Hence, human error is a possible cause of wrong mode specification; computer error is another cause.

According to ABB Atom's comments, the *mode*, and more importantly, the *change of mode*, does not have an important safety significance. If a mode is selected incorrectly, the worst outcome would be a reactor scram, and the most likely result would have no significant effects at all.

Nevertheless, display in the control room, of the mode specified by the operator, and the mode that the computer is specifying to the various algorithms, could be helpful as additional information to the operator.

29. Turbine-Trip Reactivity Excursion

(A.8-5 item 2) Approximately one-third of the heat from the reactor is removed by the turbine to produce electricity. A turbine trip abruptly removes this heat load, and the temperature of the reactor primary coolant rises. The very strong negative reactivity temperature coefficient of PIUS results in a *negative* reactivity excursion, which is followed by either an active or a passive scram.

ABB Atom's comments essentially state that the negative temperature coefficient of reactivity makes plants, including both PIUS and traditional LWRs, more "stable".

BNL's observation does not question the "stability" due to the negative temperature coefficient of reactivity. Besides, the reactivity excursion postulated is a *negative* reactivity excursion, which is followed by either an active or a passive scram. BNL's observation is related to the implementation of an active scram following turbine trip, if the turbine steam bypass system could respond quickly enough, and were effective enough so that the rise in temperature did not result in a significant *negative* reactivity excursion.

30. Scram-Caused Thermal Shock

(A.7-29,30 items 1-8) According to the PSID, PIUS scrams as a result of a mis-match between the primary flow and the buoyancy flow from heating the coolant in the reactor; the mis-match is made up by pool flow. Pool water ingress yields a reactivity reduction due to the boron the cold pool water yields a reactivity insertion. The cold water cools down the core coolant, decreases the "buoyancy flow", and decreases the inflow rate of pool water. Then, "old" warm, low-borated primary coolant is pumped to the core; power and temperature increase, and more pool water enters. In about one minute, the temperature of the reactor and components change by about 400°F.

However, as noted in information to the NRC, there is now a re-designed active scram system in PIUS, where valves open to let borated reactor pool water into the primary loop on the suction side of the RCPs. This modified design avoids the thermal shock described in the previous paragraph.

31. System 630 Breaker Failure Prevents Emergency Power Use

(A.9-10 item 4) When there is a loss of the high voltage grid, the main circuit breaker should open, thereby allowing the generator bus system (system 631) to feed power to the 6.0 auxiliary power AC system (system 642) in house load operation, if this feature is included in the plant design. This transfer is backed up by the plant startup transformer (642 T2). If the main circuit breaker does not open, then the transfer to the back up source (transformer 642 T2) may also be affected due to the interlocking logic of the relays. System 642 would remain deenergized until this situation is fixed. According to the original design of PIUS described in the PSID, the RCPs would be deenergized and the reactor scrammed, but it is very important that the plant recover AC energy as soon as possible.

However, the kind of transfers described in the previous paragraph are standard in the electrical industry and ABB Atom for sure will take into account these kind of events. Manual disconnection of the main circuit breaker or other measure to enhance continuity of electrical supply to the 6.0 kV auxiliary power AC system could be beneficial.

32. Surge Capacitors in System 630

(A.9-11 item 6) System 631 uses surge capacitors on the generator busbars as protection against voltage surges, e.g. due to lightning currents. If these capacitors short, power cannot be supplied from system 631.

If the capacitors were fused then, even if they short, the fuse can be blown and the power supplied from system 631.

33. Loss of DC and Uninterruptable AC

(A.9-18 item 29) System 670, "Battery-Backed Power Supply Systems", is comprised of two subsystems: 673, "Safety-grade 110V DC auxiliary power supply system" and 677, "440V battery-backed auxiliary power AC system". Each of these subsystems consist of two identical trains.

a) Subsystem 673, "Safety-grade 110V DC auxiliary power supply system"

In the case of the loss of a train of system 673, the plant can still be monitored and controlled by means of the normal instrumentation, fed with power from system 677, and by the channels fed from the other 673 train. This system is considered a safety-grade system and each train is located in separate safety-grade equipment rooms.

Naturally, an issue for Subsystem 673 is a poor maintenance performed on both trains, increasing the possibility of simultaneous failure of both trains. We respect the good maintenance procedures executed at ABB Atom facilities, but given the serious consequences of the loss of both trains of subsystem 673 (control and monitoring of the plant would be at least partially lost), we note the need for good training of the maintenance crew and for an effective program of quality assurance (QA) for maintenance activities.

b) Subsystem 677, "440V battery-backed auxiliary power AC system"

Loss of a 677 bus will trip the turbine and scram the reactor. The reactor can be kept at regular cold shutdown by the other 677 bus. Total loss of system 677 will activate the inherent self-protective safety features of the reactor which ensure a reliable reactor shutdown and decay heat removal. The plant may suffer capital investment losses, however.

Again, an issue is the loss of both trains of this subsystem. In addition, since, according to the PSID, the loads for this subsystem have not been completely determined yet, it is not possible to assess the extent of the consequences in case of total failure of this subsystem.

This subsystem is not a safety-grade system and the PSID does not mention the location of each train, so the susceptibility to common hazards cannot be evaluated at this point. However, general observations can be elaborated: the same as those for subsystem 673: good training of the maintenance crew and an effective program of quality assurance (QA) for maintenance activities, a risk analysis of susceptibility to common cause failures and a risk analysis of susceptibility to external events, such as fire and explosion.

34. Fire in the Cable System

(A.9-21 item 43) The cabling system is passive, but may be susceptible to common causes of failure, such as fire.

No specific fire vulnerability has been identified for PIUS, but it is an aspect that deserves attention.

35. Refueling

(Insufficient detail in the PSID precludes a FMECA of refueling) The refueling of PIUS requires disassembly of the pressurizer and the upper three flow structures. The fuel is removed by grappling under about 20 m of water for each spent element; the fuel is inserted by reversing the process. This process will require some time thus, it is possible that a foreign object may enter the primary system, which could produce local flow disturbances or even large disturbances, depending on the size of the objects.

Therefore, there are two observations regarding refueling:

- a) More detail is needed in the PSID.
- b) Refueling operations for traditional LWRs do not have to be carried out through 20 m of water, as is the case of PIUS. Therefore, the time for refueling in the case of PIUS might be longer. Consequently, assuming the risk of dropping foreign objects into the pool is mainly a function of

time, then this risk would be higher for the case of PIUS. In addition, the worker exposure to radiation could also be higher.

ABB Atom states that, given that the pool of PIUS is larger than that of current PWR plants, the risk of foreign objects should be smaller. However, given that the work of refueling is done exactly above the core, then the risk of a foreign object falling into the core is not significantly different than that for current PWR plants.

Therefore, the main observation in this case is the need for detailed information about refueling for further study and, particularly, detailed information about the refueling time and worker exposure to radiation during refueling.

36. Primary Pump Startup and Shutdown

(A.4-9,10 item 3) The key PIUS safety principle is that a balance between the buoyant flow and the pumped primary flow prevents flow from the borated reactor pool. The PSID does not discuss degraded operation on less than four pumps because the loss of one pump at normal power operation implies a reactor scram.

It appears that primary pumps cannot be brought on line or removed from service without scrambling PIUS, or shutting PIUS down by boration. The inability to operate degraded, does not appear to be an issue with PIUS, but it may result in less availability than current LWRs.

37. Blockage of the Lower Density Lock

(A.22-1 Item 3) An important failure mode to consider is if the lower density lock is blocked for whatever reason.

ABB Atom's comments first state that the blockage of the lower density lock is an extremely unlikely event due to sizing of support skirt holes and total inlet area. As long as 10% of skirt hole area is unblocked, functionality is not affected. Further, ABB states that "Simulations of sequences with totally blocked lower density lock have been run, and the results indicate that the reactor shuts down and that a natural core cooling loop will be established through the upper density lock (down the riser, through the core, up the downcomer, also)".

In addition, ABB Atom's comments state, as a comment to another item (No. 30) that now there is a dedicated scram system, letting borated reactor pool water into the primary loop on the suction side of the RCPs. This new scram system seems to be an effective way to cope with a postulated blockage of the lower density lock.

References

ABB Atom, Letter to Mr. Dino C. Scaletti, USNRC, May 11, 1993, Subject: Comments on BNL Draft Report.

5.0 DESCRIPTION OF THE HAZOP APPLICATION

The HAZOP method is given in general terms in Section 3. This section describes, first, how HAZOP was applied to search for physical states of the plant which would constitute a hazard, and second, how HAZOP was applied to search for hazardous potential errors in the startup procedure. For present purposes, it was necessary to adapt the method in certain respects. These adaptations are described here.

The main reason to consider HAZOP for review of PIUS is that it focuses on physical variables. The physical state of the PIUS plant is considerably underspecified by a listing of its component states; because FMECA focuses on component states, while HAZOP focuses on physical variables, the two methods are complementary.

5.1 Adaptations of HAZOP Methodology to the Study of the PIUS System

5.1.1 Technical Adaptations

The present application is not typical of chemical process industry applications.

Most process systems can be portrayed reasonably well by line diagrams. Flow in a given segment of pipe is a well-defined concept. However, in the PIUS design, there are patterns of circulation which cannot be characterized as simple flow or no flow. In the core region, for example, flow anisotropy may be significant. Therefore, some thought was given to the structure of the guide words to consider physical process variables which are less simple than flow. These are summarized in Section 3.

A closely related decision was made to base the HAZOP of the primary loops and the pool on a thermal-hydraulicist's nodalization scheme, supplemented by detailed enlargements of selected regions, rather than basing the discussion on a pseudo-one-dimensional line diagram representation. This decision had an interesting and very favorable effect. Because the thermal hydraulicists were part of the HAZOP team, the rest of the team was immediately updated on modelling issues, and the thermal-hydraulic modelling was changed significantly and repeatedly as a result of these interactions. These modifications include not only enhancements to deal with effects which were initially left out, but appear to be important; they also include changes to the model created by an improved understanding obtained during the team's discussions. Section 6 presents the nodalization diagram emerging from the interaction between the HAZOP methodology and the thermal-hydraulic modelling.

5.1.2 Programmatic Adaptations

According to the description given in Section 3 of HAZOP, as practiced in the chemical process industry (CPI), the team members are supposed to be plant workers who are very knowledgeable. In the present case, there is no PIUS plant, and the prospective vendors are (a) not available and (b) not necessarily appropriate choices for inclusion in a safety study sponsored by a regulatory body.

Therefore, this HAZOP study had to be conducted by a team whose knowledge of the system is based on reading conceptual design documents. The team had to be briefed by other analysts whose knowledge is also based largely on reading.

The program plan was based on an in-house team, who had already been involved in reviewing the PIUS design report, and who had especially been involved in the in-house development of a simple, but useful thermal-hydraulic model. This latter development was to ensure that when the team was discussing whether more flow in segment "x" could result from less temperature in segment "y," there would be a body of in-house work informing the team's judgment.

It was initially proposed to study four operating modes: normal power operation, shutdown, startup

(transition from shutdown to full power), and the transition from full power to shutdown. Additionally, the team hoped to find time to analyze selected transients: that is, to refer deviations to a state which was initially off-normal, rather than to impose deviations onto a state which is otherwise normal. These analyses would begin to address multiple-failure scenarios.

Based on analyzing one PSID operating mode per day, and on a solid foundation of knowledge of the PIUS design, this exercise was expected to take about one week. However, more effort was necessary, because the presumed solid foundation of knowledge was not initially available. In our application, the HAZOP has been a means to the end of a collective partial understanding of system operation, rather than the other way around; this has considerably slowed down the team's effort to generate tables of HAZOP results. It would be misleading to say that lack of resources constrained the HAZOP; rather, the effort was stopped, because we found that it is not cost-effective to continue the HAZOP until more detailed information becomes available. The need for a validated plant dynamic model to support discussions of system phenomenology was clearly identified as necessary. Therefore, the HAZOP only covered selected portions of the system, and only for full-power operation and a trial HAZOP of the startup procedure.

These limitations should not be considered as an indictment of the HAZOP method, but rather of the limitations of proceeding with the program on the basis of limited information. The present methods appear to be capable of fulfilling the original objectives, at a level of effort commensurable with that originally allocated, given a suitable information base, a plant analyzer, and a team which is conversant with both.

5.2 HAZOP of PIUS Plant During Power Operation

The guide words were described in Section 3; the PIPA nodalization is described in Section 6. Referring to the PIPA nodalization, the portions of the PIUS design which were analyzed include the following:

- Upper and Lower Downcomers,
- Core Entrance Plena,
- Core Region,
- Pressurizer (pool level, riser funnel, siphon breaker),
- Pressurization System (boiler, feed pumps, ...).

HAZOP log sheets from these sessions are included in Appendix B. Deviations having high or medium criticality are carried over into the PIT, and comments on these also are summarized in Section 5.4.

5.3 HAZOP of Startup Procedure

As discussed below, HAZOP can be applied to analyze procedures as well as systems. The start-up procedure was chosen as an example, partly because the analyst knew the French scenario, (see Section 5.4.6) and partly because few procedures are described, even at a high level in the PSID.

5.3.1 Application of HAZOP to Analysis of Procedures

Although somewhat less standard than the application of HAZOP to straight systems analysis, HAZOP of operational procedures is becoming increasingly popular. The idea is that a procedure can be flowcharted, which resembles a system in an abstract sense. The procedure has a design intent at each node, and deviations from this design intent for a procedural step can be postulated by analogy with HAZOP for systems. The design intent of a procedural step is typically to accomplish a particular system state at a particular time, based on particular cues, and the corresponding deviation will be to overshoot, undershoot, not perform the step at all, do something else instead, perform the step too soon, or perform the step too late.

Some instances in the above list correspond to errors of commission, identification is widely believed to

be beyond the state-of-the-art.

In the present application, the start-up process (based on the description in Section 34.4 of the PSID) was analyzed. Numerous potential operator errors were identified, most of which either constitute operational delays or inconveniences, or pose challenges which the safety systems should easily meet. Possible exceptions comprise instances in which unborated water could be injected without forced loop flow; this circumstance creates the possibility of a reactivity anomaly, and is discussed further below.

HAZOP log sheets from this exercise are also included in Appendix B.

5.4 Overview of Results

Major design and procedural results are summarized below. A key result of a different type is the significant input provided by the HAZOP team to the phenomenological modelling effort. This result is discussed further in Section 6.

5.4.1 The Q-List Observation

The Q-list observation relates to the problem of deciding what design attributes must be Q-listed (i.e., designated safety-grade) to achieve the level of safety performance which is claimed. The conclusion that plant response is satisfactory in all cases may rest on the assumption of initial conditions, whose realization is accomplished by systems which are presently non-safety. If the plant's response is found to be sensitive to those initial conditions, one would need to consider taking credit for pre-accident performance by the systems which maintain those conditions. However, changing the designation of those systems might have a profound effect on the licensing review. Further, the actual designation of all PIUS systems is not completely clear, as is discussed in Section 8. A detailed evaluation of this issue would need clarification of precisely what is safety grade and also a validated plant model to systematically explore this area.

5.4.2 Depressurization Transients

The question has been raised whether it is possible to achieve a condition wherein the core cannot be wetted because the heat flux is too great to allow wetting at the prevailing system pressure. This question can be broken down into two parts: (1) whether there is a point in state-space at which this is true, (2) if such a point exists, whether it is mechanistically reachable from any normal operational condition. The first of these questions is beyond the scope of HAZOP; it is our job merely to remark its significance, and recommend code runs. HAZOP can begin to address the second question if we know the coordinates of this phase-space point. Presumably, low pressure ("low" depends on context) is part of the scenario. These considerations suggest that depressurization transients be viewed as a group, with this question in mind.

The preceding discussion is not meant to imply that global depressurization would alter the logic of density lock operation. Rather, the question is whether there is any particular mechanism of depressurizing that might invalidate the existing claims for plant response. Modelling the reactivity following a density lock challenge is not trivial; the reactivity does not decrease monotonically after a scram. Thus, great care is necessary to reach conclusions about this area. No conclusion is reached here, because a debugged plant analyzer was not available.

The causes of rapid depressurization are the following:

- Primary Break,
- Break in the Pressurizer Steam Space,
- Break in the Steam System Piping,
- Inadvertent Open Relief Valve (IORV),
- Break in the Steam Dome Bolts,

Smart Break of the Siphon Breaker Injecting Cold Leg Water Into Steam Space, Rupture of the Steam Generator Tube.

These are Category III in frequency, and have "high" criticality in the sense that they should be further examined.

5.4.3 Behavior of Boron Concentration

There will be a slug of unborated water in the riser funnel, as a result of the continuous condensation of steam. There appears to be break locations, which would transport this slug through the core on its way out from the break. Unless this slug were well-mixed with properly borated water before it reached the core, the possibility of a reactivity anomaly would be present. This event warrants modelling. We did not make a conclusion about the possible significance of this event, because a validated plant model was not available.

5.4.4 Structural Excitations

Because of its size, the pressure vessel needs to be analyzed for its susceptibility to excitation by pressure oscillations at frequencies which might be realized in transient situations. A chugging relief valve is one way to achieve pressure oscillations. A drop of pool level below the steam injection point may be another way.

5.4.5 Communication Between Control Volumes

There are break locations which couple nominally isolated control volumes in ways which may go undetected until an accident occurs. Examples appear in the log sheets in Appendix B. Leakage through the siphon breaker standpipe has this characteristic. This failure mode may not have catastrophic consequences, but it should be modelled. No conclusion about this failure mode is reached here, because a validated plant model was not available.

5.4.6 French Scenario At Start-Up

In the HAZOP of the start-up procedure, attention was paid to the French scenario, which emerged from a French study of the shutdown of a PWR (Diamond, et al., 1992).

The essential elements of the French scenario are (1) the accumulation of unborated water in a stagnant loop during the process of removing boron from the RCS for start-up, and (2) the transport of this slug to the core, leading to a reactivity excursion. In the PIUS design, it is physically possible to inject unborated water without forced primary flow. However, the PIUS design includes a "balcony", whose configuration is difficult to understand from the figures in our possession, but whose purpose is to ensure that unborated water does not accumulate in a stagnant loop. This balcony is a passive structure (located near the hot legs), which accumulates the unborated water in a place from which it will be transported into the primary loops if there is forced flow, but will be transported into the pool if there is no forced flow. This balcony has the specific purpose of ameliorating concerns of the French scenario.

This structure is not the only layer of defense against this type of scenario. However, it is a potentially important layer. Not understanding its structure, we cannot speculate on whether there are passive failures which would compromise its function. However, the balcony deserves a close look, and perhaps should be modelled. Once the structure is understood better, some scoping calculations of the mixing properties could be undertaken.

5.5 References

D. Diamond, et al., "Probability and Consequences of Rapid Boron Dilution in a PWR, NUREG/CR-5819, March 1992.

6.0 SUPPORTING ANALYSES

6.1 Thermal-Fluid Flow Analyses

During the preparations for the FMECA and HAZOP analyses, it was determined that a better understanding of the thermal-fluid flow characteristics of the PIUS reactor was necessary. As a result, a simplified thermal-fluid flow model of the primary loop of the reactor was set up, using the PC-based PIPA-Code. Additionally, once these studies began, further questions arose that necessitated the use of this code. In fact, some of the questions raised could not be answered by such a simplified model as was produced. A brief description of the code and results from the simulations performed are given in Appendix E. The model was applied to the following basic transients and accident scenarios:

Full Scram - One Pump Trip

Loss of Heat Sink

Uncontrolled Boron Dilution

Inadvertent Opening of Safety/Relief Valves (IORV)

In summary, the following observations can be made, based on these simulations:

Full Scram - One Pump Trip:

In accordance with the sequence of events described in PSID, the transient analysis was initiated by tripping one coolant pump, followed by a feed-water trip 18 seconds later. As the hot leg flow decreases, the pressure balance in the pool loop is broken and natural circulation inflow of highly borated pool water, through the lower density lock into the primary loop, begins. Due to the mixing of highly borated pool water with lightly borated primary loop water an oscillatory response is observed, with a period of about 20 seconds. The oscillations are supported by the strongly negative moderator temperature coefficient, calling for power increase with decreasing core coolant temperatures. Many key parameters, like the pump inertia and the efficiency of boron mixing, had to be estimated. Thus, the exact timing and size of the observed oscillations are not certain, but the simulation results are in reasonably good agreement with the vendor's PSID analyses. In all cases analyzed. Hot shutdown was reached within two minutes or less, thus indicating successful operation of the passive shutdown feature.

Loss of Heat Sink

A feed-water trip was simulated, by specifying a rapid linear ramp-down of the secondary side heat removal as boundary condition. The simulated results indicate that the response of the primary system to such an event is an increase in primary loop pressure with a downward movement of the hot/cold interface at the lower density lock. The control system reduces the pump speed and highly borated pool water begins to enter the core through the lower density lock. The reactor was nearly in hot shutdown at the end of 100 seconds.

Uncontrolled Boron Dilution

As a multiple failure scenario, uncontrolled boron dilution was simulated with an additional assumption of scram failure, due to a total blockage of the lower density lock. Boron dilution was modelled as continuous injection of distilled water at the rate of 60 kg/s into the hot leg. As a result of the decreasing boron concentration in the core, the reactor power increased. The power increase was limited by the negative moderator temperature feedback and Doppler feedback. With the postulated scram failure, the pump speed remained essentially constant. The increase in reactor power would be expected to continue until the scram set-point of 120% is reached, leading

to a pump trip, thereby reducing the core flow and limiting the power increase. No serious consequences are expected from this multiple failure event.

Inadvertent Opening of Safety/Relief Valves (IORV)

To simulate a moderate depressurization transient, such as an IORV, flow discharge through a stuck open SRV of the largest flow area was modelled. This is equivalent to a small break LOCA. The decrease in system pressure, due to loss of mass creates a pressure imbalance at the density locks, causing a reduction in pump speed and a consequent natural circulation inflow of highly borated pool water into the primary loop. The reactor power decreased rapidly after the boron ingress begins, and the reactor is in hot shutdown after 125 seconds, with no serious consequences for this event.

The above evaluations of some of the transient characteristics of the PIUS reactor, even though they are preliminary, were of invaluable aid in supporting the probabilistic analyses of this system study. In particular, these evaluations significantly enhanced our understanding of the reactor thermal and fluid flow behavior, as required for productive HAZOP sessions.

6.2 Evaluation of Design Basis External Events

External events typically considered in the design basis of a NPP are fires, earthquakes, high winds, floods, and external missiles. Some preliminary observations of these events are discussed in this section.

The specification of Design Basis external events for PIUS, to be considered as part of the structural loading on buildings, plant systems, and components, depends on the climatic, topographic and geologic conditions which exist at the selected plant site and also on the plant's proximity to potential sources of major industrial or transportation accidents.

The major threats to structural integrity from these external events are typically categorized as follows:

- Seismic Event
- High Wind/Tornado
- Flooding
- External Missiles

The data currently available from ABB Atom focus primarily on the PIUS concept and systems performance. Site-specific considerations, such as external events, have, apparently, not received a great deal of attention.

A generic seismic analysis of the Reactor Building Compound, including a simplified stick model of the Prestressed Concrete Reactor Pressure Vessel, has been conducted by ABB for a zero period acceleration (ZPA) of 0.3g, Reg Guide 1.60 spectra, and a range of site subsurface conditions. A summary of results for a "soft site" is given in the PSID, and technical reports for both a "soft site" and a "hard site" were available for this analysis.

The evaluation of PIUS to maintain structural integrity, when subjected to external events, is primarily qualitative at this time. In the following subsections, each of the four categories, listed above, is discussed from the perspective of unique design features which could affect the severity of consequences.

6.2.1. Site Characteristics

In comparing PIUS to conventional LWRs, the PIUS reactor includes a very large and heavy Prestressed Concrete Reactor Vessel (PCRVR). While this has safety advantages, for siting considerations it means that the weight of the PIUS Prestressed Concrete Reactor Vessel (PCRVR) will result in higher foundation pressures. Ground

water conditions also become more important, because the PIUS concrete vessel foundations are sunk relatively deep.

Ground water conditions and load bearing capacity would appear to be significant considerations in selecting a site for PIUS. With respect to load bearing capacity of the media supporting the PCR, a preliminary calculation based on data contained in the PSID suggests that the foundation pressure will be approximately 19,000 pounds per square foot.

6.2.2 Seismic Event

A very significant, low-probability external event affecting the design of a nuclear power plant is an earthquake. A primary design objective is to protect the public from radioactive release, for all credibly possible scenarios of severe external loading, operational malfunction, and combinations of both. All structures and systems which are absolutely necessary to achieve this design objective, are designated as safety-related.

Safety-related structures and systems must maintain structural integrity at a minimum, and in many cases, must also maintain functionality, when subjected to the maximum postulated earthquake for the plant site, referred to as the Safe Shutdown Earthquake (SSE). These structures and systems are designated as Seismic Category I.

In addition, all non-safety structures and systems must be located and supported, as required, to preclude unacceptable seismic interaction with safety-related structures and systems. Non-safety structures and systems are designated as Seismic Category N.

ABB Atom defines a third seismic category. Quoting from the PSID, Chapter 3.2.1:

"The seismic classification principles for current design LWR plants are outlined in the US NRC Regulatory Guide 1.29. This guide defines the seismic classes, Seismic Class I or N (non-seismic).

"In ABB Atom practice the two classes of Reg. Guide 1.29 is supplemented with a Seismic Class P. Seismic Class P (Passive) corresponds to Seismic Class I regarding the passive function, implying that the structural integrity of the equipment must be maintained, but there is no requirement on its active function. As an example, if a pump is assigned to Seismic Class P this implies that the pump house shall remain intact after an earthquake (no external leakage from the pump), but that there is no requirement on its pumping capability.

"Generally, it can be said that buildings, systems and components which belong to Safety Class 1, 2 or 3 belong to Seismic class I or P.

"The equipment which belongs to Seismic Class N will be designed so that it will not jeopardize the function or integrity of equipment in Seismic Class I or P during or after an earthquake."

Based on additional input from ABB, the designation of seismic Category P is apparently an administrative convenience and does not imply any relaxation of the structural acceptance criteria.

PIUS relies on the passive concept of the ever-present, large volume of borated pool water to shut down the reactor and remove residual heat in the event of total failure of all active systems. There are four basic prerequisites to accomplish shutdown:

- 1) Availability of a sufficient volume of borated pool water.
- 2) Proper functioning of the density locks and the natural circulation pool loop.

- 3) Sufficient concentration of boron in the pool water.
- 4) Ability to relieve overpressure in the Reactor Pressure Vessel caused by pool evaporation (if the passive pool heat removal system also fails)

The conceptual ability to stabilize the core and remove residual heat without operator intervention and activation of special emergency systems significantly reduces, in ABB Atom's evaluation, the number of safety-related structures and systems. Consequently, the number of designated Seismic Class I or P structures and systems is less for PIUS than in present-day LWR plants. Table 3.A in Chapter 3 of the PSID provides ABB Atom's tentative Safety/Seismic classification of all structures and systems. This information is summarized in Section 8 and in Appendix C.

6.2.2.1 Seismic Considerations for PIUS Systems

Major System 100 - Buildings, External Plant Systems

System 121 - Reactor Building

ABB Atom classifies the section of the reactor building "containing safety-related parts" as Safety Class 3/Seismic Class I. The generic seismic analysis of the reactor building compound reported in Chapter 3.7 includes the entire reactor building structure. It is noted that the building responds as a single structural entity. Seismic Class I might be designated for the building as a whole.

System 125 - Control System Building

The central control room of the Control Systems Building is designated Safety Class 3/Seismic Class I in Table 3.A. However, in Volume 3 of the PSID, the description of System 125 does not mention any safety-related function or seismic design requirements. To the contrary, quoting from page 125-2 of Volume 3:

"Major accidents are postulated to destroy the building or force the operators to abandon the building. If the building is completely lost, the conditions in the concrete vessel and the reactor containment can be monitored from the emergency control room (system 518)."

This is an example of classification inconsistencies discussed further in Section 8.

System 151 - Concrete Vessel Monolith

This is the Prestressed Concrete Reactor Vessel (PCRV), which is a unique design feature of PIUS. It is the only designated Safety Class 1 structure or system. See Section 6.2.2.2 for further discussion of the PCRV.

System 152 - Concrete Containment

The Concrete Containment is partly integrated with the Reactor Building and shares a common basement with it. It is designated Safety Class 2/Seismic Class I. The structural delineation between the Reactor Building and the Concrete Containment subsection is virtually non-existent, with one exception. The concrete containment has a steel liner to insure leak tightness, just as in LWRs. Seismically, it must be analyzed as an integral part of the Reactor Building.

System 159 - Concrete Vessel Liner Venting System

This system is excluded from Table 3.A; however, in Volume 3 it is identified as a "safety-grade system". It is a unique design feature of PIUS. Two sets of channels are embedded in the PCRV to monitor and vent any

pool leakage through the pool liner and through the embedded liner. Because of the importance of maintaining the pool level, significant leakage must be avoided. This system might have to be designated Seismic Class I; structural adequacy of the channels and the monitoring instrumentation to resist seismic loading may need to be addressed during design certification.

System 180 - Hangers and Restraints

Per Table 3.A, the Safety/Seismic Classification of hangers and restraints throughout the plant will be consistent with the classification of the systems and components which they support or restrain. No further data are presented by ABB Atom at this time. Methodology and criteria for seismic analysis will be essentially the same as for LWR plant designs.

Major System 200 - Reactor and Auxiliary Equipment

System 211 - Pressure Vessel Upper Parts

This system provides closure of the Reactor Pressure Vessel above the PCR.V. It consists of a steel anchor flange, a steel cylindrical section, and a steel hemispherical closure head. It is designated Safety Class 2/Seismic Class I in Table 3.A. This is a complex structure, unique to PIUS. The anchor flange which mates the steel upper pressure vessel to the PCR.V is an important feature of the seismic loading design.

There are many penetrations in this vessel, including the primary coolant inlet and outlet nozzles for the four loops. In a seismic analysis, the steel upper pressure vessel should be coupled to the primary coolant loops (System 313). The steam generator seismic and deadweight supports are fixed to the outer wall of the PCR.V. Therefore, there are multiple load paths between the steel upper pressure vessel and the PCR.V in a seismic event.

System 213 - Reactor Internals

This system includes the reactor core, the upper and lower density locks, the downcomer primary coolant flow channel from the cold legs to the core entrance, the riser primary coolant flow channel from the core exit to the hot legs, the riser funnel which communicates with the pressurizer steam volume, and the siphon breaker. In Table 3.A, "parts required for reactor shutdown and core cooling" are designated Safety Class 2/Seismic Class I. This is a complex system of components; they are described in some detail in Volume 3 of the PSID.

This entire system, except for the top of the riser funnel, is immersed in the large pool of borated water, which is pressurized to essentially the same pressure as the primary coolant inside the reactor internals. Flow communication between the pool and the primary coolant is possible through the upper and lower density locks. During normal operation, these fluid boundaries are stabilized so that no flow occurs; under upset conditions, flow from the pool to the reactor core is established to ensure that core power is reduced and residual heat is removed. The fundamental safety concept of PIUS relies on the establishment of a natural circulation pool loop through the core, in which the pool water enters through the lower density lock and exits through the upper density lock.

The lower support for the reactor internals is designated System 215; it consists of a lower support skirt, which ties back to the PCR.V liner at the elevation of the reactor core. Also considered, but currently not planned, is a spoke-type seismic restraint at approximately mid-height. ABB Atom's analysis presently indicates that this seismic restraint is not needed.

At the upper end, the reactor internals are supported by System 211, just below the primary coolant nozzles. The source of seismic input motion to the reactor internals is the PCR.V. However, the seismic input at the lower and upper support points may be significantly different. Also, the interaction of the PCR.V and the reactor internals through the pool water might have to be considered.

This system will require very detailed and extensive evaluation. Because the pressure differentials across the shell walls are very small, the seismic loads may play a significant role in the detailed design of reactor internal structures. During design certification, the seismic analysis of the reactor internals might be reviewed in detail.

System 215 - Reactor Assembly Supports and Restraints

In Table 3.A, the "parts required for reactor shutdown and core cooling" are designated Safety Class 2/Seismic Class I. This system is discussed under System 213.

System 216 - Vessel Internal Instrumentation Support Equipment

In Table 3.A, the "parts required for reactor shutdown and core cooling" are designated Safety Class 2/Seismic Class I. The only safety-related reference in Volume 3 of the PSID for this system is that "ruptures of pipes in System 216 outside the reactor pressure vessel shall not be able to drain the reactor pool even over long periods of time". The information provided to date is preliminary and incomplete.

System 231 - Refueling Systems, and System 233 - Handling Equipment for Reactor Internals

In Table 3.A, these systems are designated Safety Class 3/Seismic Class P for parts necessary to prevent fuel elements and internals from being dropped. These systems have the same functions as in present-day LWR plants. The PIUS design, however, requires that the fuel and internals must be raised and lowered through a much greater height.

System 261 - Fuel Assemblies

In Table 3.A, this system is designated Safety Class 2/Seismic Class I. According to ABB Atom in Volume 3 of the PSID, the PIUS fuel assembly "closely resembles a standard present-day PWR fuel assembly except for the shorter length (2.8m)". There is no specific discussion of seismic considerations.

System 291 - Wet Thermal Insulation for the Reactor Primary System

System 293 - Other Wet Thermal Insulation Systems Inside the Reactor Pressure Vessel

These are unique features of PIUS. Wet thermal insulation is not a conventional concept and may require additional development before the design is finalized. Neither Table 3.A nor the description in Volume 3 identifies these systems as safety-related. Based on the stated functions of these systems, they would appear to be safety-related, and which may imply Seismic Class I.

System 291 is mounted to the outer wall of System 213 to minimize heat loss from the primary coolant to the pool. Maintenance of the pool temperature at approximately 50°C (compared to 260-290°C for the primary coolant) is essential for residual heat removal by the pool. The importance of the insulation in maintaining the required pool temperature is unclear. It may be achieved solely by the pool cooling system (323), but this seems unlikely.

System 293 insulates the upper conical part of the PCRV and the anchor flange for the steel upper pressure vessel. It protects the pressure vessel from thermal transients caused by heat up of the upper part of the pool following a main coolant pump trip.

Even if these systems can be demonstrated to be non-safety, they most likely should be designated as Seismic Class P. Loss of structural integrity in a seismic event could potentially defeat the natural circulation pool loop by obstructing the flow holes in the lower support skirt (System 215). Further evaluation of this issue, during design certification, appears to be necessary.

Major System 300 - Reactor Plant Process Systems

System 312 - Reactivity Control System

In Table 3.A, ABB Atom classifies this system as Safety Class 2/Seismic Class I only with respect to the structural integrity of "RCPB (Reactor Coolant Pressure Boundary) part up to and including containment isolation valve(s)". Chapter 312.1.2 states that "System 312 is not a safety-grade system".

System 312 is a collection of subsystems connected through various flow paths to the Primary Coolant Loop. The basic functions are:

- Reactor Startup
- Adjustment/Maintenance of Power Level
- Reactor Normal Shutdown
- Control of Primary Loop Water Chemistry
- Filtered Water Make-up
- Control of hot water/cold water interface level in the upper density lock during normal operation

ABB Atom does not require these functions to achieve safe shutdown following postulated Design Basis Accidents and External Events.

A demonstration of the structural integrity of the primary coolant pressure boundary, by detailed piping and component structural analyses, for load combinations, load classifications (Level A through D), and allowable stress criteria applicable to Seismic Category I piping and components is important.

System 313 - Recirculation System

In Table 3.A, ABB Atom classifies this system as Safety Class 2/Seismic Class I for "Integrity". The system includes the four primary coolant loops: hot legs, steam generators, pumps, and cold legs. The system has no safety-related function, according to ABB Atom. To ensure integrity of the primary coolant pressure boundary, the analysis discussed for System 312 also applies here.

For seismic analysis, the Primary Coolant Loops and the Steel Pressure Vessel Upper Parts (System 211) could be included in the same model. Seismic input motion is through the PCRV at the steel anchor flange and at the steam generator supports.

The structural details for the Steam Generator, particularly the steam generator tubes, is an important area of the PIUS design.

Classification of the Main Coolant Pumps (subsystem 3133) as non-safety/non-seismic for "Function" might need further explanation. A trip of one pump is cited by ABB Atom as the mechanism to activate the pool loop and achieve emergency shutdown. It is important to verify that functional failure of one or more pumps does not jeopardize the shutdown objectives.

System 314 - Pressure Relief System

In Table 3.A, ABB Atom designates this system as Safety Class 2/Seismic Class I for "Integrity". In addition, the "safety valves" are similarly classified, while "other parts" are designated Safety Class 3/Seismic Class P. Chapter 314.1.2 states that "System 314 is a safety-grade system".

This system provides three separate pressure relief functions; in each case steam from the pressurizer steam dome is discharged to the condensation pool. The functions are

- Prevent vessel overpressurization during plant transient and abnormal operating conditions. Four safety valves with separate discharge lines accomplish depressurization.
- Provide forced blow-down of the reactor pressure during certain plant abnormal conditions. Two relief valves with separate discharge lines accomplish blow-down.
- Allow manual actuation of reactor pressure blow-down following a long period of residual heat removal by reactor pool water boil-off. Two emergency blow-down valves with separate discharge loops can be actuated manually from outside containment.

This important system may not be classified highly enough for seismic protection.

For structural integrity, the procedures discussed for System 312 also apply to System 314, i.e., Standard Review Plan guidelines for Seismic Category I piping and components are applicable. Functionality of valves would be demonstrated by seismic testing, using shaker table input motion with a test response spectrum (TRS) that envelopes the required response spectrum (RRS).

System 316 - Condensation System

This system is analogous to the pressure suppression pool in U.S. BWRs. In Table 3.A, this system is designated Safety Class 2/Seismic Class P. Chapter 316.1.2 states that "System 316 is a safety-grade system".

The enclosure of the condensation pool is an integral part of the concrete reactor building, and consequently, must be analyzed as such. In a seismic event, sloshing of the pool water may occur; this additional seismically induced loading on the enclosure walls should be included in future structural evaluations.

There are 600 blow-down pipes, approximately 24" in diameter, embedded in the concrete walls, between the drywell and the wetwell. A seismic evaluation of this design detail could serve to ensure the structural integrity of the containment.

Blow-down of the reactor pressure vessel through System 314 or blow-down of containment may generate significant transient dynamic loads in the condensation pool. Because the condensation pool is structurally integral with the reactor building, these dynamic loads may be transmitted throughout the building. This situation is similar to that of a Mark II containment, where special dynamic analyses are required, in addition to seismic analysis.

System 321 - Reactor Pool

ABB Atom has designated the volume of the pool water in the reactor pressure vessel as System 321; it is specified as Safety Class 2. The significance of the pool of water was discussed earlier.

System 322 - Pool Loop Inactivation Systems

In Table 3.A, this system is designated Safety Class 2/Seismic Class I for "Integrity" of "RCPB part up to and including containment isolation valve(s)". System 322 is a unique design feature of PIUS consisting of the:

- Lower Density Lock System
- Upper Density Lock System
- Gas Lock System

In Chapter 322, ABB Atom makes no reference to safety-grade or any safety-related function.

In Chapter 213, the upper and lower density locks are described as being a part of System 213 (Reactor

Internals), for which "parts required for reactor shutdown and core cooling" are designated Safety Class 2/Seismic Class I in Table 3.A.

Functionality of the Upper and Lower Density Locks is absolutely essential to achieve reactor shutdown and core cooling by passive, natural circulation of the borated pool water through the reactor core. For the locks, functionality can be defined as maintenance of original geometric configuration.

A detailed analysis of the density locks for seismic and operational loads is an important consideration. See also the previous discussion of System 213.

System 323 - Reactor Pool Cooling System

In Table 3.A, this system is designated Safety Class 2/Seismic Class I for "Integrity" of "RCPB part up to and including containment isolation valve(s)". In Chapter 323.1.2, ABB Atom states "System 323 is not a safety-grade system". This system is entirely located inside of containment, and therefore, does not have any isolation valves. It appears as if it should be classified higher seismically.

System 325 - Reactor Pool Water Replacement System

The information provided by ABB Atom in Table 3.A and Chapter 325 for this system is not consistent with respect to Safety Classifications. Clarification would be desirable.

Major System 400 - Steam and Power Conversion Systems

System 400 is very similar to present-day PWRs. There does not appear to be any safety-related requirement associated with this system, other than the integrity of piping inside containment and the establishment of containment isolation.

System 411 - Steam Line System

In Chapter 411.1.2, ABB Atom states that "the parts of System 411 that are located inside containment, up to and including the containment isolation valves, are considered to be safety-grade equipment". Standard analysis methodology for Seismic Category I piping and components applies here.

System 463 - Feedwater System

In Chapter 463.2.7, ABB Atom notes that the feedwater lines to the steam generators (subsystem 4636) penetrate containment and are fitted with containment isolation valves. There is no specific designation of any part of this system as "safety-grade".

Major System 500 - Control Equipment

The Safety/Seismic classification of Control Equipment may have to be reviewed in detail by ABB Atom in preparation for design certification.

At this preliminary stage, only the following three "safety-grade systems" have been identified in Table 3.A and Chapter 500 of the PSID:

- System 516 - Trip and Interlock System
- System 518 - Emergency Control Room
- System 554 - Monitoring of Steam Generator Tube Rupture

These systems are designated Safety Class 3/Seismic Class I and are located in the Reactor Building, which

is Seismic Class I.

In Table 3.A, eight systems are footnoted to say "safety-related parts of these systems" are Safety Class 3/Seismic Class I, but specifics are not provided.

The Emergency Control Room (System 518) is a unique design feature of PIUS. Located in the reactor building, its existence appears to alleviate the necessity to designate the Central Control Room as "safety-grade", in ABB Atom's estimation. Consequently, the Reactor Building is the only Seismic Category I building on the PIUS plant site. Personnel access to the Emergency Control Room, following an accident which disables the Central Control Room, is not clear.

Major System 600 - Electrical Equipment

The information provided by ABB Atom in Chapter 600 is preliminary, with little detail for most systems. In Table 3.A, the following two systems are designated Safety Class 3/Seismic Class I:

System 649 - Main Coolant Pump Power Supply System (only parts for safety-related trip of main coolant pump).

System 673 - Safety-Grade 110V D.C. Auxiliary Power Supply System

ABB Atom might have to perform more detailed system design and review of the Safety/Seismic classification in preparation for design certification.

Major System 700 - Service Systems

The following three systems included in System 700 are designated as Safety Class 3/Seismic Class I in Table 3.A and identified as "safety-grade" in Chapter 700:

System 714 - Cooling Towers for Cooling the Reactor Pool

System 724 - Closed Cooling System for the Reactor Pool

System 751 - Bottle Air System

These systems are unique to PIUS.

The first two provide passive residual heat removal from the reactor pool in the event that active pool cooling systems are lost (e.g., loss of power). These systems are designed to provide indefinite removal of reactor residual heat following an emergency shutdown.

The Cooling Towers (System 714) present a unique condition. They are located on top of containment, inside the reactor building. Seismic input to the cooling towers would be expected to be high because of their elevation. A detailed analysis for integrity and function is important.

The Bottle Air system provides a backup for safety-grade pneumatic valves, if the regular plant power supply system fails. Very little detail is provided by ABB Atom.

Major System 800 - Other Plant Equipment

These systems have no "safety-grade" function, according to ABB Atom. Further review may have to be performed at the design certification stage, to ensure that other system changes do not impose safety-related requirements on System 800.

The PIUS Reactor Pressure Vessel is a unique structure in several respects. It represents the first use of a prestressed concrete reactor vessel (PCRVR) in a commercial LWR. To provide for an inherent passive shutdown capability, the vessel contains a large pool (780,000 gallons) of borated water, in which the reactor is submerged.

The PCRVR is an 88' x 88' x 140' high block of concrete with a central cavity of a maximum diameter of 40', which contains the pool and the reactor (See Figure 2-1). The approximate weight of the vessel, including the pool of water, the reactor, the steel upper closure and the primary coolant loops, is 150,000,000 pounds. The foundation pressure on the base mat, if distributed over the base area of the PCRVR, is $150 \times 10^6 \text{ #}/(88\text{ft})^2 = 19,370 \text{ #}/(\text{ft}^2)$.

ABB responded as follows when asked whether the high foundation pressure would limit the siting of PIUS or require special construction considerations:

"....The concrete vessel, the reactor primary containment and the reactor building have a common base mat (or foundation) which results in a foundation area that is about 5 times the base of the vessel proper. The total mass....amounts to about 3 times that of the vessel. The foundation pressure will, therefore, be 10,000 -12,000 psf...."

"The question on specific limitations on site selection, etc. cannot be answered at this stage of the design. Siting and construction considerations are among the aspects that will be examined as part of the PIUS study that is being negotiated in Italy. It is expected that the study period will be roughly two years."

ABB's response clarifies this matter sufficiently at this preliminary stage. However, several important factors remain to be addressed: ABB's estimate of "10,000 - 12,000 psf" assumes a uniform foundation pressure distribution on the base mat. With one-third of the weight bearing on one-fifth of the base mat area, it is unlikely that the pressure distribution will be uniform unless the base mat is extremely thick. Also, the assumption of complete seismic coupling at the common base mat, which is used in ABB's preliminary seismic analysis, would appear to require a very thick base mat. Therefore, ABB might want to address the relationship between the thickness of the base mat and the foundation pressure distribution, and also between the thickness of the base mat and seismic coupling of the structures, which share the common base mat, for the range of site conditions from "soft" to "hard".

The structural and leaktight integrity of the PCRVR under all credible loading conditions is the most critical structural-related issue for design certification of PIUS. The design internal pressure for the Reactor Vessel is 1300 psi. Prestressing tendons in the axial and circumferential directions are designed to maintain the PCRVR in a state of compressive stress at design pressure. The most significant external threat to its integrity is a seismic event. Generally, pressure vessels are sized to perform their design function, and then the ability to stand up to low-probability events, such as an earthquake, is evaluated for acceptability. However, unlike a steel pressure vessel, which has the ability to absorb a great deal of energy by plastic deformation, a concrete pressure vessel is susceptible to cracking under low-level shear and tensile loads. To improve performance, the concrete is reinforced with steel at expected locations of significant shear and tensile stress. While the tensile stresses due to internal pressure are balanced by the compressive prestress, shear and tensile stresses also originate from seismic loading. In addition, because of the brittle and porous nature of concrete, a steel liner and an embedded steel membrane have been provided (See Fig. 2-1) to prevent loss of pool water through cracks or through absorption.

Because of the special nature of concrete, seismic loading is more critical and plays a greater role in the detailed design of a concrete reactor pressure vessel. Uncertainty about the concrete strength and concrete's low ductility dictate higher factors of safety for a concrete pressure vessel than the comparable factors of safety for a steel pressure vessel. ABB states that the PCRVR is designed in accordance with ASME Code Section III, Division II, Chapter CB-3000, which specifically covers design of concrete pressure vessels. However, the NRC has not endorsed the ASME Code or any other code for design of concrete pressure vessels. This is in contrast to the

NRC's endorsement of ASME Code Section III, Division I for steel pressure vessels.

ABB has designated the Safe Shutdown Earthquake (SSE) as ASME Level C, for which "75% of the ultimate capacity of the vessel may be utilized" and has conducted several preliminary seismic analyses of the Reactor Building Compound, which includes the PCRV and the integral reactor primary containment/reactor building. Two reports, documenting the analysis for a "soft site" and a "hard site," have been received and reviewed.

A review of the specified seismic input for these analyses suggests that NRC Regulatory Guide 1.60 may have been misinterpreted. ABB has generated synthetic time histories to match horizontal and vertical ground spectra taken from Regulatory Guide 1.60. The generic design basis is 0.3g ground acceleration for SSE. It would appear that the vertical spectra used as the basis for the Vertical Synthetic Time History was scaled to 0.2g ZPA (2/3 of the horizontal ZPA). Per Regulatory Guide 1.60, both the horizontal and vertical spectra are to be scaled to the horizontal ZPA. Therefore, the Vertical Synthetic Time History used in ABB's preliminary seismic analyses may need to be scaled by 1.5 to meet ABB's stated design intent.

The seismic analyses submitted by ABB provide insight into the dynamic characteristics of the reactor building compound for assumed "soft site" and "hard site" conditions. ABB has suggested that intermediate site conditions also will be analyzed so that seismic floor spectra can be developed to envelope the entire range of possible site conditions. This approach to seismic qualification of systems and components appears conservative.

Of particular interest is the behavior of the PCRV for different site conditions and the effect of sharing a common basemat with the reactor building. To better understand the dynamic behavior of the PCRV, an independent analysis of frequencies and mode shapes was performed, assuming a rigid base support. A mathematical model of the PCRV was developed for the ANSYS Structural Analysis Computer Program, incorporating the same modeling approach used by ABB for its "stick" representation of the PCRV. To the extent possible, identical modeling techniques were employed. All material property, section stiffness, and mass data input were derived from information contained in the PSID or in the seismic analysis reports. The node and element delineation is the same.

By assuming a rigid base support, the natural vibration characteristics of the PCRV can be isolated from the effects of:

- 1) interaction with the reactor building through the basemat
- 2) flexibility of the soil/rock under the basemat
- 3) flexibility of the PCRV attachment to the basemat

When compared to the results from ABB's "soft site" and "hard site" analyses, it is possible to draw certain conclusions about the expected seismic response of the PCRV.

Tables 6.2-1 and 6.2-2 compare the significant vibration modes in the horizontal and vertical directions for the three analyses.

Table 6.2-1
Horizontal Vibration Modes

<u>Soft Site</u>	<u>RB Compound</u>	<u>Hard Site**</u>	<u>PCRVR Only</u> <u>Rigid Base</u>
0.96 Hz*.371 Hz (rocking)			
2.45 Hz* (rocking/translation)			
4.47 Hz (PCRVR "Beam" Mode)			7.93 Hz (PCRVR "Beam" Mode)

Table 6.2-2
Vertical Vibration Mode

<u>Soft Site</u>	<u>RB Compound</u>	<u>Hard Site**</u>	<u>PCRVR Only</u> <u>Rigid Base</u>
1.78 Hz* (rigid structure/ soilspring)			
9.54 Hz (RB Extensional Mode)			
			24.57 Hz (PCRVR Extensional Mode)

* Modes include rigid body rotation/translation of PCRVR

** Hard site numerical data deleted, since source was proprietary

Examination of the vibration modes points to the following conclusions about the behavior of the PCRVR:

1. For the "soft site," the soil flexibility dominates the system vibration characteristics and, consequently, the flexible modes of the PCRVR are not important. The PCRVR behaves like a rigid body attached to rotational and translational springs. The PCRVR "Beam" Mode at 4.47 Hz has a low participation factor based on the review of the generated horizontal response spectra at different elevations on the PCRVR. The spectral peaks occur at 0.96 Hz and 2.45 Hz. The generated vertical response spectra at different elevations of the PCRVR are identical, with a sharp peak at 1.78 Hz, indicating the strong dominance of rigid body translation in the vertical direction.
2. For the "hard site," the effect of subsurface flexibility is significantly reduced. The primary effect is a contribution to the fundamental vertical vibration mode of the reactor building. The PCRVR participates in this mode as a rigid body, based on review of the generated vertical response spectra at different elevations on the PCRVR. The spectra are almost identical with a sharp peak. In the horizontal direction, the PCRVR "Beam" Mode dominates the response. The steel vessel extension, above the PCRVR, exhibits a local vibration mode, which is more significant than the PCRVR "Beam" Mode. The generated horizontal response spectra at the top of the steel vessel

extension shows a second, and higher, sharp peak at this frequency.

3. The simplified analysis of the PCRV on a rigid base predicts the fundamental "Beam" and extensional vibration modes for the PCRV. This model also includes the steel vessel extension and exhibits a local vibration mode at 20.81 Hz. While the "soft site" and "hard site" analyses conducted by ABB calculate natural frequencies up to 33 Hz, the modes shapes are reported for only the first few significant modes.

The PCRV "Beam" mode is identified in all three analyses. The results for the "soft site" and the "hard site" are reasonably close, indicating only a minor effect of the site conditions in this mode. The significant difference between these predictions and the rigid base prediction (7.93 Hz) is attributed to rotational flexibility at the base of the PCRV. The mode shapes predicted by ABB's analyses show rotation at the base.

The current base support design (See Figure 2-1) would also be expected to provide vertical flexibility at the base of the PCRV. Consequently, the rigid base prediction for the PCRV extensional mode (24.57 Hz) must be considered an upper bound.

4. The current PCRV base support design would appear to be a weak link in maintaining structural integrity of the PCRV when subjected to the SSE. The vertical deadweight plus seismic load is transferred to the basemat through shear loads on the supporting lip. Similarly, the moment reaction at the base due to horizontal seismic loads on the PCRV is transmitted to the basemat by shear loads on the support lip. This structural detail warrants thorough evaluation during design certification.

6.2.3 High Wind/Tornado

In Chapter 3.3 of the PSID, ABB Atom states that these loadings will be described and evaluated "when a site has been selected". While it is true that the detailed specification of these loadings is site-specific, the generic design of building structures, particularly Safety-Related Structures, must be based on a reasonable assumption of these loadings, which envelope most (if not the most severe) conditions expected for a U.S. Site.

The current design of the Reactor Building compound can withstand some level of wind/tornado loading. To support design certification, ABB Atom might have to evaluate the critical structures to withstand these loads.

6.2.4 Water Level/Flooding

In Chapter 3.4 of the PSID, ABB Atom defers discussion until a "site has been selected and the hydrologic conditions there have been evaluated". However, ABB Atom stated in Chapter 2 that ground water conditions are important, because "the PIUS concrete vessel foundations are sunk to a certain depth".

The comments provided under "High Wind/Tornado" also are generally applicable to Water Level/Flooding effects.

6.2.5 Fire Protection

Regulatory Basis

Title 10 of the Code of Federal Regulations part 50.48 requires that all nuclear power plant licensed in the U.S. shall have a fire protection plan satisfying Criterion 3 of Appendix A. This criterion states: "Structures systems and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effects of fires and explosions. Noncombustibles and heat resistant materials shall

be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture on inadvertent operation does not significantly impair the safety capability of these structures, systems and components.

10CFR50 Appendix R sets forth the fire protection features required to satisfy 10CFR50 Appendix A and 50.48. Major headings of Appendix R:

- A. Fire protection program
- B. Fire Hazards analysis
- C. Fire prevention features
- D. Alternative or dedicated shutdown capability

PIUS Design

The PIUS-600 PSID Section 9.6.1 points out that active fire protection systems are not safety-related in ABB designed plants which extensively use passive fire protection by physical separation. All active fire protection systems in PIUS are not safety-related.

The design approach to fire protection in PIUS are:

- The plant is divided into fire zones and fire cells according to Scandinavian practice. Fire cells may share ventilation systems but fire zones do not.
- Smoke extraction systems, as specified in system 868, meet the requirements of Swedish building codes.
- The turbine and cable rooms are provided with water spray systems (system 861).

System 860 describes the fire protection system for PIUS. It is organized by:

- 861 Fire-fighting water systems
- 863 Foam-extinguishing system (no permanent foam systems)
- 866 Gas-extinguishing system
- 867 Mobile fire-fighting equipment
- 868 Smoke extraction system
- 869 Fire detection and trip system

Based on the information provided by ABB, it appears that the fire protection system addresses the areas of the previous paragraph.

Fire Analysis

Besides the broad 10CFR50 Appendix R type of review, the FMECA analysis considered fire as a potential cause of equipment failure (See Appendix A for the items identified in these analyses).

In addition, a global approach was taken to identify sources of combustion. Such identified sources are:

- 1) Hydrogen fire and/or explosion associated with the main generator. High power/voltage electrical generators use hydrogen for insulation and for heat transfer. If such a fire occurred, it would be in the turbine building thus isolated from the reactor which is protected by the reactor building and containment. The turbine

building has a water-spray fire protection which operates automatically or manually. (Fire protection systems and alarms in PIUS are actuated by optical and ion detectors.) The worst-case need for water for fire protection would result in a flow rate of 200 kg/s (3170 gpm)

A fire in the turbine building would result in loss of house power. This is similar to loss of off-site power with loss of power to the primary pumps that automatically results in reactor scram and passive core cooling.

2) Oil fire associated with a power transformer would cause loss of house power if it is a main or auxiliary transformer. The effect is similar to loss of off-site power and results in automatic scram. A fire in a lower-power transformer would result in loss of a power service, but would be covered according to the electrical design.

An oil fire associated with the diesel generator would eliminate emergency power but would not prevent scram. (The emergency power is provided for property protection - not for safety.) The diesel rooms are protected by the gas extinguishing systems.

3) Protection from electrical insulation fires is provided in various ways. Fires in transformers would be extinguished by water spray. An insulation fire in wiring leading to and in the control room would be suppressed by a gas (halon) extinguishing system. If the fire forced evacuation of the control room, the reactor can be shutdown and monitored from the emergency control room.

Note: A fire in the control room has the greatest accident potential of the fire accidents studied because of the psychological effect and the effect of halon-breathing on the operators. There is the possibility of erroneous operator action resulting in a reactivity transient. This should lead to an automatic, passive scram, however under fire conditions, it may be possible that interactive effects may occur that will stress the reactor.

Summary

Fires in the plant could generate one of several initiators (e.g., turbine trip, loss of offsite power, feedwater transient and a computer malfunction). Fires could also generate a number of failures noted in the FMECA or HAZOP analyses. The greatest potential radiological accident identified is associated with improper human actions that might be taken under the stress of a fire, however, detailed fire initiator scenarios have not been developed. Significant fire accidents have not been identified that could potentially prevent PIUS from scrambling and passively cooling the core for up to 7 days.

6.2.6 External Missiles

Chapter 3.5 of the PSID discusses missile protection in general, for both internally and externally generated missiles. External missiles are assumed from three sources:

- Wind and Flood Generated Missiles
- Explosions in the Plant Near Environment
- Crashing Aircraft and Other Vehicles

The containment is designed to protect all equipment which it encloses from external missiles. Chapter 152.3.6 provides general criteria for protection against external events. ABB Atom states, however, that "the external events to be taken into account for the containment design are highly site-specific".

The current containment design should be capable of withstanding some reference level of external missiles, representative of a typical U.S. site. To support design certification, ABB Atom might define the loadings and present sufficient detailed information to facilitate an independent evaluation.

In Chapter 3.5 of the PSID, ABB Atom further states that "safety-related equipment located outside the

containment is designed with redundancy and the redundant subsystems are physically separated in such a way that the missile can only affect one subsystem". This is an appropriate design objective; however, to support design certification, considerably greater detail about its implementation for these subsystems may be required. At a minimum, the affected subsystems and the postulated missiles should be identified; the criteria which insures physical separation should be defined; and a representative analysis should be done.

6.2.7 Summary of External Event Evaluations

This section summarizes some of the main issues identified during the external events evaluations.

The issue of the central control room being non-safety grade and not depending on operator manning for safe shutdown may need thorough evaluation.

A number of issues in the seismic performance of structures, systems, and components may need to be further addressed, as discussed above. Some of these are noted here:

- 1) Various inconsistencies and apparent misclassifications as to Seismic Class I, P and N,
- 2) The seismic classification (I or P) and functional integrity of the wet thermal insulation (Systems 291 and 293),
- 3) The seismic classification of MCPs,
- 4) Loadings on the condensation pool,
- 5) Integral nature of the seismic performance of basemat, PCR/V, primary containment and reactor building,
- 6) Interpretation and application of Regulatory Guide 1.60.

No analyses of high wind or tornado loadings or flooding appear to have been performed to date by ABB Atom.

6.3 Materials Evaluations

There is incomplete information in the PSID to estimate accurately the potential for materials and component degradation and failure during reactor service. However, from a knowledge of generic degradation and failure modes in reactor and non-reactor materials and components, it is possible to identify areas of concern for the PIUS system.

For this study, attention will be focussed on components that are thought to be critical to reactor safety. Evaluations will be made on the basis of known or anticipated operating and startup and shutdown conditions. The most important factors that may lead to component failure are:

- high stresses, including cyclic stresses
- elevated temperature
- neutron embrittlement
- corrosion and stress-assisted corrosion
- hydrogen embrittlement
- mechanical creep

The conditions under which selected PIUS components could be susceptible to degradation and failure are

described below. The consequences of failure will be outlined where possible.

6.3.1 Stainless Steel Pool Liner

Function

The function of the liner is to prevent leakage of borated pool water, to prevent interaction between the pool water and the PCRV, and to transfer loads from most systems in the concrete vessel cavity to the PCRV.

Description

The liner is made of Type 304 stainless steel, 15 mm in thickness. It is constructed using double butt-welded joints. The liner is fixed to the PCRV using anchors that keep the linear strains in the liner to values below the elastic limit. Only under extreme conditions (not defined) will the strain be double the elastic limit. Temperatures in the upper vessel region will be at a maximum of 50 - 60°C. Wet thermal insulation will help keep the liner cool and excess heat in the PCRV will be removed by the PCRV cooling system. Figure 6.3.1, taken from the PSID, shows a schematic of the liner and associated systems.

Possible Failure Modes

Historically, austenitic stainless steels, such as Type 304, have performed well in boric acid environments. However, it is possible that stress-corrosion cracking of the pool liner could occur in extreme circumstances where halide, sulfur, and oxygen levels become elevated.

Data to support this possibility are available from an analysis of Type 304 stainless steel piping failures at the spent fuel pool at Three Mile Island (TMI) (Jones, et al., 1981). The failure occurred after 5 years of service in boric acid (2300 ppm boron), at temperatures from 7-33°C, pH 4.47 to 7.0. The cracks were predominantly intergranular and in the heat-affected zone. Sensitization during welding of the pipe was an important factor in the failure, together with chloride and sulfur contamination. Surface grinding also was determined to be important in causing cracking, because of the production of surface defects that act as crack nuclei.

Tsuge and coworkers (Tsuge, et al., 1984) also showed that Type 304 stainless steel exposed to borated water is susceptible to intergranular stress-corrosion cracking if crevices are present. The crevices act as areas where localized concentration of ions occur, resulting in decreases in the pH. In their work, they cite sensitization as being caused by welding, which creates a heat-affected zone adjacent to the weld.

In the PIUS system, the higher pool water temperature (50-60°C compared to the 7-33°C for the TMI Spent Fuel Pool) should enhance the possibility of stress-corrosion cracking. Although significant chloride, oxygen, and sulfur contamination should not be present in the PIUS pool water during normal operation, they could be present in higher-than-normal concentrations, if there is a failure in the pool water purification system, operator error, or during refueling operations. Oxygen, in particular, should increase during refueling when the reactor head is removed. Any contamination of the pool water could allow the chloride and sulfur species to "hide-out" in crevices and stagnant water regions where they could concentrate. If corrosion products fall to the bottom of the liner, this could also be a region of accelerated attack because of the stagnant water. Welded areas of the liner, which are sensitized and under tensile stress could suffer from cracking if these contaminants remain in place.

Another region of the liner that is highly stressed, and possibly more susceptible to failure from stress-induced effects, is the region where the liner is welded to the upper part of a carbon steel membrane that is embedded in the PCRV to collect water that escapes from the liner. A stress analysis has not been conducted, but cyclic stresses during start-up, shutdown, and refuelling may cause metal fatigue and cracking in the weld region. If the cracks lie at an elevation above the weld line, pool water would escape into the PCRV in locations external to the steel membrane. Aqueous corrosion of the membrane could become important if oxygen is able to enter this

region of the PCRV.

Consequences of Liner Failure

Since the liner, steel membrane, and concrete vessel prevent loss of pool water, loss of leaktightness of any two will not immediately lead to a massive loss of borated water. Small leaks in the liner and the membrane are accommodated by a venting system (System 159) in the PCRV that will monitor and remove water that has escaped (Figure 6.3.1). Nevertheless, leaking water in the presence of oxygen that can diffuse into effected regions will enhance the rate of metal corrosion, which might decrease the life of the PCRV and liner.

Under very extraordinary circumstances, such as extremely severe seismic events, it is not physically impossible for the PCRV to crack. In conjunction with the cracks present in the liner and steel membrane, a major loss in pool water could occur.

6.3.2 Carbon Steel Membrane

Function

The membrane is an embedded liner that will act as an extra vessel to collect any pool water that leaks through failed sections of the PCRV liner (Figure 6.3.1). As mentioned above, however, any cracks in the liner that are at a higher elevation than the weld joint between the liner and membrane will allow pool water to escape collection by the membrane. Such water will be collected and removed by the venting system exterior to the membrane.

Description

The embedded membrane will be made from A442, Grade 60, carbon steel or equivalent. It will be made from sections that are double butt welded. Welds will be radiographed to confirm integrity.

Possible Failure Mode

The most likely failure mode for carbon steel membrane is general corrosion, which would occur if borated pool water escaped from the stainless steel pool liner and migrated through the PCRV to the membrane. Although boric acid corrosion may be a significant problem for carbon steel components in PWRs (Czajkowski, 1990), it is envisioned here that the boric acid will be neutralized by the alkaline concrete to the extent that the normal boric acid corrosion does not occur. Also, the temperature of the membrane is much lower than that at which boric acid corrosion is a problem ($\geq 150^{\circ}\text{C}$). Nevertheless, alkaline water collecting at the base of the membrane will cause the steel to corrode with the rate of attack being controlled to a major extent by the supply of oxygen. Presumably, oxygen will be able to migrate to the membrane via the concrete-membrane interface.

Figure 6-3-1: Inner Venting System of Concrete Vessel (System 1591)

Over very extended periods, it is not inconceivable that the membrane could be breached by corrosion.

Consequences of Membrane Failure

As discussed in Section 6.3.1, serious breaching of the pool liner and membrane could contribute to a severe loss of pool water if the PCRV cracks.

6.3.3 Gas Lock Cover

Function

The gas lock baffle is used during reactor start-up. By filling the baffle with nitrogen, borated pool water is prevented from entering the reactor core via the lower density lock. Once normal operating conditions have been established, the nitrogen is removed through an outlet pipe and replaced by water. By adjusting the primary pump speed, the primary loop pressure is maintained at a value that prevents the pool water from entering the primary system.

Description

The gas lock cover is part of the natural circulation inlet plenum (Figure 6.3.2). It is connected to the lower density lock, thermal insulation, and piping. Parts of the system are bolted together.

Possible Failure Modes

No analyses have been carried out to determine thermal fatigue effects in the inlet plenum components. Therefore, stresses on the system and, especially, the bolts that hold the gas lock cover in place are not known. If cracks develop in the bolts from stress-corrosion or irradiation effects, there might be a potential for the gas lock cover to be dislodged in such a way that it will fall and "cap" the lower density lock.

Consequences of Gas Lock Cover Failure

Core cooling can be provided by the primary pumps, with heat removal to the steam generators, or by passive natural circulation of borated pool water. If the lower density lock were completely blocked due to the above failure, and the primary loop pumps were not available, core damage would have to be expected.

6.3.4 Wet Thermal Insulation

Function

Wet thermal insulation is widely used in the PIUS reactor system. Placed around the reactor vessel, it minimizes heat transfer from the primary system to the cooler pool water. Insulation also is located in regions near the top of the pool liner to minimize heating of the PCRV.

Figure 6-3-2: Reactor Internals (System 213)

Description

The wet thermal insulation consists of "mattresses" made up of stainless steel wire mesh held together by clips. The mattresses are separated by thin stainless steel sheets and protected by stainless steel cover plates. Figure 6.3.3 shows a manufacturer's illustration of wet thermal insulation (ABB, 1991); the dotted regions are the wire mesh mattresses. The manufacturer's trade name for the insulation is "Metalisol."

Possible Failure Modes

The PSID states that the wire in the mattresses may be susceptible to general corrosion, and to some extent, pitting. Because of stresses left in the woven wire mesh, stress-corrosion cracking also is possible. Because water flow in the insulation is low, ABB states that reactor pool chemistry and concentrations of corrosion products will not be affected to any important degree.

However, the low water flow in the insulation may actually be detrimental with respect to the integrity of the wire mesh. Low flow conditions will encourage contaminants, such as halides, and sulfur, to lodge in crevices which will accelerate stress-corrosion and pitting effects. Small pieces of wire could break free and fall to the bottom of the pool liner as corrosion debris. Corrosion processes under the debris could be accelerated because of abnormal water chemistries present in these regions.

Detailed information is not given on the methods for attaching the insulation to the reactor vessel and the pool liner. Fasteners that are highly stressed and exposed to halide, oxygen, or sulfur, especially during refuelling when oxygen will enter the pool system, could become cracked. It is not impossible that during a seismic event, or during reactor start-up or shutdown when stresses may be high, the fasteners could fail completely and the insulation would be dislodged.

Consequences of Wet Insulation Failure

It seems likely that the wire mesh in the insulation will corrode, cracking after long periods of service. If wire fragments break loose and the mesh configuration changes significantly, then water flow through the insulation and heat transfer rates across the insulation will change. However, the magnitude of these effects and the impact on safe reactor operation cannot be judged at this time.

If a significant number of insulation mattresses break free and block some of the coolant holes in the funnel-shaped inlet plenum structure (System 2133), at the least it should slow reactor shutdown. Also, the decreased temperature differential between the pool and the primary water caused by the loss of insulation would decrease the driving force for natural convection through the core.

6.3.5 Reactor Internals (Core Region)

Function

The components in this system (System 213) are designed to support the fuel assemblies and to provide the necessary flow of coolant for normal operation, startup, and shutdown. Loss of dimensional integrity could lead to diversion or restriction in coolant that could cause the fuel to overheat and fail.

Description

See Section 2 of this report and Figure 6.3.2.

Figure 6-3-3: Wet Thermal Insulation (ABB, 1991)

Possible Failure Modes

Neutron irradiation at sufficiently high doses will lead to severe losses in the ductility of austenitic stainless steels. For example, Type 347 stainless steel (which is very similar in composition to the Type 304 stainless steel used in many PIUS core components) showed a decrease in total elongation from 64 to 9.5 percent after irradiating at 400°C to a neutron dose of 4.0×10^{21} n/cm² (Kangilaski, et al, 1970). The test temperature was 315°C. For Type 348 stainless steel, the elongation decreased from 61.4 to 6.6 percent after an irradiation of 3.3×10^{21} n/cm² at 400°C, with the test being conducted at 315°C.

Because many of the reactor components are bolted together, irradiation damage to the bolts needs to be addressed. Welds in the core region often have less starting ductility than base metals so that irradiation embrittlement of these welds will compound the embrittlement problem.

Finally, stress-corrosion cracking of the above-mentioned bolts and crevice corrosion under the heads of the bolts, could be additional failure modes that need to be addressed.

Consequences of Reactor Internals Failure

If neutron doses to stainless steel core components reach doses of about 5×10^{21} n/cm², there will be severe losses in high temperature ductility. High loads from seismic or other events could lead to mechanical disruption of the core, which, in turn, could cause losses in core cooling efficiency and even meltdown.

6.3.6 PCR/V Tendons and Rebars

Function

The tendons in the PCR/V are pre-tensioned steel cables that reinforce the high-strength concrete in the vessel. If cracks develop in the vessel, the tendons and rebars will prevent catastrophic fracture of the brittle concrete.

Description

Figure 6.3.4 is a schematic of the PIUS tendon/rebar configuration (ABB, 1991). The tendons are placed within steel conduits that are embedded within the concrete vessel. There apparently is no detailed information regarding the filler material, if any, that is placed inside the conduit to protect the tendons from corrosion. In many cases, the filler material is either Portland cement grout or organic materials composed of petroleum-based greases or waxes that contain additives to enhance performance (Griess and Naus, 1980).

Possible Failure Modes

Griess and Naus (1980) reviewed failure modes in PCR/V tendons. In general, there were few reported incidences of failure since pre-stressed concrete was first used for nuclear pressure vessels in 1960. However, one case of tendon degradation that they cited concerned cables in the containment vessels of the Marcoule G2 and G3 reactors in France. After five years of operation, the initial tension in the cables had decreased by 30 percent. About 50 percent of the wires in the cables had failed by brittle fracture caused by corrosion. It was concluded that the failure of the tendons was caused by hydrogen embrittlement as a result of excessive humidity in the conduits. To overcome the problem, the conduits were continuously purged with air at 10 percent relative humidity.

2 of 6

Figure 6-3-4: PCRV Prestressing Tendons (ABB, 1991)

Griess and Naus also cited extensive pitting in tendons at the Wylfa Power Station in Britain. The root cause was determined to be salt and moisture from the coastal environment. Corrosion inhibitors in the grease protecting the cables was rendered ineffective at the prevailing high chloride concentrations.

Consequences of Tendon Failure

Failure in the tendons and rebar will result in a loss of mechanical reinforcement of the PCR. Catastrophic fracture of the vessel is high unlikely but the reduction in the safety margin for the structural integrity of the vessel could result in expensive remediation or in decreased service life of the PIUS reactor.

6.3.7 Secondary Side Piping

Function

The secondary piping transports steam from the steam generator to the turbines. The water is condensed and returned to the steam generator.

Description

The design of the PIUS steam and power conversion systems have yet to be fully optimized. However, the piping is made from low carbon steel. In a typical PWR system, the maximum steam temperature is about 290°C.

Possible Failure Modes

Erosion-corrosion is a mechanism of accelerated attack caused by a rapidly flowing corrodent. Attack and metal loss is often increased at locations where the corrodent undergoes a change in direction, such as at an elbow or pipe bend. Numerous examples of erosion-corrosion have been observed in the power industry (see Table 6.3.1 from Wu, 1989). A catastrophic incident occurred at the Surry Unit 2 plant in which an 18-inch feedwater suction line ruptured causing four fatalities.

Based on surveys, carbon and low-alloy steels are much more susceptible to this type of failure than stainless steels. Since the PIUS secondary side pipelines are fabricated from low-alloy steels, failures from erosion-corrosion are conceivable during service. There are no procedures for guaranteeing the elimination of pipe failures from this source. However, current LWR practice includes:

- a) reduce water/steam flow rates to the extent practicable
- b) minimize pipe elbows and bends
- c) add oxygen to the feedwater (PIUS does this)
- d) increase pH of the secondary side water

A second possible failure mode for the secondary piping is thermal fatigue. Weaver (1984) reviewed the failures that occurred in the late 1970s in feedwater lines of PWR steam generators. Thermal fatigue was initially attributed to stress corrosion cracking, but was later concluded to be a result of fatigue arising from thermal stratification of the coolant during hot standby, and subsequent mixing during operation. Because PIUS is likely to use a feedwater heater, this type of failure should be less likely as compared to earlier PWRs.

Table 6.3.1 Plants Experiencing Wall Thinning in the Feedwater Condensate System (Wu, 1989)

Plant/Unit	Type of Reactor	Initial Criticality Date	Degraded Components, Fittings, or Straight Runs
Dresden 2	BWR	January 1970	elbows
Duane Arnold	BWR	March 1974	elbows, reducers, straight runs
Pilgrim 1	BWR	June 1972	elbows
Oyster Creek 1	BWR	May 1969	elbows
River Bend 1	BWR	October 1985	recirculation line
Perry 1	BWR	June 1986	straight runs
Arkansas 1	PWR	August 1974	elbows, drain pump discharge piping
Arkansas 2	PWR	December 1978	undefined
Calvert Cliffs 1	PWR	October 1974	elbows, reducers, straight runs
Calvert Cliffs 2	PWR	November 1976	elbows, reducers, straight runs
Callaway 1	PWR	October 1984	recirculation line elbows
Diablo Canyon 1	PWR	April 1984	elbows, straight runs
Diablo Canyon 2	PWR	March 1978	elbows
Ft. Calhoun 1	PWR	August 1973	elbows, straight runs
Haddam Neck	PWR	July 1967	recirculation line
Harris 1	PWR	October 1986	recirculation line
Millstone 2	PWR	October 1975	elbows, heater vent piping
North Anna 1	PWR	April 1978	elbows, straight runs
North Anna 2	PWR	June 1980	elbows, straight runs
Robinson 2	PWR	September 1970	recirculation lines
San Onofre 1	PWR	June 1967	reducers, heater drain piping
San Onofre 2	PWR	July 1982	heater drain piping
San Onofre 3	PWR	August 1983	heater drain piping
Salem 1	PWR	December 1976	recirculation line
Salem 2	PWR	August 1980	recirculation line
Surry 1	PWR	July 1972	fittings
Surry 2	PWR	March 1973	fittings
Sequoyah 1	PWR	July 1980	elbows, straight run
Sequoyah 2	PWR	November 1981	elbows
Trojan	PWR	December 1975	elbows, reducers, straight runs
Turkey Point 3	PWR	October 1972	feedwater pump suction line fittings
Fort St. Vrain	HTGR*	January 1974	straight run in emergency feedwater line
Rancho Seco 1	PWR	September 1974	straight runs downstream of main feedwater (MFW) loop isolation valve or MFW pump min/flow valve

* high-temperature gas reactor

Consequences of Secondary Piping Failure

Cracking of the secondary piping will lead to major outages and cost for repairs. However, most serious consequences would be associated with erosion-corrosion, because it may cause catastrophic pipe rupture, injure workers or cause fatalities.

6.3.8 Primary Side Carbon- and Low-Alloy Steel Components

Function

The PIUS primary system delivers hot coolant to the steam generator to produce steam in the secondary side for power generation. Water is returned to the core via a system of piping, pumps, and valves.

Description

See Section 2 of this report.

Possible Failure Modes

A recent survey shows that several PWR carbon- and low-alloy steel components are susceptible to severe boric acid corrosion caused by primary coolant leaks (see Table 6.3.2 from Czajkowski, 1990). Leakages occur through faulty seals and gaskets, and the concentrations of boric acid are greatly increased by evaporation of the water. Massive wastage of the steel is possible in a short time. For example, in 1980, the Calverts Cliffs plants suffered severe boric acid attack of the primary coolant pump studs, as well as the manway studs of the steam generator.

Other components that may be affected include pumps, valves, threaded fasteners, and piping. Not only is the leaking component susceptible to attack, but adjacent components also may be corroded, if the coolant trickles to their location. In some cases, insulation may hide the movement of water, and the discovery of the problem may occur after significant damage and failure of the component.

Consequences of Component Failure

Boric acid attack of components will result in costly repair or replacement of the component and, possibly, plant shutdown. In an extreme case, e.g., degradation of the manway studs, there could be loss of the cover and depressurization.

6.3.9 Summary of Material Evaluations

Due to the unique nature of the PIUS design, careful evaluation of potential material degradations and/or failures of all safety related components appear important. This section briefly described the most important possible degradation/failure modes in PIUS components that have a direct bearing on safety. Because of the preliminary nature of the information currently available, in particular on anticipated service conditions, the analyses are tentative at this time. The most important possible materials-related failures, that were identified here, are summarized in Table 6.3.3.

Table 6.3.2 Summary of Corrosion of Carbon Steel by Boric Acid (Czajkowski, 1990)				
Date	Plant (NRC Notice, etc.)	Corroded Component	Type of Corrosion	Source of Leakage
1977	St. Lucie	Steam Generator Manway Closure Studs	Corrosion Wastage	Manway Gasket Leakage
1978	St. Lucie	Pressurizer Manway Closure Studs	Corrosion Wastage	Manway Gasket Leakage
1979	Zion, Unit 1	CVCS Valve	Erosion/Corrosion	Unknown
*5/80	Fort Calhoun I&E Notice 80-27	Reactor Coolant Pump Studs	Wastage: 2.4 in. Max. in 14 months	Pump Case/Cover Interface
6/80	Calvert Cliffs Units 1 & 2	Reactor Coolant Pump Studs steam Generator Manway Studs	Corrosion Wastage	Leaky Gasket
1/81	Calvert Cliffs Units 1 & 2	Suction Piping to Reactor Coolant Pumps	1/8 in. Corrosion Wastage	Leak Pump Seal Pressure Lines/ Bleed Off Lines
7/81	Kewaunee	Instrument Isolation Valve's Bonnet	---	Leaking Valve's Diaphragm
2/81	Oconee Units 1 & 2	Reactor Coolant Pump Studs	Corrosion Wastage	Leaky Gasket
1981	Arkansas 1	Steam Generator Manway Closure Studs	Corrosion Wastage	Leaking Closure Gasket
1981	D.C. Cook 2	Check Bonnet Bolts	Corrosion Wastage	Valve Body to Bonnet Gasket Leakage
*3/82	Maine Yankee I&E Notice 82-06	Steam Generator's Manway Closure Studs	Corrosion Wastage and Cracks	Small Primary Coolant Leak
*10/86	ANO-1 I&E Notice 86-108	High Pressure Injection Nozzle attached to Cold Leg	1/2 in. Corrosion Wastage	Leaky Isolation Valve
*3/87	Turkey Point Unit 4 I&E Notice 86-108 Supplement 1	Reactor Vessel Head, Closing Studs and CRDM	Corrosion Wastage	Leaky Lower Instru- ment Tube Seal (500 lbs. of boric acid crystals formed)
*8/87	San Onofre Unit 2 I&E Notice 86-108 Supplement 2	Bolts on Valve Packing Follower Plate in Shutdown Cooling System	Corrosion Wastage	Leaky Valve Packing

Table 6.3.2 Summary of Corrosion of Carbon Steel by Boric Acid (Czajkowski, 1990)				
*8/87	Salem Unit 2 I&E Notice 86-108 Supplement 2	Reactor Vessel Head	Corrosion Wastage 360 mils	Seal Weld on Instrument Penetration (crystals)
5/88	Millstone Unit 2	RPV Nozzels 1A & 1B	Corrosion .060 mils	RPV "O" Rings
12/89	ANO - 1	CRD Housing Nut Rings	Corrosion Wastage	Leaking Gaskets CRDM Flanges

* IE Information Notices were issued

Table 6.3.3 Summary of Possible Materials Related Safety Issues		
COMPONENT	MATERIAL	POSSIBLE SAFETY ISSUE
Pool Liner	Stainless Steel	Borated water and oxygen contamination could cause stress-corrosion cracking
C-Steel Membrane	Carbon Steel	Could be attacked by concentrated boric acid if pool liner fails. Failure of liner and membrane will cause leakage of pool water
Gas Lock Cover	Stainless Steel	Fatigue, corrosion, and irradiation could cause failure of bolts holding cover in place. Cover could fall and cap lower density lock and restrict flow
Wet Thermal Insulation	Stainless Steel Mesh	Insulation could be fragmented by stress corrosion and be transferred in such a way that coolant flow could be disrupted
Reactor Core Components	Stainless Steel	Core support structure will become embrittled by irradiation. Seismic loads could cause disruption of core leading to fuel damage
PCR V Tendons and Rebars	Carbon Steel	Corrosion-induced brittle fracture could reduce strength of PCR V and lower safety margin
Secondary Side Piping	Carbon Steel	Erosion-corrosion or fatigue could cause catastrophic rupture
Primary Side Steel Components	Carbon and Low-Alloy Steel	Leaking pool water could give severe corrosion from boric acid. In extreme cases, such as failure of steam generator manway studs, depressurization could occur

6.3.10 References

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7.0 EVENT TREE ANALYSIS

This section describes how the FMECA, HAZOP and deterministic analyses are combined to identify accident initiators and accident mitigators and to depict how the PIUS systems work together in an accident. Two approaches are taken. One is the conventional PRA approach of estimating initiator frequencies, and system failure probabilities to determine the probability of an accident sequence with estimated consequences. The frequencies probabilities are categorized I to VI, and the consequence classes are CC-1 to 3. (See the Table of Definitions for details on these items.) The NRC Event Category methodology is introduced in this Section and classifies event sequences according to the initiator frequency and combinations of failures. The accident sequences are discussed for each initiating event. The sequences are represented in Event Tables that are used to construct the event trees. The sequences are also shown as event sequence diagrams. Each event tree classifies the sequence according to both the consequence category (CC) and the event category (EC.)

7.1 PIUS Intermediate Table

The FMECA (Section 4), the HAZOP (Section 5) and the supporting deterministic evaluations (Section 6) each provide insights into the safe operation of PIUS. Consolidation of these independent, overlapping studies for the purpose of event tree (ET) construction is done with PIUS Intermediate Table (PIT). The PIT condenses information, that is selected for safety relevance, from the three sources. Of necessity, consolidation only includes information on the selected items that is needed for constructing the integrated ET analysis of PIUS. Reference to more detailed information is provided by identifying the page number and location on the page of the item as shown in the PIT (Table 7-1). This is followed by a brief description of the component/subsystem, whether concurrent failures have been identified for the concern, and the failure mode/cause code table. Next we identify the support system needed for the item's operation, the criticality and estimated failure frequency class. The two extreme right columns of the PIT provide keys to integrating the analysis i.e. classification of an item as contributing either to an accident initiator or to the failure of an accident mitigator. Since each row of the PIT Table condenses more than one item from Appendix A, one should consult the appropriate line of the corresponding Appendix A table for more specific details.

The PIT was initially prepared from the FMECA (appendix A). This was followed by a review of the HAZOP and supporting analyses. However the extensiveness of the PIT was such that it included these other items with regard to event tree construction; they were not explicitly added to the PIT. These analyses provide distinct contributions to the study in areas beyond the event tree analyses and are presented in detail in Sections 5 and 6. Those, that contribute to the major findings of the study, are presented in the appropriate tables of Section 9, Results.

The following table explains the acronyms used in the PIT.

TABLE OF SYMBOLS FOR USE WITH TABLE 7-1

b - binding	fe - failure of electronics	re - radiation embrittlement
ba - bad adjustment	fs - fuel swelling	s - seismic
c - corrosion	h - heat	se - starting equipment
cf - control failure	he - human error	sf - sensor failure
cr - crud buildup	i - icing	sfwr - software failure
crh - control room habitability	lp - loose part	shrt - short circuit
d - dinting e - electrolysis	losp - loss of off-site power	st - sticking
er - erosion	m - moisture	t - tornado
el - electronics failure	me - material defect	tr - signal transmission failure
emi - electromagnetic interference	md - maintenance damage	v - vibration
f - fatigue	ms - missile	vf - valve failure
fc - failure of controls	pm - poor maintenance	w - wiring failure
fd - flood	pt - pump trip	
	pwr - power	

Table 7-1 PIUS Interim Table (PIT)

Page	Item	Component	Concur. Failure	Failure mode/cause	Support system	Criti- cality	Frequency	Initiator class	Mitigator/ Aggravator
A1-8	1	Head retention tendon		fracture/c,me,e,f,re		L	V	LPPB	
A1-8	2	Head nuts for ten- dons		fracture/c,me,e,f,re		L	V	LPPB	
A1-8	3	Pressure dome		fracture/me,f,re		L	V	LPPB	
A1-8	4	Steam lines to pressure dome		pipe break/me,f,re		L	V	SPPB	
A1-9	5	Pressure relief		fail to open/c,b,ba		L	III		Pressure relief valve
A1-9	6	Pressure relief		fail to close/c,b,pm		L	IV	SPPB	
A1-9	7	Water drain to reactivity control		pipe break/me,f,re		L	IV	SPPB	
A1-10	13	Offgas release valve for non- condensable gases		valve fail to oper- ate/he,vf,sf	dc power	L	V	LPPB	
A2-20	1	Lower density lock		flow blockage/lp, me,s,c,me,f		M	IV		Activate pool loop
A2-20	2	Gas lock		flow blockage/lp, me,s,f,c,vf,he		M	IV		Activate pool loop
A2-20	3	Core support		drop core/c,me,e,f, re,pm,he,me,s		L	V	Over- power	

Table 7-1 PIUS Interim Table (PIT)

Page	Item	Component	Concur. Failure	Failure mode/cause	Support system	Criti- cality	Frequency	Initiator class	Mitigator/ Aggravator
A.2-22	7	Upper density lock		flow blockage/lp, me,s,c,me,f		M	IV		Activate pool loop
A.2-23	2	Core cooling channel		flow blockage/c, cr,fs		M	IV	Under- cooling	
A.3-8	1	Reciprocating pump		break envelope/c, me,f,b,ba, pm,lp,me		M	IV	SPPB	
A.3-8	2	Piping		break/c,me,f		M	V	SPPB	
A.3-9	3	Check valve	failure of both pumps or stop pumps	failure to close/st,lp, me		M	V		Reactivity control
A.3-10	2	Piping		break/c,me,f		M	V	SPPB	
A.3-10	2	Piping		break/c,me,f		M	V	SPPB	
A.3-11	3	Check valve	failure of both pumps or stop pumps	failure to close/st,lp, me		M	V		Reactivity control
A.2-22	7	Upper density lock		flow blockage/lp, me,s,c,me,f		M	IV		Activate pool loop
A.2-23	2	Core cooling channel		flow blockage/c, cr,fs		M	IV	Under- cooling	

Table 7-1 PIUS Interim Table (PIT)

Page	Item	Component	Concur. Failure	Failure mode/cause	Support system	Criti- cality	Frequency	Initiator class	Mitigator/ Aggravator
A.3-8	1	Reciprocating pump		break envelope/c, me,f,b,ba, pm,lp,me		M	IV	SPPB	
A.3-13	3	Check valve	failure of both pumps or stop pumps	failure to close/s,lp, me					Reactivity control
A.3-14	1	Piping		break/c,me,f		M	V	SPPB	
A.3-13	2	Check valves		failure to close/s,lp, me		L	V	SPPB	
A.4-7	1	Piping		break/c,me,f		M	V	LPPB	
A.4-8	2a	SG tube rupture		rupture/c,f, d,er		M	IV	SPPB	
A.4-9	3a	Main coolant pumps (4)		Pump lockup/lp,f,b		M	V	LPPB	
A.5-6	1a	Safety valves		Fail to open/b,c,me,ba,cf		M	IV		Pressure relief
A.5-6	1b	Safety valves		Fail to close/b,c, me,ba,cf		L	II		Pressure relief
A.5-6	2a	Relief valves	Safety valves	Fail to open/b,c,me,ba,cf		M	IV		Pressure relief

Table 7-1 PIUS Interim Table (PIT)									
Page	Item	Component	Concur. Failure	Failure mode/cause	Support system	Criticality	Frequency	Initiator class	Mitigator/Aggravator
A.5-7	2b	Relief valves	Safety valves	Fail to close/b,c, me,ba,cf		L	II		Pressure relief
A.6-7	1	Containment	Release inside of containment	Rupture of walls, s, t,m		M	V		Containment
A.7-29	1	Manual scram trip Sys. 5161	Scram requirement	Fail to scram/ he, fe,fc	active scram	H	IV		Manual scram
A.7-29-30	2-8	Auto scram trip, Sys 5161	Scram requirement	Fail to scram/ he, fe,fc		M	IV		Active scram
A.7-31	9	Manual scram trip Sys. 5162	Scram requirement	Fail to scram/ he, fe,fc	active scram	H	IV		Manual scram
A.7-31-32	10-16	Auto scram trip, Sys 5162	Scram requirement	Fail to scram/ he, fe,fc		M	IV		Active scram
A.7-33-36	17-26	Containment isolation trip, Sys 5163	Radiation release inside of containment	Fail to isolate/sf, cf		M	III		Containment isolation

Table 7-1 PIUS Interim Table (PIT)

Page	Item	Component	Concur. Failure	Failure mode/cause	Support system	Criticality	Frequency	Initiator class	Mitigator/Aggravator
A.7-37	27-28	Startup interlock, Sys 5165	Scram requirement	Fail to scram/ he, fe,fc	active scram	H	III		Active scram on startup
A.7-38	29-32	VDU	Multiple VDU failures	Fail to scram & control reactor/ he, emi,pwr	electric power	H	IV		Active, manual scram, reactivity control
A.7-39	33	Scram button		Fail to scram/ shrt, pt		H	V		
A.7-39	34-36	Display	Multiple display failures	Fail to indicate/ w, shrt, el		H	V		Manual scram, reactivity control
A.7-40	37-38	Central processor	Both units	Fail to operate/he, el,fire,s, fd,cr		M	III	Computer	
A.7-41	41-43	Status display		Unknown status/ el,crh		H	IV	Computer	
A.7-41	44	Operator control of computer		Fail to control/ el, crh		H	III	Computer	
A.7-41,42	45-49	Computer control		Fail to operate/el, crh, fire, fd, sfwr		H	IV	Computer	
A.7-45	54-55	Boron concentration	boron control upset	Fail to operate/ el		M	II		Reactivity control
A.7-50	70-74	Reactivity control	reactivity control upset	Failure of control/ el	power	M	II		Reactivity control

Table 7-1 PIUS Interim Table (PIT)

Page	Item	Component	Concur. Failure	Failure mode/cause	Support system	Criti- cality	Frequency	Initiator class	Mitigator/ Aggravator
A.7-52	78	Outlet temperature		Loss of scram signal		M	III		Active scram
A.7-53,54	83-85	Mode selector		Mode selector control failure/ el, sfwr		M	III		Reactivity control
A.7-55,56	88-93	Pipe break detection		Failure to detect pipe break/ el, tr, sfwr,		M	IV		Active scram
A.7-57-60	94-101	Containment isolation		Failure to isolate/ el, sf, sfwr, tr		M	IV		Containment
A.8-5	1-2	Mode identification		Computer failure/ el, swfr, he		H	II	Computer	
A.8-5,6	3,5-10	Turbine protection		Turbine trip/ el, he, losp		M	II	Over-cooling transient	
A.8-5	4	Missile protection		Turbine overspeed/ el		M	IV	LPPB	
A.9-9-13	1-16	Electrical power		Loss of off-site power /i,t,s,h,m,v		M	II	Loss of off-site power LOSP	
A.9-14	17-18	Local power		Local power/h,m,v		L	III	Loss of local power	

Table 7-1 PIUS Interim Table (PIT)

Page	Item	Component	Concur. Failure	Failure mode/cause	Support system	Criticality	Frequency	Initiator class	Mitigator/Aggravator
A.9-15	20	Elect. noise suppression		Failure to suppress noise/ w, shrt,h,m,v		H	III	Computer	
A.9-17	25-26	Emergency power	Main power	Diesel failure /se,-fuel,pm, el	Diesel fuel, starting motors, DC power	L	II		Emergency power
A.9-18-20	27-38	DC power	Main power	DC power/ el,fir-e,ex,		M	III		DC and uninterruptable power
A.10-28,33	6,7, 19-22	Passive reactor pool coolers	Active pool cooling	Pipe failure or plugging/ ms,cr, f, v		L	IV		Passive pool cooling
A.10-29-32	8-18	Active reactor pool cooler	passive pool cooler	Pipe failure, plugging, pump failure/ ms, cr,f,v,el,pwr		L	IV		Active pool cooling

Table 7-1 PIUS Interim Table (PIT)

Page	Item	Component	Concur. Failure	Failure mode/cause	Support system	Criticality	Frequency	Initiator class	Mitigator/Aggravator
AppB PZR-2		Riser Funnel	Break	Boron Dilution in Top Layer		H	I		
AppB PZR-3		Relief Valve/ Pressure Oscillations		Chugging		M	IV		
AppB PZR-5		Pool Level Control System		Fails Low		M	IV		
AppB PZR-5		Steam Injection Pipe Break	Vessel Head Fails Consequentially	Break		M	IV		
AppB PZR-6		Siphon Breaker	Break above Venturi			M	IV		
AppB PZR-6		Siphon Breaker		Break above cold leg		M	IV		
AppB PZR-7		Dome Leak				M	IV		
AppB PZR-7		Bolted Flange		Fails/Thermal Cycling		M	V		
AppB PZR-8		Steel Pressure Vessel		Fails/Thermal Cycling		M	V		
AppB PZR-10		Level Oscillation in Siphon Breaker		Cold Leg/Hot Leg Interface Failure		M	III		

Table 7-1 PIUS Interim Table (PIT)									
Page	Item	Component	Concur. Failure	Failure mode/cause	Support system	Criti- cality	Frequency	Initiator class	Mitigator/ Aggravator
AppB STP-1		Pressure		Overpressurize		M	IV		
AppB STP-1		Pressure		Leave P Control in Manual		M	III		
AppB STP-4		Chemistry		Add Too Little Hydrazine		M	III		
AppB STP-15		Mode (Switch Setting)		Incompatible with Conditions		M	III		
AppB STP-6		Boron Concentration		Deborate Before MC Pumps Started		H	III		
AppB STP-7		Reactivity		Start Reactor in Natural Circulation		H	IV		
AppB PHS-3		Boiler		Break		H	IV		
AppB CC-3		Lower Density Lock		Failure Due To Thermal Cycling		M	IV		
AppB CC-6		Inlet Plenum		Flow Tilt		H	III		
AppB CC-7		Gadolinium		Asymmetry		H	IV		
AppB UDC-4		Insulation		Loss		M	IV		

Legend: Appendix B page numbers refer to the HAZOP Logsheets, numbered in the upper right-hand corners of each grouping. PZR: Pressurizer group; PHS: Pressurizer Heating System; CC: Core Channel; UDC: Upper (and lower) Downcomer.

7.2 Assembly of the Initiators and Mitigators

Tables 7-2A and B summarize the results of collecting accident initiators and mitigators from the PIT for use in constructing event trees. Table 7-2A lists the initiators i.e. those items that can initiate an accident, and the associated frequency categories for each event using the categories presented in Section 3.2.4 and the Definitions Table. These initiator frequencies were selected to be somewhat conservative, when varying data existed (e.g., more recent EPRI studies would place LOSP in Frequency Category II). Table 7-2B lists the actions or systems designed to mitigate an accident sequence. Associated with each mitigator is the probability that it fails to mitigate the accident sequence. This probability is a dimensionless number i.e. the probability that the mitigator fails to function on demand. For standby equipment, this is the probability to fail start and/or to fail to run for the mission time. For operating equipment, it is the probability of failure to operate for the mission time¹. To avoid confusion between categories for initiators which are per year, and mitigator failure probabilities which are per demand, a single category is used which is numerically the same for both. A few mitigators that appear in only one ET are not listed here.

Table 7-2A PIUS Initiators	
Accident Initiator	Initiator Frequency (per yr)
Large Primary Pipe Break (LPPB)	V
Small Primary Pipe Break (SPPB)	IV
Loss of Off-Site Power (LOSP)	I
Turbine Trip (TT)	I
Steamline Break (SB)	IV
Feedwater Transients (FT)	II
Computer Malfunction (CM)	III
Severe Seismic (SS)	V
Fuel Element Drop (FED)	III
Flow Blockage during Refueling (FB)	IV

The initiators in Table 7-2A are self-explanatory and often considered in PRA analysis of nuclear power reactors. One exception is computer malfunction (CM) which is included because PIUS lacks hard-wired control and because of the FMECA-identified concern for electromagnetic interference (EMI) associated with the thyristor control of the primary cooling pumps.

The first five mitigators of Table 7-2B are explained by noting that PIUS has five ways to achieve subcriticality (see also Appendix F). These vary considerably in the speed at which they do so:

¹ This probability is generally calculated in the linear approximation of the exponential failure modeled as $\lambda \tau$ where λ is the failure rate and τ is the mission time.

1) Active (or automatic) scram which means a scram initiated by the main computer when it detects monitored parameters outside their allowable ranges. When this occurs, the computer causes power to be removed from one primary pump. This results in a flow imbalance and pool loop activation.

2) Manual scram is a scram initiated by an operator pressing a scram button. The following actions are the same as for the active scram.

3) Passive scram is a scram that results from any flow imbalance in the reactor causing pool loop activation.

4) Hot shutdown is effected by a feedwater reduction causing the reactor coolant temperature to rise, shutting down the reactor by the strong negative reactivity temperature coefficient. This must be accompanied by boration of the primary coolant so that the reactor remains shutdown after cooling.

5) The reactor can be shutdown by boration of the primary coolant, using the reactivity control system.

The next four mitigators in Table 7-2B are concerned with decay heat removal. When the pool loop is activated, the heat in the reactor core is transferred to the pool thus requiring heat removal. Active pool cooling refers to the pumping of pool water through heat exchangers to remove the heat by the plant heat sink. Passive pool cooling refers to the passive heat exchangers submerged in the reactor pool that transfer the heat to the atmosphere via passive heat exchangers on the roof.

Table 7-2B Major PIUS Mitigators	
Accident Mitigators	Mitigator Failure Probability (per demand)
Active Scram (AS)	III
Manual Scram (MS)	II
Passive Scram (PS)	IV
Feedwater Reduction	III
Reactivity Control	III
Active Pool Cooling	III
Passive Pool Cooling	IV
Closed Cooling Water	III
Service Water	III
Steamline Isolation	II
Containment	III
Emergency Water Makeup	II
Steam Generator Tube Rupture	II
Critical Heat Flux Exceeded	II
Remove Blockage	II

The closed cooling water system is the isolated heat exchanger system. The service water refers to the heat removal system that exposes the coolant to the atmosphere by cooling towers.

Containment is the last line of defense to protect the public from an accident. It is the enclosure designed to confine any accidental release of radioactivity from gaining access to the atmosphere.

Emergency water makeup is the provision for externally adding water to the reactor pool to prevent core uncover.

The remaining mitigators are associated with specific accident types, discussed in Sections 7.3 through 7.11. Before discussing details of the accident scenarios, the NRC Event Category methodology is presented.

7.3 The NRC Event Category Method²

One concept underlying NRCEC Method is that the best known information on reactor risk is the frequency of initiators³. Initiator frequencies, together with subsequent type and number of failures, are used to classify sequences in the NRCEC. For this study, failures following the initiator, that degrade the protection to a degree that an accident may affect the public, are not estimated in terms of numerical frequency but in terms of probability class. The classes are based on the general observation that failure probabilities of active equipment or component are bounded by a certain range which also encompasses human error. Entire systems fail at a lower frequency because of the use of redundancy of active components and passive components. Common mode failures fall into this class as do double active redundancies. Figure 7-1 presents a general diagram of the NRC Event Category methodology. For purposes of more clearly describing the NRCEC methodology, the failure classes (FCs) utilized are identified by number in Table 7-3.

Table 7-3 NRCEC Mitigator Failure Classes for Use in the NRCEC Methodology	
FC-0	No failures
FC-1	One active component failure or one operator error
FC-2	One system failure ⁴ or one common mode failure or two component failures or operator errors
FC-3	More failures than FC-2

The four NRCEC event categories are defined in Table 7-4. For an event to be EC-1, it must have an initiating event frequency (IEF) such that, $IEF \geq 1E-2$ per year and no failures i.e. the only occurrence in the accident is the occurrence of the initiating event or one additional failure of Class FC-1. For an event to be EC-2,

² The NRC Event Category method was developed at the NRC for the purposes of this study and does not represent any regulatory position.

³ It should be noted that frequency of observed initiators such as transients is well-known, actuarial data. However, there has never been a guillotine break of a large primary pipe; hence, the frequency of unobserved initiators is about the same as unobserved failures of passive systems or highly redundant active systems.

⁴Passive system failure is included in this category.

the IEF must be between $1E-2$ and $1E-4$ per year (thus the frequency is in an intermediate range), and it also includes an additional failure of class FC-1. Event class EC-3 is more complex. It involves two initiating frequency classes: a) if the IEF $\geq 1E-4$ per year and failure class, FC-2 occurs, the event is EC-3; b) it is also EC-3 if IEF is between $1E-4$ and $1E-7$ per year and no additional failures or failure class FC-1 occurs. Sequences that go beyond those shown in Table 7-4, either by having more failures or lower initiating event frequencies, are identified as NRC Event Category 4 (EC-4). This can also be seen in Figure 7-1. In this figure any additional failures beyond those shown still result in a classification of EC-4.

Figure 7-1 NRC Event Category Methodology

Using Table 7-4, together with the frequency categories of the initiating events one may determine the possible corresponding event categories. These are given in Table 7-5. For example, the low frequency (Category V) of the LPPB means that the event tree end states must be either EC-3 or EC-4; it is not possible to have an EC-1 or EC-2 end state for an LPPB. It should be noted that this table lists the possible event categories of all sequences beginning with a specific initiator.

Table 7-4 NRC Event Categories				
	Frequency of Initiator P_{IE}			
Failure Class	$P_{IE} > 10^{-2}$	$10^{-2} > P_{IE} > 10^{-4}$	$10^{-4} > P_{IE} > 10^{-7}$	$P_{IE} < 10^{-7}$
FC-0	EC-1	EC-2	EC-3	EC-4
FC-1				
FC-2	EC-3			
FC-3				

Table 7-5 NRCEC Classification of Sequences by Initiators		
Accident Initiator	Frequency Category	Event Categories
Large Primary Pipe Break (LPPB)	V	EC-3
Small Primary Pipe Break (SPPB)	IV	EC-2 or EC-3
Loss of Off-Site Power (LOSP)	I	EC-1 or EC-3
Turbine Trip (TT)	I	EC-1 or EC-3
Steamline Break (SB)	IV	EC-2 or EC-3
Feedwater Transients (FT)	II	EC-1 or EC-3
Computer Malfunction (CM)	III	EC-1 or EC-3
Severe Seismic (SS)	V	EC-3
Fuel Element Drop (FED)	III	EC-1 or EC-3
Flow Blockage during Refueling (FB)	IV	EC-2 or EC-3

Note: Any initiator may result in a sequence that is EC-4, if the sequence has a sufficient number of failures.

This classification scheme results in a considerable simplification. Using it, we can see that for PIUS, a large primary pipe break (LPPB) and severe seismic (SS) can only be EC-3 or EC-4; loss of off-site power (LOSP), turbine trip (TT), feedwater transients (FT), and computer malfunction (CM) can only be EC-1, EC-3 or EC-4; small primary pipe break (SPPB), steam line break (SB), and flow blockage during refueling (FB) can only be EC-2, EC-3 or EC-4.

The NRC Event Category methodology is factored into the event trees using Tables 7-3 through 7-5.

The NRCEC methodology is useful in that it provides a way of correlating the information from the event

tree analysis to current standard deterministic analysis methods. For example, within the NRC EC-2 is contained the class of Design Basis Accidents (DBAs.) There will of course be many other sequences or events that are EC-2 that are not DBAs. In fact, later in this section one will see that there are many events in EC-2 (and even EC-3) that fall into consequence class CC-1. Further, the purpose of this project is not to specify which events are the DBAs, but rather to provide a listing of potential candidates that can be evaluated in more detail later. With these precautions in mind, we list here the deterministic analogues to the NRC ECs:

EC-1	Abnormal Operational Occurences (AOOs)
EC-2	Design Basis Accidents (DBAs)
EC-3	Severe Accidents (SAs)
EC-4	Very Low Probability Residual Risk Events (RRs).

Method of Presentation of Results

Figure 7-2 Event Sequence Diagram Symbols

The scenarios, to be evaluated below, will first be presented, using Event Sequence Diagrams (ESDs), followed by Event Tables, and Event Trees (ETs.) The meanings of the the event sequence diagram symbols are given in Figure 7-2. These symbols are used to diagram the event sequences (see, for instance, Figure 7-3, below) that correspond to the sequence descriptions. An alternative method is the construction of an Event Sequence Table (see, for instance, Table 7-6, below) in which each sequence of events following an initiator is indicated by 1 for a successful mitigator action, 0 for mitigation failure and - if the mitigator is not relevant. The event table serves several important functions: it concisely shows the sequence logic, its compactness assists in logical completeness, and it provides the input to code BETA (Fullwood and Shier, 1979), the event tree code. Preparing the code input with a word

processor along with writing the sequence descriptions is a convenient and efficient way to perform the analysis. A comparison of event tables/trees with the event sequence diagrams shows greater detail in the event tables. It is impractical to indicate all of the combinations of sequences in an ESD figure, that can easily be indicated in the Event Table. Thus one can find additional paths or sequences in the ETs that are not represented in the ESDs. The Event Trees show the results from the BETA analysis and present the endstates according to both CC and EC categories.

The Concept of Frequency Categories (FC) and Consequence Categories (CC) was discussed in detail in Section 3.

7.4 Large Primary Pipe Break (LPPB) Event Tree

Rupture of the Prestressed Concrete Reactor Vessel

The rupture of the reactor vessel is typically considered as part of the large break analysis for current LWRs. Hence it is addressed here. The PCRV has a wall thickness of 7.4 m (24.3 ft), and a very complex reinforcing steel network (see Figure 6-4) with a stainless steel liner and an imbedded membrane. There is a system that detects water leakage, indicative of liner leakage. Because of the thickness of the wall, its strength and

monitored leakage, rupture of the PCRV appears to be highly unlikely (i.e., Frequency Category VI). Therefore, it is not addressed here any further.

Large Primary Pipe Break (LPPB)

This accident is discussed in Section 15.9.6 of the PIUS-600 PSID. In the following paragraphs the accident scenario is described, and shown as an event sequence diagram.

The accident initiator is a break in any of four, hot-leg 665 mm (26.2 in) diameter or the four, cold-leg, 635 mm (25 in) diameter pipes that connect with the pressure dome closing the PCRV. These pipes circulate primary coolant through the steam generators. A rupture of any of these pipes, while the reactor is at power, will result in primary water flashing to steam. (A rupture in another operating mode will have less severe consequences because of the lower temperature and/or pressure.) Such a break would raise the pressure and temperature inside containment. A break is detected by:

- 1) Low primary pressure,
- 2) Low pressurizer pressure,
- 3) Low pressurizer level,
- 4) Change in primary flow,
- 5) Low coolant level in the PCRV,
- 6) Pipe break detection instrumentation, and
- 7) Pressurization of the containment.

Out-of-normal indications from any of these monitors will initiate an active scram (automatic scram: AS) via the computer control system to remove power from the scram main coolant pump.

If an automatic scram is not initiated, a reactor operator may initiate a scram by pressing the scram button (manual scram: MS) which, through the computer control system, also removes power from the scram MC pump.

Scram can also be the result of a thermally-induced-flow imbalance in the primary system (passive scram: PS). Prior to pipe break, this flow is balanced to exclude pool loop flow. When the pipe break occurs, the flow becomes imbalanced and the pool loop is activated in the same way as it is activated by removing power from the scram pump.

A concern is that depressurization due to the pipe break may result in fuel damage, due to the critical heat flux being reached in the coolant.

The PSID states that ANSALDO Spa (Genova) analyzed this accident using the RELAP 5 code for a double-ended guillotine break in the cold leg and concluded that fuel temperatures do not exceed those of normal operation. The PSID concludes that no fuel damage will occur.

Nevertheless, PIUS operates at a pressure that is only about 1.4 MPa above the saturation pressure. It does not have a gas-filled accumulators for safety injection such as a conventional PWR ECCS, hence it is expected to depressurize rapidly. For these reasons, the possibility of fuel damage on depressurization is included in the event tree (critical heat flux: CHF). Success in the critical heat flux means that there is no damage to the fuel or cladding as a result of the initial depressurization. Exceeding the critical heat flux is a function of the rate of depressurization, and hence, the break size, assuming that a double-ended guillotine break will lead to fuel damage, inhibiting core cooling.

As described above, the pool loop activates to scram the reactor and transfer the decay heat to the pool water. The heat from the reactor pool is removed by the pool cooling system, using diesel and electric pumps through heat exchangers to cooling towers (active pool cooling: APC).

Should active cooling of the pool fail, the heat is removed from the pool by two passive heat exchangers and transferred to the passive water-air heat exchangers located on the reactor roof (passive pool cooling: PPC).

Failure of all pool cooling will result in water being lost from the primary system as follows: this water will condense on the inner walls of the containment, but the PIUS design does not have a means for returning this condensate from the condensate pool back to the primary circuit. Water may be replenished externally with a fire hose connection outside of containment. This operation requires depressuring the reactor, which occurs automatically for a large pipe break, and manually opening the valve that connects the primary with the fire hose connection. A water truck may be connected to transfer water to the pool (emergency water make-up: EWM).

If active or passive pool cooling fails, the containment (CNM) will fail from overpressure regardless of whether the core remains covered because of steam pressure buildup in the containment with no release mechanism. Opening the EWM to replenish the pool water will result in injection of water to the primary or containment atmosphere because of the coupling through the postulated LPPB. No decrease in containment pressure occurs through the EWM systems.

Failure of these mitigators will result in a radiological release by containment leakage if the core remains covered; a larger release if the core is uncovered. These possibilities are depicted for the large pipe break event in the event sequence diagram of Figure 7-3 and in Table 7-6, the event table. A "1" in the event table indicates success, while a "0" indicates failure.

Figure 7-3 Large Primary Pipe Break Event Sequence Diagram

Table 7-6 Large Pipe Break Event Table

No.	AS	MS	PS	CHF	APC	PPC	EWM	CNM	CC	EC
1	1	-	-	1	1	-	-	1	1	3
2	0	1	-	1	1	-	-	1	1	3
3	0	0	1	1	1	-	-	1	1	4
4	0	0	1	0	1	-	-	1	2	4
5	0	0	1	0	0	1	-	1	2	4
6	0	0	1	0	0	0	1	0	2	4
7	0	0	1	0	0	0	0	0	3	4
8	0	0	0	-	-	-	-	-	3	4

The initiator for this event tree is a large pipe break event, which is assigned a frequency category IV¹. The assignment of frequency classes to initiators and mitigators was done for the purpose of calculating sequence frequencies. Since this is a preliminary study of a qualitative or scoping nature, the frequency categories were assigned somewhat conservatively. This should be taken into consideration when evaluating the results.

In sequence 1 of Table 7-6, the large break is followed by an automatic scram (AS - failure probability III) hence the manual scram (MS - probability II) and the passive scram (PS - probability IV, see Appendix F) are not needed (or not applicable). The critical heat flux is not exceeded (CHF - probability II) and the active pool cooling (APC - probability III) works therefore, the passive pool cooling (PPC - probability IV) and emergency water make-up (EWM - probability II) are not needed. The containment (CNM - probability III) thus also succeeds. Since everything succeeds, the large break is successfully mitigated, no core damage occurs and a consequence category 1 (CC-1) is assigned.

Sequence 2 is similar to Sequence 1 except the active scram has failed and the manual scram succeeds. This sequence is written in shorthand as LPPB*AS. Otherwise it is like sequence 1, hence assigned CC-1. In Sequence 3 (LPPB*AS*MS), both active and manual scrams fail but scram is achieved by the passive scram system, giving CC-1. Sequence 4 (LPPB*AS*MS*CHF) is like sequence 3 except CHF has failed, releasing fission products into the coolant. Because of this, radioactivity will be released inside of containment and leak to the atmosphere. There also will be some worker exposure to radiation; hence, this sequence is assigned CC-2. Sequence 5 (LPPB*AS*MS*CHF*APC) is like sequence 4 except the active pool cooling failed but the passive pool cooling succeeds. In sequence 6 (LPPB*AS*MS*CHF*APC*PPC*CNM), PS and EWM succeed but everything else fails. Sequence 5 is assigned CC-2; sequence 6 is also assigned CC-2 even though the containment fails, since the core remains covered. The radioactivity release from sequence 6 is probably greater than from sequence 5. In sequence 7 (LPPB*AS*MS*CHF*APC*PPC*EWM*CNM), only the passive scram works so the core overheats and is uncovered; hence, it is assigned CC-3. In sequence 8 (LPPB*AS*MS*PS), there is complete scram failure which overwhelms the pool cooling systems, the containment fails as a result, and thus the sequence is assigned CC-3. The releases in Sequence 8 will be more severe than in Sequence 7.

Table 7-6 is converted to an ASCII file and analyzed using the BETA code to produce the event tree in Figure 7-4. The frequencies for these sequences are presented in Table 7-7, which also includes summary frequencies for the consequence categories. These results show that the probability of impacting

¹Frequency/probability classes are presented in Section 3.2.4. These are expressed as a range of values. The event tree calculations require point values therefore the geometric mean of the range is used: I = 3/year or demand; II = 0.3/year or demand; III = 0.03/year or demand; IV = 1E-3/year or demand, V = 3E-6/year or demand, and VI = 1E-7/year or demand.

Figure 7-4 Event Tree for the Large Primary Pipe Break

Table 7-7 BETA Results for PIUS Large Pipe Break Scenarios

<u>Sequence Number</u>	<u>Sequence Probability Category</u>	<u>Consequence Category</u>
1	IV	1
2	IV	1
3	V	1
4	V	2
5	VI	2
6	VI	2
7	VI	3
8	VI	3
Total CC-1	IV	-
Total CC-2	V	-
Total CC-3	VI	-

the public (CC-3) by a large primary pipe break is remote, even with the conservative frequency estimates that were used.

It is worthwhile to reflect why the LPPB risk is so low. The reason appears to be because of the several ways of scrambling the PIUS reactor (active, manual and passive). The active scram is only credited with 0.03

failures/demand and the manual scram with 0.3; both of these depend on stopping the scram pump - a FMECA finding. But these are backed-up by the passive scram with a failure probability of $1\text{E-}3$; thus, the total failure to scram probability is conservatively estimated to be $9\text{E-}6$, which, when combined with the LPPB frequency, gives $9\text{E-}9/\text{yr}$ before other mitigators are considered. At the least, to get a radioactive release the cladding must be damaged. In this model, exceeding the critical heat flux was given a probability of 0.3 which seems very high, considering the fact that the initial RELAP calculations indicate no overheating on a pipe break. If only the cladding is damaged, the release inside of containment will only be that of the gases that were contained within the damaged fuel. (Radioactive iodine will be largely suppressed because of the underwater release.) This release is not very large, but will escape the containment; hence, the classification of CC-2. To produce a large release, it is necessary to overheat the core which requires boiling of the cooling water resulting from failure of both the active and passive pool cooling. As long as the containment is intact there probably would be considerable condensation on the containment wall and the evaporated water would return to the pool. However, no path has been identified that results in transfer of condensate to the primary system (this issue needs further review). Hence, it must be assumed that the containment will fail if both active and passive cooling fails.

7.5 Small Primary Pipe Break Event Tree

Definition of a Small Primary Pipe Break

Section 15.9.5 of the PSID discusses leaks and ruptures of the primary piping inside the PCR/V; Section 15.9.6 discusses leaks and ruptures of the primary piping outside of the PCR/V but, inside the containment. Section 7.3 analyzed a large primary pipe break, namely, in the four large pipe loops connecting the steam generators with system 211. The PSID is not sufficiently detailed to show all of the smaller diameter primary pipes, therefore for purposes of this analysis, a small primary pipe break (SPPB) is defined as a break in a primary pipe of sufficiently small diameter that a double-ended guillotine break will not result in exceeding the critical heat flux during reactor depressurization. This definition implies that a small pipe break is sufficiently large that the pressurizer cannot maintain operating pressure. Thus, a SPPB is a primary pipe break small enough that the critical heat flux in any part of the core is not exceeded but sufficiently large that reactor operating pressure cannot be maintained.

Small Primary Pipe Break (SPPB)

A rupture of a small pipes, while the reactor is at power, will result in primary water flashing to steam. (A rupture in another operating mode will have less severe consequences because of the lower temperature and/or pressure.) Such a release will raise the pressure and temperature inside the containment. A break is detected by the following indications:

- 1) Low primary pressure,
- 2) Low pressurizer pressure,
- 3) Low pressurizer level,
- 4) Change in primary flow,
- 4) Low coolant level in the PCR/V,
- 5) Pipe break detection instrumentation, and
- 6) Pressurization of the containment.

Out-of-normal indications from any of these monitors will initiate an active scram (automatic scram: AS) via the computer control system to remove power from the scram MC pump.

If an automatic scram is not initiated, a reactor operator may initiate a scram by pressing the scram button (manual scram: MS) which, through the computer control system, removes power from the scram pump.

Scram can also be the result of primary - thermally-induced-flow imbalance (passive scram: PS). Before pipe break, this flow must be balanced to exclude pool loop flow. When the pipe breaks, the flow is imbalanced and the pool loop is activated in the same way as it is activated by removing power from the scram pump.

In the event of a LPPB, a concern is that depressurization due to the pipe break may result in fuel damage, due to the critical heat flux being reached in the coolant. By definition, this is not possible for the SPPB.

As described above, the pool loop activates to scram the reactor and transfer the decay heat to the pool water. The heat from the reactor pool is removed by the pool cooling system, using diesel power electric pumps through heat exchangers to cooling towers (active pool cooling: APC).

Should active cooling of the pool fail, the heat is removed from the pool by two passive heat exchangers and transferred to the passive water-air heat exchangers located on the reactor roof (passive pool cooling: PPC). Failure of these mitigators will result in a radiological exposure of the public if the containment fails (CNM), allowing a radionuclide release to the atmosphere.

If the containment fails, primary water will evaporate. Water lost from the pool may be replenished externally to the containment by a fire hose connection. This operation requires depressuring the reactor, which occurs automatically for a pipe break, and opening the valve that connects the primary system with the fire hose connection. A water truck may be connected and the water transferred to the pool (emergency water make-up: EWM).

These possibilities are depicted in Figure 7-5, the event sequence diagram, and in Table 7-8, the small pipe break event table. Event 1 assumes a small pipe break initiator (frequency class IV). (It should be emphasized that this and the following probabilities are believed to be very conservative that is they are somewhat higher frequencies and probabilities than expected from experience with conventional nuclear power plants.)

Table 7-8 Small Pipe Break Events

No.	AS	MS	PS	APC	PPC	EWM	CNM	CC	EC
1	1	-	-	1	-	-	1	1	2
2	0	1	-	1	-	-	1	1	2
3	0	0	1	1	-	-	1	1	3
4	0	0	1	0	1	-	1	1	4
5	0	0	1	0	0	1	0	2	4
6	0	0	1	0	0	0	0	3	4
7	0	0	0	-	-	-	-	3	4

In sequence 1, the small break is followed by an automatic scram (AS - failure probability III) hence the manual scram (MS - probability II) and the passive scram (PS - probability IV) are not needed (i.e., not applicable). The active pool cooling (APC - probability III) works, therefore, passive pool cooling (PPC - probability IV) is not needed. The emergency water make-up (EWM - probability II) is not needed because the pool cooling works. The containment (CNM - probability III) succeeds because it is not challenged. Since everything succeeds (after the SPPB initiator), this is assigned event category CC-1.

Sequence 2 is similar to 1 except the active scram has failed but the manual scram succeeds; hence, it is assigned CC-1. In sequence 3, both active and manual scrams fail but scram is achieved by the passive scram system, giving CC-1 because everything else works. Sequence 4 is like sequence 3, except the active pool cooling fails but the passive pool cooling succeeds giving CC-1. In sequence 5, passive scram succeeds, but both active and passive pool cooling fails. EWM succeeds in keeping the core covered but the containment fails. As described for the LPPB, since the core is covered at all times and the fuel has not been damaged, the only radiation that can be released is in the form of gases and vapors in the primary system that escape from the compromised containment, hence CC-2. In sequence 6, only the passive scram works; the core uncovers and radioactive material is released from the failed containment. This is thus assigned CC-3. In sequence 7, there is complete scram failure which overwhelms the pool cooling systems and EWM, and hence fails the containment; thus, it is assigned CC-3.

Figure 7-5 Event Sequence Diagram for the Small Pipe Break Scenarios

Table 7-8 is converted to an ASCII file to use with the BETA code to produce the event tree in Figure 7-6. The frequency categories for analyzing the event tree are presented in Tables 7-2. Table 7-9 presents the results of the BETA analysis of the event tree.

Table 7-9 BETA Results for PIUS Small Pipe Break Scenarios

Sequence Number	Sequence Probability Category	Consequence Category
1	IV	1
2	V	1
3	V	1
4	VI	1
5	VI	2
6	VI	3
7	VI	3
Total CC-1	IV	1
Total CC-2	VI	2
Total CC-3	VI	3

It is worthwhile to reflect on why the SPPB risk is so low. The reason appears to be the various ways of scrambling the PIUS reactor, as was found for the LPPB. The active or automatic scram is only credited with 0.03/demand and the manual scram with 0.3; both of these depend on stopping the scram pump -a FMECA finding. But the automatic and manual scrams are backed-up by the passive scram with a failure probability of 1E-3 thus the total failure to scram probability is conservatively estimated to be 9E-6 which when combined with the SPPB frequency gives 9E-9/yr. This values does not allow for the other mitigators.

Figure 7-6 Small Primary Pipe Break Event Tree

7.6 Loss of Off-Site Power

Loss of off-site power (LOSP) results from the loss of load for PIUS when it is operating at power. It can also include the loss of offsite power being supplied to PIUS while refueling, in hot shutdown, or cold shutdown. An LOSP at power necessitates plant shutdown in any case. When considering the effects of LOSP for different operating modes, the risk from LOSP while operating at full power bounds the risk of the other modes hence this is the situation that is considered here.

When LOSP occurs, switchgear sense the loss of load and trip the plant from the transmission line. The generator excitation is removed to prevent damage to the generator and the turbine steam valves close. In principle, the plant could continue to operate and supply its house loads. Unfortunately, the output cannot be quickly reduced to such a low power. In this discussion, it is assumed that LOSP immediately results in loss of normal AC power in the plant. This being the case, neither active scram (AS) nor manual scram (MS) is required because pump power is removed with the LOSP. The LOSP, thus, initiates the passive scram (PS - IV) provided by the pool loop that shuts down the core with borated water and transfers the residual heat to the pool. (Because the PCRV is under normal pressure, there should be no possibility of exceeding the critical heat flux, thus CHF is not included in the LOSP event tree.) The pool heat is removed by the diesel-powered active pool cooling (APC - note: the failure probability of the APC under diesel power only is less than under both normal AC and diesel however, under diesel only, the failure rate is still within category III). Should APC fail, the heat is removed by passive pool cooling. If both APC and PPC fail, the decay heat will cause pressure to build up in the PCRV and trip the pressure relief valves to discharge the steam into the wet well. Eventually, the wet well will heat sufficiently that the steam will

no longer condense. The steam may pressurize and fail the containment (CNM - III). The core dryout can be prevented by emergency makeup water (EMW - II) otherwise the core may overheat.

Scenarios are shown as event sequence diagrams (Figure 7-7) and by the event table (Table 7-10). The Boolean logic and the event categories are copied as an ASCII file for input to the BETA code. The resulting event tree is shown in Figure 7-8, the results of which are presented in Table 7-11

Figure 7-7 Loss of Off-Site Power Event Sequences

Table 7-10 Loss of Offsite Power Event Table

No.	PS	APC	PPC	EWM	CNM	CC	EC
1	1	1	-	-	1	1	1
2	1	0	1	-	1	1	3
3	1	0	0	1	0	2	4
4	1	0	0	0	0	3	4
5	0	-	-	-	-	3	3

Table 7-11 BETA Results for PIUS Loss of Off-Site Power

Sequence Number	Sequence Probability Category	Consequence Category
1	I	1
2	III	1
3	V	2
4	VI	3
5	IV	3
Total CC-1	I	1
Total CC-2	V	2
Total CC-3	IV	3

Figure 7-8 Loss of Off-Site Power Event Tree

7.7 Turbine Trip

A turbine trip can be actuated during normal operation or during startup by actuation of the scram system, the steam bypass system, or the feedwater system. In addition, a turbine trip can be caused by various types spurious activations. The normal sequence of events following a turbine trip includes a rapid pressurization of the secondary system, opening of the steam bypass valves, scram, and a reduction in feedwater flow. Failure of the bypass valves or the scram (both active and passive) could activate the steam generator relief and safety valves. An additional failure of these valves could result in overpressurization and possible failure of the steam generators. Sequences that include a successful scram require the active or passive cooling systems for long term decay heat removal. The following paragraphs give additional information on the events in the sequences. The event sequences begin with the turbine trip (TTRIP - I) which is followed by steam bypass valves opening (BYPOP - III) to discharge steam to the turbine condenser and to relieve the potential overpressure in the steam generator secondary side. The BYPOP should be sufficient to prevent the actuation of the steam generator relief and safety valves. The turbine trip event results in high primary coolant temperature and high pressurizer pressure giving an automatic scram (AC - III), by tripping the designated scram pump, which in turn activates the pool loop and shuts down the reactor. If the active scram fails, the reactor operator can initiate a manual scram (MS - II) to trip the scram pump and shutdown the plant. Should these scram mechanisms fail, passive scram (PS - IV) would be actuated by the turbine trip causing an increase in primary system pressure and temperature that upsets the flow balance and actuates the borated pool loop to scram the reactor. Failure of the steam bypass system could cause overpressure on the

steam generator secondary side. The BYPOP should be sufficient to prevent the actuation of the steam generator relief and safety valves. The turbine trip event results in high primary coolant temperature and high pressurizer pressure giving an automatic scram (AC - III), by tripping the designated scram pump, which in turn activates the pool loop and shuts down the reactor. If the active scram fails, the reactor operator can initiate a manual scram (MS - II) to trip the scram pump and shutdown the plant. Should these scram mechanisms fail, passive scram (PS - IV) would be actuated by the turbine trip causing an increase in primary system pressure and temperature that upsets the flow balance and actuates the borated pool loop to scram the reactor. Failure of the steam bypass system could cause overpressure on the steam generator secondary side that will actuate the steam generator relief and safety valves (RVSV - III). Failure of these valves could result in steam generator tube rupture with discharge of primary coolant into the containment with possible overpressurization of the containment. After a successful scram and shutdown of the reactor, the active pool cooling (APC - III) continues to transfer heat from the pool, through heat exchangers to the cooling towers. If the active pool cooling fails, heat is removed from the pool by passive pool cooling (PPC - IV). If both the active and passive pool coolers fail, the pool will eventually boil to trip the primary pressure relief and pressurize and possibly fail the containment (CNM) in a similar way to failure of the containment from RVSV failure. If the containment fails, water will boil out of the PSRV and eventually uncover the core. This may be prevented by emergency water makeup.

Table 7-12 presents the event table for the turbine trip sequences but does not include failure of the containment and emergency water makeup for conservatism. In Sequences 1 through 9, the bypass system is assumed to be operable. These nine sequences were developed assuming failures in the active, passive, and manual scram. In addition, failures of the active and passive cooling systems were considered. The most severe sequence (22) assumes the failure of all scram functions, overwhelming the pool heat removal system and resulting in a Category 3 event. The event sequence diagram is shown in Figure 7-9.

Table 7-12 Turbine Trip Events

No.	BYPOP	AS	MS	PS	RVSV	APC	PPC	CC	EC
1	1	1	-	-	-	1	-	1	1
2	1	1	-	-	-	0	1	1	3
3	1	1	-	-	-	0	0	2	4
4	1	0	1	-	-	1	-	1	1
5	1	0	1	-	-	0	1	1	4
6	1	0	1	-	-	0	0	2	4
7	1	0	0	1	-	1	-	1	3
8	1	0	0	1	-	0	1	1	4
9	1	0	0	1	-	0	0	2	4
10	0	1	-	-	1	1	-	1	1
11	0	1	-	-	1	0	1	1	3
12	0	1	-	-	1	0	0	2	4
13	0	0	1	-	1	1	-	1	3
14	0	0	1	-	1	0	1	1	4
15	0	0	1	-	1	0	0	2	4
16	0	0	0	1	1	1	-	1	4
17	0	0	0	1	1	0	1	1	4
18	0	0	0	1	1	0	0	2	4
19	0	1	-	-	0	-	-	2	3
20	0	0	1	-	0	-	-	2	4
21	0	0	0	1	0	-	-	2	4
22	-	0	0	0	-	-	-	3	4

Figure 7-9 Turbine Trip Event Sequence Diagram

Sequences 10 through 21 are similar to Sequences 1 through 9 except that the bypass system is assumed to fail. Sequences 19 through 21 assume failure of the steam generator relief and safety valves concurrent with failures of the bypass system, resulting in an overpressurization and failure of the steam generator. An estimate of the consequences of each sequence is shown in the last column of the table.

The event table provided in Table 7-12 was analyzed with the BETA code to produce the event tree shown in Figure 7-10. In addition, Table 7-13 gives a summary of the frequency categories for each of the consequence categories.

Figure 7-10 Turbine Trip Event Tree

Table 7-13 BETA Results for PIUS Turbine Trip Event Tree

Sequence Number	Sequence Probability Category	Consequence Category
1	I	1
2	II	1
3	V	2
4	III	1
5	IV	1
6	VI	2
7	IV	1
8	V	1
9	VI	2
10	III	1
11	IV	1
12	VI	2
13	IV	1
14	V	1
15	VI	2
16	V	1
17	V	1
18	VI	2
19	IV	2
20	V	2
21	V	2
22	VI	3
Total CC-1	I	1
Total CC-2	IV	2
Total CC-3	VI	3

7.8 Steamline Break (SB)

An analysis of a steamline break was performed in Section 15.6 of the PIUS-600 PSID. This analysis assumed that a break was located between the steam generator and the inside containment-isolation valve of the steamline. It is not apparent that this is a bounding location for a steam system malfunction. ABB Atoms should provide justification for the selection of the bounding HELBs to be analyzed. Because of the safety significance of high energy line breaks (HELBs), steamline breaks both inside and outside the containment are investigated herein.

7.8.1 Steamline Break Inside Containment (SBIC)

The postulated SBIC was analyzed in Section 15.6.6 of the PIUS-600 PSID for the location of the break sated above. From of the information available, it appears that a more bounding accident would result from a break located between the inside isolation valve and the containment penetration. Figure 7-11 presents the event sequence diagram for this event.

Figure 7-11 Steamline Break Inside of Containment Event Sequence Diagram

A break in any one of the four 26-inch main steamlines at this location while the reactor is at 100% power will result in steam flowing directly into the containment air space. Flow from the postulated break would be detected by any one of the following:

- 1) High containment pressure,
- 2) High containment temperature, or
- 3) Low steamline pressure.

Any of these indications would initiate an automatic scram (AS probability class III) by the computer control system and remove power from the designated primary scram pump. The reactor operator could initiate a manual scram (MS - II) if the AS was not initiated. The manual scram also uses the computer control system to remove power from the designated pump. If these mechanisms fail, a passive scram (PS - IV) is actuated by an

imbalance between the thermally induced flow through the core and the flow from the primary pumps. A steamline break would greatly increase the rate of heat removal from the primary loop via the steam generators, thus lowering the core inlet temperature and causing a reactivity increase that will upset the flow balance and activate the pool loop (passive scram). In turn, this will change the rate of heat removal from the reactor via the steam generator to alter the thermally induced flow through the core, thus disrupting the flow balance, causing an influx of borated pool water through the lower density lock to scram the reactor. Residual decay heat from the core following the reactor scram is removed by the active pool cooling (APC - III). If these fail, the heat is removed by the passive pool cooling (PPC - IV). If both fail, the pressure in the intact primary system will rise, tripping the pressure relief valves to discharge steam into the wet well. Water lost from the primary system may be replaced by the emergency water makeup (EWM - II), supplied by a water truck. If these measures fail, the core will uncover in about 7 days with resulting fuel damage.

The integrity of the containment was not mentioned in the above analysis. The steamline break challenges the containment in two ways. First, a steamline break rapidly releases a large amount of energy, and consequently, pressure inside of the containment. This pressure pulse may be mitigated by rapidly isolating both ends of the break with fast-operating isolation valves (ISO - II) to limit the amount of energy released to that which the containment (CNM - III) can withstand.

According to the PSID, the inside steamline isolation valves close in less than 3 seconds, necessitating the use of pneumatically actuated valves. The outside steamline isolation valves are all motor operated, and would, therefore, require approximately 40 to 50 seconds to close a 26-inch valve. The steam drum provides a common flowpath between the four steamlines. If the inside steamline isolation valves fail to close, all four steamlines blowdown through the break. Critical flow results from releasing the steam from the four 26-inch lines through the postulated 26-inch line break. Therefore, the three safety relief valves on each line would not lift because the opening pressure would not be reached until either isolation is established, and the secondary side of the steam generators boil dry following coastdown of the steam-driven feedwater pump, or the motor-operated steam bypass valves to the main condenser are fully opened following the turbine trip. This discharge inside the containment results in a more rapid containment pressurization than that which results from relief valves discharge into the suppression pool. The rapid rise in pressure increases the probability of containment failure compared with the probability of containment failure from accidents that do not rapidly pressurize the containment. Failure of the containment would cause an uncontrolled release of radioactivity into the environment although the primary system remains intact. The source of this radioactivity is the suppression pool that receives primary reactor coolant blowdown. The level of radioactivity in the pool will be increased the fuel has been damaged.

The response time of the secondary system to this initiator is very important. Lacking detailed thermal-hydraulic analyses, the response was inferred from an analysis of a high energy line break (HELB) in a conventional light water reactor.

The second challenge to the containment is like the LPPB challenge (section 7.3); it is the failure to remove decay heat (failure of APC and PPC) which results in excessive steam pressure discharging through the relief valves into the wet well. The wet well eventually heats and no longer condenses the steam that pressurizes and fails the containment.

Table 7-14 depicts the sequences of events just described. Categories of failure probability were given earlier; the frequency category of the steamline initiator is given in Table 7-2 as IV.

Sequences 1-3 represent successful scram and timely isolation of the steamline break followed by successful active pool cooling, which precludes overpressurization of the containment while maintaining core cooling and avoiding fuel damage: CC-1. Sequences 4-6 are like 1-3 except the isolation fails; nevertheless, the containment does not fail from the steam overpressure. Sequences 7-9 are like 4-6 except the containment fails, releasing radioactive gases from the containment; CC-2. Sequences 10-12 are like 1-2 except active pool cooling fails but the passive pool cooling succeeds, and the containment succeeds, giving CC-1. Sequences 13-15 are like 10-12

Table 7-14 Event Table for Steamline Break Inside Containment

No.	AS	MS	PS	ISO	APC	PPC	CNM	EWM	CC	EC
1	1	-	-	1	1	-	1	-	1	2
2	0	1	-	1	1	-	1	-	1	3
3	0	0	1	1	1	-	1	-	1	4
4	1	-	-	0	1	-	1	-	1	2
5	0	1	-	0	1	-	1	-	1	3
6	0	0	1	0	1	-	1	-	1	4
7	1	-	-	0	1	-	0	-	2	4
8	0	1	-	0	1	-	0	-	2	4
9	0	0	1	0	1	-	0	-	2	3
10	1	-	-	1	0	1	1	-	1	4
11	0	1	-	1	0	1	1	-	1	4
12	0	0	1	1	0	1	1	-	1	4
13	1	-	-	0	0	1	1	-	1	4
14	0	1	-	0	0	1	1	-	1	4
15	0	0	1	0	0	1	1	-	1	4
16	1	-	-	0	0	1	0	-	2	4
17	0	1	-	0	0	1	0	-	2	4
18	0	0	1	0	0	1	0	-	2	4
19	1	-	-	0	0	0	0	0	3	4
20	0	1	-	0	0	0	0	0	3	4
21	0	0	1	0	0	0	0	0	3	4
22	0	0	0	-	-	-	-	-	3	4

except the isolation fails but the containment succeeds, also resulting in CC-1. Sequences 16-18 are like 13-15 except the containment fails and releases the radioactive gases it contains, to give CC-2. Sequences 19-21 are like 16-18 except the emergency water makeup fails with resulting core damage, to give an CC-3. Lastly, Sequence 25 represents everything failing and giving an CC-3.

This boolean logic and the event categories were copied as an ASCII file for input to the BETA code. The resulting event tree is shown in Figure 7-12, the results of which are presented in Table 7-15.

Table 7-15 BETA Results for PIUS Steamline Break Inside Containment

Sequence Number	Sequence Probability Category	Consequence Category
1	IV	1
2	V	1
3	V	1
4	IV	1
5	V	1
6	V	1
7	V	2
8	V	2
9	VI	2
10	V	1
11	V	1
12	V	1
13	V	1
14	V	1
15	VI	1
16	V	2
17	VI	2
18	VI	2
19	VI	3
20	VI	3
21	VI	3
22	VI	3
Total CC-1	IV	-
Total CC-2	V	-
Total CC-3	VI	-

Figure 7-12 Event Tree for Steam Line Break Inside of Containment

7.8.2 Steamline Break Outside Containment (SBOC)

This postulated accident was not analyzed in the PSID. It is included here because of concerns about it being automatically detected fast enough to initiate steamline isolation (ISO). Otherwise, the resulting pressure drop on steam generator tubes could cause steam generator tube failure (SGTR). The Event Sequence Diagram for this event is shown in Figure 7-13.

Figure 7-13 Event Sequence Diagram for Steam Line Break Outside of Containment

Because the break is located outside the containment, two of the three modes of detection applicable to the SBIC do not apply to the SBOC. However, this location precludes failure of the containment due to overpressure. The location of the break is assumed to be any one of the four steamlines between the steam drum and the main turbine. Detection of the break flow comes from one of the following:

- 1) Low steamline pressure
- 2) High temperature in the turbine hall

A third mode of detection, high pressure in the turbine hall, was mentioned in the PSID. However, because of being located in a high elevation of the turbine hall, is believed to be ineffective for the SBOC because of the distance between a steamline location and the pressure sensors as well as the openness of the turbine hall. Depending on the setpoint of the low pressure sensors, a similar situation may exist for them as well. The progression of this event is similar to that of the SBIC except that the blowdown occurs into a large open volume that essentially maintains atmospheric pressure. This lack of pressure retention enhances the pressure differential across the tubes in the steam generators, thereby increasing the probability of failure of a steam generator tube. If this occurs, primary coolant will discharge from the steamline break as uncontrolled release of radioactivity to the environment..

As in the SBIC event, the timing of the responses of the plant and the secondary system to this initiator is important in determining the outcome. Table 7-16 shows Possible event sequences associated with the SBOC. Failure probability categories were presented previously except for steam generator tube rupture SGTR which is category II; the frequency category of the steamline initiator for the SBOC is taken to be the same as that for the SBOC, which is IV (see Table 7-2).

Table 7-16 Event Table for Steamline Break Outside Containment

No.	AS	MS	PS	ISO	SGTR	APC	PPC	CNM	EWM	CC	EC
1	1	-	-	1	1	1	-	-	-	1	2
2	0	1	-	1	1	1	-	-	-	1	3
3	0	1	1	1	1	1	-	-	-	1	4
4	1	-	-	0	1	1	-	-	-	1	2
5	0	1	-	0	1	1	-	-	-	1	4
6	0	0	1	0	1	1	-	-	-	1	4
7	1	-	-	0	0	1	-	-	-	2	4
8	0	1	-	0	0	1	-	-	-	2	4
9	0	0	1	0	0	1	-	-	-	2	4
10	1	-	-	0	0	0	1	1	-	2	4
11	0	1	-	0	0	0	1	1	-	2	4
12	0	0	1	0	0	0	1	1	-	2	4
13	1	-	-	0	0	0	0	0	1	2	4
14	0	1	-	0	0	0	0	0	1	2	4
15	0	0	1	0	0	0	0	0	1	2	4
16	1	-	-	0	0	0	0	0	0	3	4
17	0	1	-	0	0	0	0	0	0	3	4
18	0	0	1	0	0	0	0	0	0	3	4
19	0	0	0	-	-	-	-	-	-	3	4

Sequences 1-3 represent successful scram and timely isolation of the steamline break followed by successful active pool cooling, hence a CC-1. Sequence 4-6 are like 1-3 except the isolation fails; nevertheless, the steam generator tubes do not rupture which gives CC-1. Sequences 7-9 are like 4-6 except the steam generator tubes rupture which releases radioactive gases from discharged primary water, hence CC-2. Sequences 10-12 are like 7-9 except the active pool cooling fails but the passive pool cooling succeeds, thus leading to CC-2. Sequences 13-15 are like 10-12 except the passive pool cooling fails which causes the containment to fail but the emergency water makeup keeps the core covered, resulting in CC-2. Sequences 16-18 are like 13-15 except the emergency water makeup fails, uncovering the core, and a CC-3. Lastly, Sequence 19 represents everything failing, to give a CC-3.

The boolean logic and the event categories of Table 7-16 were copied as an ASCII file for input to the BETA code. The resulting event tree is shown in Figure 7-14, the results of which are presented in Table 7-17.

Table 7-17 BETA Results for PIUS Steamline Break Outside Containment

Sequence Number	Sequence Probability Category	Consequence Category
1	IV	1
2	V	1
3	V	1
4	IV	1
5	V	1
6	V	1
7	V	2
8	V	2
9	V	2
10	V	2
11	VI	2
12	VI	2
13	VI	2
14	VI	2
15	VI	2
16	VI	3
17	VI	3
18	VI	3
19	VI	3
Total CC-1	IV	1
Total CC-2	V	2
Total CC-3	VI	3

Figure 7-14 Steamlinebreak Outside of Containment Event Tree

7.10 Feedwater Transients

Feedwater transients (FWT-II) include a group of malfunctions of the steam generator secondary side system that result in the complete or partial loss of the feedwater supply to the plant. These malfunctions include failures of the main feed pump, errors of the feed pump control system and control valve, and ruptures of the feedwater system. The response of the primary and secondary systems will be similar following each of these events and the loss of the main feed pumps is expected to provide a worst-case representation of the plant response in terms of the potential for fuel damage. Thus, the analysis presented in this section will be based on the loss of all sequence of main feed pumps. The event sequence diagram is presented in Figure 7-15.

Figure 7-15 Feedwater Transient Event Sequence Diagram

After the initiation of a FWT, the heat removal capability of the secondary system will be degraded and the primary system pressure and temperature will increase. Based on either high pressure of the primary system or high loop temperature, the reactor trip and interlock system will provide a signal to trip one of the main coolant pumps (AS-III) and shutdown the plant. If this active scram function fails, the reactor operator can initiate a manual scram (MS-II) to trip the scram pump and accomplish the shutdown. Should both of these scram functions fail, a passive scram (PS-IV) would be actuated as the increased primary temperature and pressure caused by the FWT upsets the pressure balance between the primary and the reactor pool, causing an influx of borated pool water to the primary system and completing the shutdown. Failure of all three scram mechanisms would cause overpressure condition on the primary side and result in a challenge to pressure relief system (PRS-III). An additional failure of the primary pressure relief system could cause overpressure in the containment overpressure condition with consequences similar to the scenarios of large and small primary pipe breaks discussed in Sections 7.3 and 7.4.

Following a successful scram and reactor shutdown, the active pool cooling (APC-III) system is required to transfer decay heat from the reactor pool, through the heat exchangers, to the cooling towers. If the active pool cooling fails, decay heat can be removed from the system by passive pool cooling (PPC-IV). Failure of both could result in the loss of the pool water inventory through boiling and overpressure in the primary system. As discussed in the previous paragraph, challenges to the primary relief system could have consequences similar to those resulting from large and small primary pipe breaks.

The events discussed above have been assembled into Event Table 7-18. Ten sequences were identified with Sequences 1 through 9 including success of one of the three scram functions combined with various failures

of the active and passive pool cooling systems. Sequence 10 assumes the failure of all scram mechanisms. It is assumed that it is not possible to remove full reactor heat through either active or passive pool cooling. Thus, core melt (CC3) will result regardless of subsequent actions.

Table 7-18 Feedwater Transients Event Table

No.	AS	MS	PS	PRS	APC	PPC	CC	EC
1	1	-	-	-	1	-	1	1
2	1	-	-	-	0	1	1	3
3	1	-	-	-	0	0	2	4
4	0	1	-	-	1	-	1	1
5	0	1	-	-	0	1	1	4
6	0	1	-	-	0	0	2	4
7	0	0	1	-	1	-	1	3
8	0	0	1	-	0	1	1	4
9	0	0	1	-	0	0	2	4
10	0	0	0	-	-	-	3	4

This event table was processed with the BETA code to produce the event tree shown in Figure 7.16. In addition a summary of the frequency categories for each of the consequence categories is shown in Table 7.19.

Table 7-19 BETA Results for PIUS Feedwater Transients

Sequence Number	Sequence Probability Category	Consequence Category
1	I	1
2	IV	1
3	V	2
4	IV	1
5	IV	1
6	V	2
7	V	1
8	V	1
9	VI	2
10	V	2
Total CC-1	I	1
Total CC-2	V	2
Total CC-3	VI	3

Figure 7-16 Event Tree for Feedwater Transient

7.11 Computer Malfunction (CM)

PIUS uses extensive main and function (local microprocessors and local microcomputers) computer control and display. Reliability is assured by tandem operation i.e. dual computers, one receiving process information and executing algorithms for the control and display functions, and the other in a backup mode performing the same processes but not performing control or display functions. Should the active computer fail, control is automatically transferred to the standby computer. The event sequence diagram for computer malfunction is presented in Figure 7-17.

Figure 7-17 Computer Malfunction Event Sequence Diagram

Computer failure may result from the causes presented in Appendix A.7, such as multiple independent failures, software failures, or common cause failures. A potential common cause that has been identified may result from the variable speed control of the primary pumps by thyristors. The rapid rise-time of a semi-conductor, thyristor switch produces harmonic components in the mega-Hertz region (called electromagnetic interference - EMI) which will couple into all plant circuits unless prevented. While such interference is minimized at full power, even then, large amounts of power are being switched. The PSID did not discuss this problem, and it is assumed that

it will be addressed by some design features such as, electrical noise suppression circuits, separation, shielding, etc. Here, it is assumed that the design feature (e.g. noise suppression equipment) fails and the EMI causes the computer to malfunction. Such an EMI might also be caused by lightning or a large electromagnetic pulse (EMP) external to the plant. The most likely malfunction is that the noise will cause the computer to address an unallowable memory field and the computer will fail to operate. The PSID does not describe assurance that the plant will go to a safe condition on computer failure (i.e., fail safe).

This section bounds the computer malfunction by considering the following possible scenarios:

1) Cessation of boration for the primary system causing a positive reactivity excursion or ramp (RR). It is assumed that the primary-pump speed control tracks the excursion to the maximum speed of the pumps (105%), and

2) Closure of the Main Steam Isolation Valves (MSIV), thus isolating the reactor from the normal heat sink.

The event sequences are: computer malfunction (CM - III) results in a reactivity ramp (1-RR - IV)¹, and/or a closure of the main steam isolation valve (1-MSIV - IV). Because of the computer failure, active scram and manual scram via the computer are not possible; thus, they are not included in the sequences. However, when a pump reaches its limiting speed, a further increase in reactivity activates the pool loop, hence, a passive scram (PS - IV). Similarly, the MSIV closure upsets the heat balance and activates the pool loop and passive scram. The MSIV closure at power will result in excessive steam pressure tripping the steam generator pressure relief and blowdown to the suppression pool. This sequence would not pose a challenge to the containment unless the reactor fails to scram and feedwater continues to supply sufficient steam to overpressurize the containment, however a failure to scram will overpressurize the containment regardless of MSIV closure. After scram, active pool cooling (APC - III)² removes the decay heat, or if it fails, passive pool cooling (PPC - IV) will remove the decay heat. If these fail, the primary pressure will trip the relief valves and primary steam will discharge into the suppression pool until it becomes overheated, and then steam will pressurize the containment (CNM - III) - possibly causing it to fail. If emergency water makeup (EWM - II) is brought in, the core will remain covered; otherwise the core will be damaged.

Table 7-20 presents these scenarios as an event table. This table was converted to be input to the BETA code, and processed to generate the event tree shown in Figure 7-18; the summary of results is presented in Table 7-21.

To summarize the computer malfunction scenarios, even though extreme actions are hypothesized to result from the failure of the computer, the passive scram and passive cooling provide protection to reduce the probability of core melt to the extremely rare category (Frequency Category V).

¹ The notation (1-RR) means the event is "not RR", since this is an aggravating, not a mitigating event. Thus, a failure of (1-RR) is a reactivity ramp. This terminology is adopted to be consistent with the lower part of an event tree, indicating that there is more damage than the upper part of the event tree. Similarly, (1-MSIV) means "no MSIV closure."

² It is not known whether a computer malfunction could affect the operation of the active pool cooling. There seems to be no need for this system to be computer-controlled, thus, it may be operated from local control centers. However, if it is computer controlled, the rather high failure rate assigned to APC probably is sufficient to subsume this probability.

Table 7-20 Computer Malfunction Event Table												
No.	I-RR	I-MSIV	PS	APC	PPC	CNM	EWM	CC	EC			
1	1	0	1	1	-	-	-	1	1			
2	1	0	1	0	1	-	-	1	4			
3	1	0	1	0	0	0	1	2	4			
4	1	0	1	0	0	0	0	3	4			
5	0	1	1	1	-	1	-	1	1			
6	0	1	1	1	1	0	-	2	4			
7	0	1	1	0	1	1	-	1	4			
8	0	1	1	0	0	0	1	2	4			
9	0	1	1	0	0	0	0	3	4			
10	1	1	1	1	-	-	-	1	1			
11	1	1	1	0	1	-	-	1	3			
12	1	1	1	0	0	0	1	2	4			
13	1	1	1	0	0	0	0	3	4			
14	1	1	0	-	-	-	-	3	3

Figure 7-18 Computer Malfunction Event Tree

Table 7-21 BETA Results for PIUS Computer Malfunction

Sequence Number	Sequence Probability Category	Consequence Category
1	V	1
2	VI	1
3	VI	2
4	VI	3
5	V	1
6	VI	2
7	VI	1
8	VI	2
9	VI	3
10	III	1
11	V	1
12	VI	2
13	VI	3
14	V	3
Total CC-1	III	1
Total CC-2	V	2
Total CC-3	V	3

7.12 Severe Seismic Event

Seismic effects on PIUS, like any other complex system, are difficult to anticipate even with very detailed spring-mass models.

Here, we will develop an extreme scenario and explore the possibilities using an event tree. The postulated scenario is that the plant receives a shaking considerably in excess of the design basis acceleration (Figure 7-19). The event sequence diagram is presented in figure 7-20.

Primary concerns are associated with the low resonant frequency of the core-flow structure in a very large amount of water. Comparing PIUS with other LWRs suggests that in the absence of the pool the core-flow structure would have a low frequency because of the length between supports and the comparative thinness of the flow structure (it is not a pressure boundary and, therefore, does not require strength). The core is located close to the PCR support, thereby producing a lever arm that amplifies the horizontal motion of the flow structure in the mid plane - this also tends to lower the resonant frequency of the core-support structure. Horizontal motion of the flow structure causes water to be displaced in the direction of motion which must flow around the core-flow structure. When the horizontal motion of the flow structure reverses, water must flow back. The transit time of this motion produces a phase shift which can, at certain frequencies, result in resonance between the water motion and the core-flow support structure. In addition, the effects of coupling the core-flow structure to the pool water is to lower the resonant frequency of the core-flow structure. The NRC Regulatory Guide 1.60 for seismic design spectrum shows increasing power at lower frequencies. Under these circumstances, the core-flow structure may be susceptible to damage at lower seismic levels than other parts of the plant. Damage to this structure could inhibit the passive core cooling and lead to core damage.

Figure 7-19 Postulated Seismic Motion

The postulated scenario is the following:

- The reactor is at full power when the severe seismic (SS - V) event occurs.
- The PCRV being 10 meters thick concrete with extensive steel reinforcing maintains its integrity,
- Offsite power is lost,
- Primary pipes may be severed (LPPB - II) ,
- Critical heat flux may be exceeded (CHF - II),
- Passive scram may fail (PS - III),
- Active pool cooling may be lost (APC - II),
- Passive cooling may be lost (PPC - III),
- The containment may crack and leak (CNM - II), and
- Emergency water makeup (EWM - II) may be ineffective in cooling the core.

Higher probability categories are used than shown in Table 7-2 because of the increased probability of failure caused by the seismic event. The scenarios that may result are presented in Table 7-22.

Figure 7-20 Severe Seismic Event Event Sequence Diagram

Table 7-22. PIUS Severe Seismic Events

No.	1-LPPB	CHF	PS	APC	PPC	CNM	EWM	CC	EC
1	1	-	1	1	-	-	-	1	3
2	1	0	1	1	-	1	-	1	3
3	1	0	1	0	1	1	-	1	4
4	0	1	1	1	-	0	-	2	4
5	0	0	1	1	-	1	-	1	4
6	0	0	1	0	1	1	-	1	4
7	0	0	1	0	0	0	1	2	4
8	0	0	1	0	0	0	0	3	4
9	1	1	0	-	-	-	-	3	4

In sequence 1, the pipes do not break, the critical heat flux is not exceeded, passive scram and active pool cooling all take place. In Sequence 2, the critical heat flux is exceeded causing fuel damage but the containment holds. Sequence 3 is like 2, except active pool cooling fails but the passive pool cooling succeeds. In sequence 4, the pipes break releasing contaminated water inside the containment, the critical heat flux is not exceeded, but the containment fails from seismic effects. In sequence 5, the pipes break, critical heat flux is exceeded but the containment holds. Sequence 6 is like 5, active pool cooling fails, the passive pool cooling succeeds as does the containment. Sequence 7 is like 6 but the passive pool cooling fails and emergency water makeup keeps the core covered. Sequence 8 is like 7 except the emergency water makeup fails to protect the core from overheating. In sequence 10, the passive scram fails because of damage to the flow structure, the reactor at power overheats and melts the core.

The event table logic was copied as input and run in the BETA code to produce the event tree shown in Figure 7-21. The results are shown in Table 7-23.

Figure 7-21 Severe Seismic Event Tree

Table 7-23 BETA Results for PIUS Severe Seismic Event

<u>Sequence Number</u>	<u>Sequence Probability Category</u>	<u>Consequence Category</u>
1	V	1
2	V	1
3	V	1
4	V	2
5	V	1
6	VI	1
7	VI	2
8	VI	3
9	VI	3
Total CC-1	V	1
Total CC-2	V	2
Total CC-3	VI	3

7.13 Refueling Accident (RA)

The description of refueling is incomplete in the PIUS-600 PSID; hence, it was necessary to apply engineering judgment to the material provided.

Description of Refueling PIUS

Before beginning refueling, the reactor primary is borated to the same concentration as the reactor pool. Activation of the pool loop will fill the primary circuit enclosed by the flow structure. (It may not be necessary to completely fill the primary loop with borated water.) Figure 7-22 shows the upper part of the PIUS plant structure used for refueling.

Figure 7-22 PIUS Refueling Structures

Figure 7-23 presents the event sequence diagram for flow blockage during refueling.

PIUS refueling begins by removal of the dome identified as (A) in the figure. It is assumed that the reactor's internal pool is empty, as is the volume between (A) and the pressure dome (B) that closes the primary circuit. (If this space were normally filled with water, there would be considerable heat loss.) After removing dome (A), using the service bridge, it is stored in the reactor's internals storage pool. The reactor's internals pool is flooded with water, borated to the same concentration as the reactor pool, to prevent accidental criticality. Flooding submerges the outside of primary piping where it connects with the flange of the pressure dome. Additional borated water is added to the reactor primary until the pressure dome is filled after which the pressure dome (B) is removed with the service bridge and relocated, out of the way in the reactor internals storage pool. At this point, access to the reactor internals has been gained.

Figure 7-23 Refueling Flow Blockage Event Sequence Diagram

System 231 consists of the two subsystems, the refueling machine and the service bridge. At this stage in the refueling, the service bridge is used to replace burnable absorber rods in fuel assemblies and to act as a platform for the tools needed to remove and replace the vessel internals.

Refueling Machine

The refueling machine (2311 - item D, Figure 7-22) transports fresh fuel to the intermediate storage racks in the reactor service pool (system 244) for refueling, and transports fuel from the reactor core to the intermediate storage racks in system 244 and back to the core for reload. Several times in the plant's life, the machine is used to transport fuel from the spent fuel storage racks to spent fuel transportation casks.

The refueling machine consists of a trolley operating on a tracked bridge. The trolley has two working platforms, control desk, two hoists, and a telescope. At the bottom of the telescope is a fuel assembly grapple. The grapple is approximately positioned by geometric coordinates and a level; it is precisely positioned manually by telescopic and television observation using vernier tilting of the telescope to engage fuel elements.

The bridge has two end bogies and two transverse girders. It straddles the protective railing around the pool. Thyristors control is used to position the bogies that ride on rails mounted on the floor of the service room. The trolley consists of a framework that constitutes the lower working platform and a tower for the telescope. Thyristors control the positioning of the trolley which runs on rails mounted on the bridge girders. The welded plate tower includes thyristor-controlled hoists, limit switches, and the upper working platform. The control desk on the service bridge, which includes warning lights, indicating instruments, a TV monitor, a sight drum, and communications to the control room, is located on the lower platform which contains tool racks for the equipment needed for refueling.

The telescope has one fixed tube and four telescoping tubes. The telescoping tubes are constrained from rotation. The fixed tube, mounted on the trolley, indicates and may be rotated 360°. A guard ring prevents the telescope touching the pool wall. The bottom of the inner telescoping tube has a pneumatic grab, TV camera, and floodlights. The grab cannot be opened when supporting a weight.

When the telescope is fully extended, the grapple is at the same level as the bottom edge of the core grid. Grapple positions and the unloaded condition is sensed by microswitches that monitor them on the control desk. The load on the elevator cable is sensed and monitored to provide overload protection.

Service Bridge

The service bridge (2315 - item C in Figure 7-22) contains an electric wrench which is lowered into the reactor internals to remove the studs and nuts holding the internals assemblies. The first parts removed are the internals associated with the pressurizer, including flow structures (item E in the figure) and the siphon breaker. These are unbolted and removed with the bridge and stored in the internals service pool. Next, the upper three parts of the flow structure are removed and stored in the reactor service pool so that the core in the bottom flow structure is exposed for refueling.

The service bridge consists of a traveling bridge, service platform, and crane. It runs on rails along the reactor service pool (244) and the adjacent pools. It can operate from the floor of the reactor service room and over the reactor service pool. The swinging crane can bring fresh fuel from the fuel service pool to the intermediate storage racks and can handle tools for vessel internals.

When the plant is operating, the refueling machine and the service bridge are parked on the reactor service room floor.

Accident Scenarios

At refueling, PIUS-600 is shutdown, and pool water convectively circulates through the core with the heat being removed with the Active pool cooling (APC - III). Should APC fail, passive pool cooling (PPC - IV) will remove the decay heat. Radioactive releases will be confined by the containment (CNM - V). The general types of accidents that may be envisioned at refueling are:

- 1) Criticality while refueling,
- 2) Fuel element drop,
- 3) Drop of a heavy object to damage many fuel elements in situ,
- 4) Flow blockage, preventing convective core cooling causing overheating of the fuel.

1) Criticality while Refueling

In a light water reactor, criticality may occur by inadvertent withdrawal of the control rod³. In PIUS-600, the core is held subcritical by the highly borated pool water. Only by deborating the pool is it possible to go critical. This is a lengthy process (we estimated several days) which cannot be done accidentally because of the accumulation and storage of boric acid.

Would a change in geometry change the neutron multiplication? Light water reactor cores, and presumably PIUS is no exception, are designed to provide maximum neutron multiplication in the normal geometry. Thus, changes in the geometry from earthquakes, or refueling will result in less multiplication, and hence, greater safety from criticality.

In conclusion, we judged that criticality while refueling is bounded by other refueling risks, and will not

³ Improper placement of rods in the spent fuel storage pool and insufficient boration of this pool are other criticality mechanisms. Without details about the spent fuel storage pool, it is assumed that the pool is designed such that it is impossible to achieve criticality by the fuel elements. It is also assumed that the neutron lifetime is continuously monitored to avoid criticality misplacing and that the boron concentration is administratively controlled.

be further considered.

2) Fuel Element Drop

The PSID says that a "pneumatic grab" is used moving the fuel elements into place. This statement is interpreted to mean that instrument air is used to close the grab⁴. It is conceivable (frequency category III) that the grab may close improperly or a part of the grab may fail. The element may be dropped when it has nearly reached the refueling platform, thus falling about 30 m through water to hit the remaining fuel in the core. No attempt has been made to estimate the velocity at the time of impact - it is assumed that both the dropped element and the impacted element are damaged and release radioactivity (CC-2).

There are some modifying effects on this scenario: It is quite possible that the dropped element will hit structural material and only the dropped rod damaged. In order to impact more than one fuel element, the dropped element must fall sideways; but if it did, the flow drag would greatly reduce the velocity, and hence, the force of the impact as well as distributing the force over a large area (less pressure).

In conclusion, the dropped-element accident is estimated to have a frequency class III and CC-2.

3) Heavy Object Drop

It is clear that no heavy object can be dropped until the reactor is open and sufficient flow structure has been removed to expose the core. The refueling bridge and the service bridge are heavy objects that could be dropped, but these are believed to be so large that they cannot fall through the opening at the top of the reactor.

The most obvious heavy object that could be dropped is the telescope⁵. Failure of the lift cable would allow the telescope to move to its full extent, which allows the grapple to reach the core support. The telescope would fall rapidly because of its stable motion with low fluid drag. The portion actually impacting the fuel is the bottom section of the telescope, grab, optics, and TV equipment.

The frequency category of the telescope drop is estimated to be IV, and the damage would be CC-2. Some local heating may occur from blockage of flow channels.

4) Flow Blockage during Refueling

Dropping a sheet of material onto the top of the reactor would block the convective coolant flow and cause the core to overheat. Such material must have a density greater than 1 to settle onto the core and not be lifted by the convective flow. Such material has not been identified, however details of the refueling process are not available. It is believed that such an occurrence would be very rare (FB - frequency category IV). If this happened, the operator's first action would be to remove the blockage (RB - II) although tools would have to be improvised and may not be successful. Blockage would cause local overheating (possibly exceeding the critical heat flux CHF - II) unless the blockage is removed. In the blocked region, steam bubbles would leak from the blockage and from the bottom of the core. Problems could be compounded by local fuel damage, releasing radioactivity that

⁴ Another interpretation could be that the grab uses pumped water flow to engage the element by differential pressure. While this would be an effective means of initially engaging the element, it is not sufficient under reduced pressure as the element is lifted.

⁵ Spent fuel casks are other heavy objects that could be dropped, but this is not considered because they are not passed over the core, there does not appear to be enough hook clearance, and because their diameter is too large to pass through the opening to the core.

could force an evacuation of the refueling bridge and stop efforts to remove the blockage. In the most extreme case, the containment (CMN - III) may fail due to overpressure; the pool water may boil away tending to uncover the core unless emergency water makeup (EWM - II) is provided. This scenario is presented for completeness and it is again noted that such a material capable of causing a blockage has not been identified. Further, the possibility of fuel failure, pool boil-off and containment failure should be quantified.

Table 7-24 presents the scenarios associated with this type of accident.

Table 7-24 PIUS Flow Blockage during Refueling Event Table

No.	RB	CHF	CNM	EWM	CC	EC
1	1	1	-	-	1	2
2	1	0	1	-	2	2
3	1	0	0	-	2	4
4	0	0	0	1	2	4
5	0	0	0	0	3	4

Table 7-24 is converted to be inputted into the BETA code and processed to generate the event tree shown in Figure 7-24 and the summary of results, presented in Table 7-25.

Figure 7-24 Flow Blockage during Refueling Event Tree

Table 7-25 BETA Results for Flow Blockage during Refueling		
Sequence Number	Sequence Probability Category	Consequence Category
1	IV	1
2	IV	2
3	V	2
4	V	2
5	V	3
Total CC-1	IV	1
Total CC-2	IV	2
Total CC-3	V	3

7.14 Section Summary

Table 7-26 summarizes the results from the event tree analysis bearing in mind the definitions of the consequence categories (CCs) as presented in Section 3.5.

Table 7-26 Summary Frequency Categories from Event Tree Analysis of PIUS

Accident Title Section	CC-1	CC-2	CC-3
Large Primary Pipe Break (7.4)	IV	V	VI
Small Primary Pipe Break (7.5)	IV	VI	VI
Loss of Off-Site Power (7.6)	I	V	IV
Turbine Trip (7.7)	I	IV	VI
Steamline Break Inside Containment (7.8.1)IV	V	VI	
Steamline Break Outside Containment (7.8.2)	IV	V	VI
Feedwater Transient (7.9)	I	V	V
Computer Malfunction (7.10)	III	V	V
Severe Seismic (7.11)	V	V	VI
Fuel Element Drop (7.12)	-	III	-
Flow Blockage during Refueling (7.13)	IV	IV	V

These analyses show the considerable protection provided by the passive protection features to PIUS-600. The most frequent CC-2 accident is the fuel element drop; the most frequent CC-3 class accident is the loss of off-site power.

References

R.R. Fullwood and W.G. Shier, "PRA Using Event Tables and the Brookhaven Event Tree Analyzer," Reliability Engineering and System Safety, Elsevier London, 30 pp. 79-82, 1979.

NRC Memorandum regarding NRC Event Category Method, Z.R. Rosztoczy to J.E. Dyer, 8/17/92.

8.0 PIUS SYSTEM DEPENDENCIES AND SAFETY CLASSIFICATION

8.1 Introduction

The previous section presented the results of the event tree analysis for the accident initiators that were identified for this study. Each event tree included, across the top of the event tree, the so-called front-line systems that are needed to mitigate the initiator. This section expands the discussion of those front-line systems by identifying the support systems that are required to function during these events. That is, the dependencies of the front-line systems are identified. In addition, the safety classification scheme of PIUS is discussed, and the safety classification of both front line and support systems is given, according to the PSID. This Appendix incorporates comments, as appropriate, from the May 11 ABB Atom letter.

8.2 System Safety Classification

Section 3.2 of the PSID discusses the "Classification of structures, components, and systems." The PSID states that the safety classification for PIUS is based on the same principles as for present day LWRs. Mention is made of ANSI/ANS 51.1 (1983), "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," however no commitment to following this standard is given. Most (but not all) of the various classification schemes used for PIUS: are defined in the PSID, are similar to those in ANSI/ANS 51.1, and will be summarized below. In section 3.2 of the PSID, Table 3.A (which is similar to Table 8-1 below) correlates some of the classifications used. The notes to Table 3.A use the undefined term safety-related. Additionally, in the PSID "Systems Chapters" (Chapters 100 - 800 of Volume III), a slightly different terminology is used. In a number of these system and subsystem chapters, the PSID designates that system as "safety grade" or "not safety grade." Other system chapters are silent relative to the designation of that system as safety grade. Further, the PSID does not appear to provide a definition of safety grade, as used in these chapters, or a correlation between safety grade and the designations in Table 8.1 below.

Table 8-1 Summary of PIUS Safety Classifications			
PIUS Safety Class (SC-)	Seismic Class	Mechanical Class	Electrical Class
1	I	A	IE
2	I	B	IE
3	I	C	IE
N	N	D	C

The four PIUS classifications of this Table are discussed in more detail below, using quotes from the PSID:

Class 1 "shall apply to pressure-retaining portions and supports of mechanical equipment that form part of the RCPB, whose failure could cause a loss of reactor coolant in excess of the reactor coolant normal makeup capability and whose requirements are within the scope of the ASME Boiler and Pressure Vessel Code, Section III." This means the PCR/V up to and including isolation valves. However consideration must be given to PIUS where the upper and lower parts have very different safety implications. The lower PCR/V (e.g. PCR/V monolith) is assigned to a higher safety class (1) than the upper parts (2).

Class 2 "shall apply to pressure-retaining portions and supports of primary containment and other mechanical equipment, requirements for which are within the scope of the ASME Boiler and Pressure Vessel Code,

Section III, that is not included in SC-1, and is designed and relied upon to accomplish the following nuclear safety functions:"

- a) Fission product barrier, primary containment, radioactive material holdup or isolation,
- b) Emergency heat removal of the primary atmosphere to an intermediate heat sink, or emergency removal of radioactive material from the primary containment atmosphere,
- c) Introduce negative reactivity into the reactor, or restrict the addition of positive reactivity,
- d) Emergency core cooling,
- e) Maintain coolant inventory for emergency core cooling.

PIUS does not have many of the systems mentioned in the code. The PIUS interpretation is:

- The upper parts of the PCRV including containment isolation valves,
- Reactor internals that are important for maintaining core geometry and coolant flow,
- Safety valve function,
- Reactor shutdown via coolant pump trip,
- Containment.

Safety Class 3 "shall apply to equipment, not included in SC-1 or SC-2, that is designed and relied upon to accomplish the following nuclear safety functions:"

- a) Provide for functions of SC-2 equipments or portions are not within the scope of ASME Section III,
- b) Provide secondary containment radioactive material holdup, isolation or heat removal,
- c) Ensure hydrogen concentration control,
- d) Removal of radioactive material from spaces outside of the containment,
- e) Introduce negative reactivity to achieve or maintain subcriticality,
- f) Provide or maintain reactor coolant inventory,
- g) Maintain geometry within the reactor to ensure reactivity control and core cooling,
- h) Structurally load-bear or protect SC-1, -2 or -3 equipment,
- i) Radiation shielding of the control room and offsite personnel.
- j) Cooling of liquid-stored fuel,
- k) Functionability of SC-1, -2, -3 equipment,
- l) Power for SC-1, -2, -3 equipment,
- m) Signals or power for SC-1, -2, -3 equipment,
- o) Manual or automatic interlock for functionability of SC-1, -2, -3 equipment,
- p) Acceptable environment for SC-1, -2, -3 equipment.

The two seismic classes of Regulatory Guide 1.29, I (fully seismic) and N (not seismic), are supplemented with a Seismic Class P (passive). This means that the passive function and structural integrity of the equipment must be maintained, but there is no requirement for active function i.e. a pressure boundary must be maintained but pumps need not work. It should be noted that PIUS considers the combined occurrence of an SSE and a LOCA.

Mechanical classifications correspond to the mechanical quality classes, and electrical equipment classification correspond to IEEE 308 (1974). However, PIUS uses the terminology IE rather than 1E for safety related electrical items. An electrical classification of "C" indicates conventional electrical equipment and a "-" indicates that there is no electrical function for that equipment.

The simplified Table (8-1) shows the relationships between these four classification systems. Table 8-2, which was adapted from Table 3.A of the PSID, provides a detailed breakdown of PIUS subsystems, showing their various safety classifications. Non-safety equipment or subsystems (that is, equipment having a safety designation

N, seismic classification N, mechanical quality D, and electrical classification C are not included in this table.

For a listing of all PIUS systems including those designated as non-safety, see the PSID Table 3.A.

Table 8-2 Safety Classification of PIUS Structures, Components and Systems					
System No.	Description	Safety	Seismic	Mechanical	Electrical
100	Buildings, External Plant Systems				
121	Reactor (Rx) building (bldg) containing safety-related parts	3	I	C	-
125	Control systems bldg, main control room	3	I	C	-
151	Concrete vessel monolith	1	I	A	-
152	Concrete containment	2	I	B	-
157	Concrete vessel liner cooling system	N	P	D	-
159	Concrete vessel liner venting system	3	I	C	-
200	Rx & Rx Auxiliary Systems				
211	Pressure vessel upper parts ¹	2	I	B	IE
213	Reactor internals - for shutdown and core cooling (sd&cc)	2	I	B	-
215	Reactor assembly supports and restraints for sd&cc	2	I	B	-
216	Vessel internal instrumentation support equipment for sd&cc	2	I	B	-
231	Refueling parts to prevent dropping of fuel elements	3	P	C	-
233	Refueling handling parts to prevent equipment dropping on reactor	3	P	C	-
241	Fresh fuel storage equipment	N	P	D	-
243	Spent fuel storage equipment	N	P	D	-
261	Fuel assemblies	2	I	B	-
300	Reactor Plant Process Systems				
312	Reactivity control system integrity ¹	2	I	B	-
313	Recirculation system integrity	2	I	B	-
314	Pressure relief system (PRS) integrity	2	I	B	-
314	PRS safety valves	2	I	B	-
314	PRS other parts	3	P	B	C
316	Condensation system	2	P	B	IE

Table 8-2 Safety Classification of PIUS Structures, Components and Systems					
System No.	Description	Safety	Seismic	Mechanical	Electrical
321	Reactor pool	2 (only waterfill of 151 cavity)			
322	Pool loop inactivation system integrity ¹	2	I	B	-
323	Reactor pool cooling integrity ¹	2	I	B	-
325	Reactor pool water replacement system ¹	2	I	B	-
328	Pressurizer system integrity ¹	2	I	B	-
331	Reactor water cleanup system ²	3	P	C	C
332	Plant shutdown reactor pool cleanup integrity ¹	2	I	B	-
341	Offgas delay system ²	3	P	C	C
342	Liquid waste system ²	3	P	C	C
343	Solid waste system ²	3	P	C	C
345	Monitored floor drains, parts for safety-related room	3	I	C	IE
349	Offgas filter systems (if excess radioactivity)	3	P	C	C
400	Steam & Power Conversion Systems				
500	Control Equipment				
511	Main control room	N	P	D	C
515	Control cables ³	3	I	C	IE
516	Trip and interlock system	3	I	-	IE
518	Emergency control room	3	I	C	IE
520	Computer Systems ³	3	I	C	IE
531	Neutron flux monitoring systems ³	3	I	C	IE
533	Water chemistry control system, boron monitoring ¹	3	I	C	IE
536	Reactor, pool, concrete vessel monitoring ³	3	I	C	IE
554	Steam generator tube rupture mon. sys.	3	I	C	IE
555	Room radiation monitoring systems ³	3	I	C	IE
583	Containment monitoring systems ³	3	I	C	IE
600	Electrical Equipment				
649	MCP power supply system ³	3	I	C	IE
673	Safety grade 110 VDC aux. power supply	3	I	C	IE

Table 8-2 Safety Classification of PIUS Structures, Components and Systems

System No.	Description	Safety	Seismic	Mechanical	Electrical
680	Electrical power system, control system ³	3	I	C	IE
690	Power cable systems ³	3	I	C	IE
700	Service Systems				
714	Cooling towers for reactor pool coolers	3	I	C	IE
721	Diesel-backed closed cooling system for reactor pool ⁴	3	I	C	IE
722	Diesel-backed closed cooling system for component cooling ⁴	3	I	C	IE
724	Closed cooling system for reactor pool coolers	3	I	C	IE
727	Local air coolers ³	3	I	C	IE
733	Demineralized water distribution system ⁴	3	I	C	IE
734	High pressure distilled water system ⁴	3	I	C	IE
741	Containment ventilation system ⁴	3	I	C	IE
742	Reactor building ventilation dampers	3	I	C	IE
746	Control building ventilation system	3	N	C	IE
749	Emergency ventilation system	3	N	C	IE
751	Bottled air system	3	I	C	IE
753	Compressed air system ⁴	3	I	C	IE
800	Other Plant Equipment				
810	Transport equipment, reactor service room overhead crane	N	P	C	C

Notes to Table:

1. RCPB portion of system up to and including containment isolation valves (CIVs).
2. These systems may contain significant amounts of radioactive material. If the consequences of a rupture may be close to regulatory limits, then classify as shown, otherwise system is non-safety related (N/N/D/C).
3. Safety-related parts of these systems only.
4. These systems may have lines penetrating the containment. Only the CIVs and the lines between the CIV and containment wall are classified as shown.

8.3 System Dependencies

Figure 8-1 and Table 8-3 provide a top level representation of the dependencies of the major PIUS systems. This figure and table use the PIUS terminology presented in Section 2. A more detailed system dependency listing (down to the subsystem level) is provided in Appendix C.

The event tree analysis described in Section 7 identified the following PIUS systems as being significant to the mitigation of the transients that were analyzed:

AS Active Scram
MS Manual Scram
PS Passive Scram
APC Active Pool Cooling
PPC Passive Pool Cooling
BYPOP Steam Bypass System
ISO Steamline Isolation
CNM Containment
EWM Emergency Water Makeup

Appendix E gives a detailed description of the PIUS system dependencies. Table E-2 and the PIUS PSID was used to construct the following tables for the systems that have been identified in the event tree analysis. The systems listed under each mitigating system are support systems that are believed to be necessary to operate in order to successfully complete the mitigating function. In addition, for each support system the following information is included: PIUS system number; the safety classification associated with each system using the nomenclature of Appendix E, Table E-2 (i.e., ND-not designated, NS-non safety, PS-passive safety, and SG-safety grade), and the safety class per Table 3.A of the PSID. In some cases not all of the system was the same class; this is indicated by giving the higher class and noting it as "part."

Active Scram

5161	Reactor Scram Trip System	(SG) (3)
536	Reactor, Pool, and Vessel Monitoring	(NS) (3, part)
325	Rx Pool Water Replacement System	(PS) (2, part)
520	Main Process Computer	(NS) (3, part)

Manual Scram

5161	Reactor Scram Trip System	(SG) (3)
5162	Hot Reactor Shutdown Trip System	(SG) (3)
---	Operator Manual Actions	(ND) (-)

Passive Scram

3221	Lower Density Lock	(ND) (N)
3222	Upper Density Lock	(ND) (N)

Table 8-3 PIUS 600 Major System Dependencies

Uses From -> Feeds to:	100 Building External Plant Struc- tures	200 Reactor & Reactor Auxiliary Equip- ment	300 Reactor Process Systems	400 Steam & Power Con- version	500 Control Equip- ment	600 Electri- cal Equip- ment Systems	700 Service Systems	800 Other Plant Equip- ment
100 Building External Plant Structures	1	0	0	0	0	0	0	0
200 Reactor & Reactor Auxiliary Equipment	0	1	0	0	0	0	0	0
300 Reactor Process Systems	1	0	1	1	1	0	1	0
400 Steam & Power Conver- sion	1	0	1	1	1	1	1	0
500 Control Equipment	0	0	1	1	1	1	1	1
600 Electrical Equipment Systems	0	0	1	1	1	1	1	1
700 Service Systems	1	1	1	1	0	1	1	0
800 Other Plant Equipment	0	0	0	0	0	0	0	1

Figure 8-1 PIUS 600 Major System Dependencies

Active Pool Cooling

323	Rx Pool Cooling System	(NS)	(2,CIVs)
661	Diesel Backed 660 VAC Power	(NS)	(N)
721	Diesel Backed Closed Cooling System for Reactor Pool	(NS)	(3,CIVs)
712	Diesel Backed Service Water System	(NS)	(N)

Passive Pool Cooling

724	Passive Closed Cooling System For Reactor Pool	(SG)	(3)
321	Reactor Pool	(SG)	(2)
714	Cooling Tower	(ND)	(3)

Steam Bypass System

413	Steam Bypass System	(NS)	(N)
592	Turbine Protection System	(ND)	(N)
597	Turbine Control System	(ND)	(N)
441	Turbine Condenser	(ND)	(N)
461	Condensate System	(ND)	(N)

Steamline Isolation

4111	Steamline Isolation Valves	(SG)	(N)
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Containment

No support systems

Emergency Water Makeup

---	Operator Manual Actions	(ND)	(-)
314	Pressure Relief System	(SG)	(2)
316	Condensation System	(SG)	(2)

The containment system does not have any other system that must function to assure containment integrity.

For the above mitigating systems not all of the support systems are clearly designated as safety-related or safety grade.

Further, in order to properly evaluate the appropriateness of the PIUS safety classification system and the assignment of safety classifications to the various front line and support systems, the apparent contradictions and missing information first needs to be corrected. As a summary, all definitions are not clear, not all of the RCPB is class 1, some class 3 systems are listed as non-safety grade, and some safety grade systems have non-safety grade support systems.

9.0 RESULTS

This integrated study of PIUS consists basically of two parts: a component level analysis and an integrated systems analysis. Sections 4 and 6 gave the results from the component level FMECA and deterministic analyses. Sections 5 and 7 gave the integrated HAZOP and accident sequence analyses. The FMECA approach is inductive in the discovery of faults; the HAZOP is deductive. The supporting deterministic analyses use a more conventional approach in the form of thermal-hydraulics, structural, and corrosion analyses. The accident sequence analyses use event trees to study accident progression from event to event until a final equilibrium state is achieved. This section condenses results from sections 4 through 8, and where possible orders items according to significance. This section uses the definitions of frequency and consequence categories from the Table of Definitions in the front, and the definitions of NRC Event Categories from Section 7.3.

The probabilistic and deterministic analyses in Sections 4 through 6 do not provide a complete picture. The FMECA and HAZOP approach the overall picture to some extent, but a more complete representation requires preparation of event trees of accident sequences. This was done in conventional PRA/PSA fashion (Section 7) using estimates of the frequency of initiating events and probabilities of the failure of the mitigation. The accident sequences associated with these initiators were presented in the form of event trees. The end-state core damage of each accident sequence was estimated indicating the amount of material that might be released. The postulated failures are further examined in Section 8 for system dependencies.

The limitations of methodologies used in the integrated PIUS systems study are discussed separately below.

Failure Modes Effects and Criticality Analysis (FMECA)

A major limitation of FMECA is that it assumes that the plant is overall designed correctly, that proof of principle tests have been successful, that the design is properly executed, and that the plant has been started and is operating. These assumptions render FMECA ineffective for detecting most design errors. Some exceptions are presented in Appendix A, such as raising the possibility of exceeding the critical heat flux in certain depressurization accidents.

FMECA focuses on small units of the plant to discover ways in which these could fail, and probable outcomes of failure. Thus, FMECA is very useful as a tool for improving operations and for detecting safety hazards that may result from equipment and human error failures. Its limitations are in the knowledge and insight of the person(s) performing the FMECA, regarding modes of failure and the consequences of failure. This area is where a FMECA may be improved by expert consultation and review.

FMECA's are also limited in their ability to model the results of multiple system failures that may lead to serious consequences. In this study, we used conventional event trees to perform the systems integration.

Hazards and Operability Study (HAZOP)

HAZOP begins by examining the process variables, the tolerance bands within which operation is safe and the effect of perturbations that take the process variables out of range. After identifying critical process variable perturbations, HAZOP determines design errors and equipment malfunctions or failures that could cause the perturbations. All consequences of such errors, malfunctions and failures are then analyzed. These investigations are conducted in a group setting.

HAZOP was developed by the chemical process industry, presumably for plants with minimum documentation. Therefore, the HAZOP teams typically include plant designers to provide details about the design and reasons for the design selection. For PIUS, the primary source of information is the PSID. There were no

members of the PIUS design team participating in the HAZOP part of the integrated PIUS study. As a result many questions were asked that were not answered in the PSID. A few answers were obtained from the vendor; other answers were inferred by PIUS integrated study support analyses (such as PIPA). Nevertheless, some questions remain unanswered, the result of which weakens the HAZOP analysis.

Additionally, the HAZOP study was found to be particularly labor intensive in this case due to some lack of detailed design and operational information. As a result, the HAZOP analyses were only conducted on a sampling basis for PIUS systems and operational modes.

Supporting Deterministic Analyses

Computer models of the thermal hydraulics were constructed using the Plant Interactive Plant Analyzer (PIPA). Various reactor transients were evaluated, primarily in support of the HAZOP analysis. Unfortunately, PIPA was not suited for LOCA analysis and could not support the full range of accidents postulated in Section 7.

Resources were not available for detailed studies of the performance of PIUS when subjected to external events. The reviews concentrated on the safety classifications of systems for withstanding these events. Section 7 casts the seismic problem as an event tree to show possible mechanical-fluid coupling that may damage the core-flow structure under seismic events that are not disruptive to the PCRV, but could possibly lead to core damage.

The material analyses, using conventional engineering, specifically provide many insights into potential equipment failures. Unfortunately, there are little data on operation in the specific conditions that will be experienced in the reactor pool.

Event Tree Construction

Section 7 extends the analyses presented in Sections 4, 5, and 6 through the use of accident sequences diagrammed as event trees. The initiating events were selected from the PIUS Intermediate Table (PIT) as supplemented by plant operating experience and generic initiating events used in LWR Probabilistic Risk Assessments (PRA). Major systems that mitigate the accident, started by the initiating event, were selected from the PIT and from a knowledge of the plant. The initiating events were assigned, based on generic data, into six frequency classifications; the probabilities of mitigating system failure were estimated from generic data and assigned to the same frequency categories. Event trees were constructed for 11 accident sequences and evaluated in the BETA code using the underlying numerical values for frequency and probability. Three categories of accident consequences were assigned to the event sequence end states based on the amount of radioactive material that could be released (see Table of Definitions.) There were no severe accident studies of PIUS to support these assignments but the categories are broad and we believe assignments based on engineering judgment provide insights for the analyses, which are preferable to ignoring the consequences altogether.

Through this process, the ranking of the accidents presented in this section were obtained. These results should not be considered as quantitative. The frequency and consequence categorizations are used to indicate that the results are for guidance. A detailed PRA would be needed for more definitive statements.

NRC Event Category Methodology

We analyzed the same accident scenarios using the NRC Event Category method which is based on initiator frequency and combinations of failures. This method is defined and discussed in Section 7.3. In a simplified fashion EC-1 sequences are Abnormal Operational Occurrences (AOOs), EC-2 sequences are Design Basis Accidents (DBAs), EC-3 sequences are Severe Accidents, and EC-4 sequences are Residual Risk related sequences. Many of the sequences were in the EC-4 category representing sequences beyond or outside of the classification scheme. This method makes no attempt to estimate the consequences of accident sequences. As a result, the

ranking of sequences based on this method, is based on frequency and number/types of failures only; thus they contain a mix of events of varying estimated consequences.

The results of these analyses are summarized below. Prioritized tables of accident sequences will be given, using the NRC Event Categories in Section 9.1 and the Consequence Categories in Section 9.2. The symbols used in the accident sequences, to identify initiators and top events of the event trees are given in Tables 9-1 and 9-2. Prioritized tables of accident sequences are presented in Tables 9-3 to 9-5 (using the conventional Boolean logic symbol of "*" to designate "and".)

Following the prioritized tables of accident sequences in Sections 9.1 to 9.3, systems important to safety are given in Section 9.4. Key operator actions are presented in Section 9.5 and a summary of important general observations is provided in Section 9.6. Section 9.7 lists maintenance, test and inspection observations, while section 9.8 considers the lessons learned in the course of this study. Limitations imposed by lack of engineering detail are summarized in Section 9.9.

Table 9-1 Initiator Identification	
Initiator	Symbol
Loss of Off-Site Power	LOSP
Computer Malfunction	CM
Flow Blockage	FB
Feedwater Transient	FWTR
Small Primary Pipe Break	SPPB
Large Primary Pipe Break	LPPB
Steamline Break Inside Containment	SBIC
Steamline Break Outside Containment	SBOC
Turbine Trip	TTRIP
Severe Seismic	SS

Table 9-2 Top Event Identification	
Passive Scram	PS
Active Scram	AS
Manual Scram	MS
Emergency Water Makeup	EWM
Containment	CNM
Remove Blockage	RB
Exceeding Critical Heat Flux	CHF
Passive Pool Cooling	PPC
Active Pool Cooling	APC
Steamline Isolation	ISO
Steam Generator Tube Rupture	SGTR
Large Primary Pipe Break under Seismic Conditions	1-LPPB
Exceeding Critical Heat Flux under Seismic Conditions	CHF

9.1 Prioritized Tables of Accident Sequences Based on NRC Event Classification

Each of the paths through an event tree shows an accident sequence. Section 7 classified the accident sequences using both consequence categories (CCs) and the NRC Event Category (EC) method. The differences in classification schemes result in a somewhat different prioritization¹. Tables 9-3 to 9-5 present the prioritization of the sequences within each of the three NRC ECs. These tables also contain the information on consequence categories. The prioritization by Consequence Categories will follow in Section 9.2.

Table 9-3 presents the accident sequences leading to EC-3. We note that event category EC-3 is not a classification of the consequences, but relates to the probability of occurrence (see definition in Section 7.3). The Table does, however, also list the consequence category. Using this method, the most probable sequence is Loss of Off-Site Power and failure of the active pool cooling. The next most frequent sequence is a Turbine Trip and Active Pool Cooling failure. An important point to note is that most sequences have no significant consequences (CC-1.) There is only one sequence in the Table of CC-2 and only three sequences of CC-3. The CC-3 sequences all involve a failure of the passive scram. The lack of many sequences with notable consequences is an indication of the overall robustness of the PIUS design.

A similar prioritization may be applied to NRCEC classification leading to EC-2 as shown in Table 9-4. The most frequent four sequences leading to EC-2 are the four initiators on frequency category IV without any additional failures. Again, we note that there is only one sequence with any consequences (and it is CC-2.)

¹ The NRCEC results were prioritized by ordering the sequences according to decreasing frequency based on the numerical sequence calculations performed by the BETA code.

Table 9-5 summarizes the results of accident sequences in EC-1. There is one sequence, namely the Fuel Element Drop, that is postulated to have CC-2. This is worthy of note, since EC-1 is a somewhat frequent condition.

Table 9-3 Decreasing Frequency Ordering of NRCEC Accident Classifications for EC-3		
Frequency Category	Accident Sequence	CC
III	LOSP*APC	1
III	TTRIP*APC	1
III	TTRIP*AS*MS	1
IV	FWTR*APC	1
IV	FWTR*AS	1
IV	LOSP*PS	3
IV	TTRIP*BYPOP*RVSV	2
IV	TTRIP*BYPOP*AS	1
IV	TTRIP*BYPOP*APC	1
IV	TTRIP*BYPOP*MS	1
IV	CM*APC	1
V	CM*PS	3
V	SBIC*APC	1
V	LPPB	1
V	SS	1
V	LPPB*AS	1
V	SBOC*AS	1
V	SPPB*AS*MS	1
V	SBIC*AS*MS	1
V	FWTR*AS*MS	1
V	SS*1-CHF1	1
V	SS*1-LPPB*1-CHF	1
VI	SS*PS	3

Table 9-4 Decreasing Frequency Ordering of NRCEC Accident Classifications for EC-2		
Frequency Category	Accident Sequence	CC
IV	SPPB	1
IV	SBIC	1
IV	FB	1
IV	SBOC	1
IV	SPPB*AS	1
IV	SBIC*ISO	1
IV	FB*CHF	2

Table 9-5 Decreasing Frequency Ordering of NRCEC Accident Classifications for EC-1		
Frequency Category	Accident Sequence	CC
I	LOSP	1
I	TTRIP	1
II	FWTR	1
III	TT*BYPOP	1
III	TT*AS	1
IV	FWTR*AS	1
III	CM	1
III	FD	2
V	CM*(1-RR)	1
V	CM*(1-MSIV)	1

9.2 Prioritized Tables of Accident Sequences Based on Consequence Categories

Further understanding of plant safety is afforded by identifying and prioritizing the accident sequences that lead to the same type of accident consequences. Table 9-6 shows the sequences of accidents leading to CC-3. These sequences are prioritized according to decreasing frequency of occurrence. Also included in the Table is the NRC EC for the sequence. For example, the sequence of first importance in Table 9-6 is a Loss of Off-Site Power (LOSP) followed by a failure of the Passive Scram (PS). The second most important sequence is failure of the Passive Scram, following a Computer Malfunction. Both of these sequences are EC-2, namely an initiator between E-2 and E-4 combined with only a single failure. This illustrates the importance of Passive Scram to PIUS safety. The sequence LOSP*PS is so important, because loss of offsite power occurs relatively frequently and also appears to defeat the active and manual scrams (further explanation of the PSID or design changes may revise this assessment.) However, the PIUS scram system (as analyzed in this study) was not diverse in that the passive scram underlies the scram mechanism whether or not the scram is initiated as an active or manual scram. Thus, if there is a failure in the passive portion of the scram system, PIUS will not scram regardless of how it was initiated. This said, one must still recognize that the passive scram is very reliable.

The consistency of these conclusions will be shown by later Tables that rank initiators and systems, and which rank LOSP second in initiator importance and rank the PS first in system importance.

The list of prioritized accident sequences is different for the less severe accident consequence category, CC-2, than for the most severe category, CC-3. This fact results because only some accident sequences can lead to severe accidents, the remaining, different sequences lead to other consequence classes. Table 9-7 presents a prioritization of the accident sequences that lead to CC-2.

Here, we note that refueling accidents head the list, namely the fuel element drop and exceeding the Critical Heat Flux following Flow Blockage. The next most frequent sequence is the Steamline Break Outside of Containment followed by a failure of the steamline isolation and a steam generator tube rupture.

Another important observation is that taking both Tables 9-6 and 9-7 together only three sequences are above Frequency Category V (E-4 to E-7.)

Table 9-6 Decreasing Frequency Ordering of Accident of Sequences Leading to CC-3		
Event Category	Frequency Category	Accident Sequence
3	IV	LOSP*PS
3	V	CM*PS
4	V	FB*RB*CHF*CNM*EWM
4	VI	LOSP*APC*PPC*EWM*CNM
3	VI	SS*PS
4	VI	SBOC*AS*MS*PS
4	VI	SBIC*AS*MS*PS
4	VI	SPPB*AS*MS*PS
4	VI	CM*APC*PPC*CNM*EWM
4	VI	LPPB*AS*MS*PS
4	VI	SS*1-LPPB*1-CHF*APC*PPC*CNM*EWM
4	VI	SBIC*ISO*APC*PPC*CNM*EWM
4	VI	FWT*AS*MS*PS
4	VI	SBOC*ISO*SGTR*APC*PPC*CNM*EWM
4	VI	CM*1-RR*APC*PPC*CNM*EWM
4	VI	CM*1-MSIV*APC*PPC*CNM*EWM
4	VI	TTRIP*BYPOP*AS*MS*PS*RVSVVI
4	VI	SBOC*AS*ISO*SGTR*APC*PPC*CNM*EWM
4	VI	SBIC*AS*ISO*APC*PPC*CN*EWM
4	VI	SIBC*AS*MS*ISO*APC&PPC*CNM*EWM
4	VI	SBOC*AS*MS*ISO*SGTR*APC*PPC*CNM*EWM
4	VI	SPPB*AS*MS*CHF*APC*PPC*EWM*CNM
4	VI	LPPB*AS*MS*CHF*APC*PPC*EWM*CNM
4	VI	TTRIP*AS*MS*PS*APC*PPC
4	VI	TTRIP*BYPOP*AS*MS*PS*APC*PPC

**Table 9-7 Decreasing Frequency Ordering of Accident of
Sequences Leading to CC-2**

Event Category	Frequency Category	Accident Sequence
1	III	FD
2	IV	FB*CHF
2	V	SBOC*ISO*SGTR
4	V	TTRIP*BYPOP*AS*RVSV
4	V	SBOC*AS*ISO*SGTR
4	V	FWTR*APC*PPC
4	V	SBIC*ISO*CNM
4	V	FB*CHF*CNM
4	V	LPPB*AS*CHF
4	V	SBOC*ISO*SGTR*APC
4	V	TTRIP*BYPOP*AS*MS*RVSV
4	V	FB*RB*CHF*CNM
4	V	CM*1-RR*CNM
4	V	SBOC*AS*MS*ISO*SGTR
4	V	SBIC*ISO*APC*CNM
4	V	LOSP*APC*PPC*CNM
4	V	FWTR*AS*APC*PPC
4	V	SBIC*AS*ISO*CNM
4	V	SS*1-LPPB*CNM
4	VI	FWTR*AS*MS*APC*PPC
4	VI	LPPB*AS*MC*CHF*APC
4	VI	SBIC*AS*MS*ISO*CNM
4	VI	SBOC*AS*ISO*SGTR*APC
4	VI	SBOC*AS*MS*ISO*SGTR*APC
4	VI	CM*APC*PPC*CNM
4	VI	TTRIP*AS*APC*PPC
4	VI	SBIC*AS*ISO*APC*CNM
4	VI	SBIC*AS*MS*ISO*APC*CNM

Table 9-7 Decreasing Frequency Ordering of Accident of Sequences Leading to CC-2 (continued)		
Event Category	Frequency Category	Accident Sequence
4	VI	SS*1-LPPB*1-CHF*APC*PPC*CNM
4	VI	TTRIP*AS*MS*APC*PPC
4	VI	TTRIP*BYPOP*AS*APC*PPC
4	VI	SBOC*ISO*SGTR*APC*PPC*CNM
4	VI	CM*1-MSIV*APC*PPC*CNM
4	VI	CM*1-RR*APC*PPC*CNM
4	VI	SPPB*AS*MS*APC*PPC*CNM
4	VI	TTRIP*BYPOP*AS*MS*APC*PPC
4	VI	LPPB*AS*MS*CHF*APC*PPC
4	VI	SBOC*AS*ISO*SGTR*APC*PPC*CNM
4	VI	SBOC*AS*MS*ISO*SGTR*APC*PPC*CNM

9.3 Prioritized Table of Accident Initiators

Table 7.2A shows the initiators that resulted from the integrated study of PIUS. An additional perspective of the significance of the initiators is gained by prioritizing them by decreasing order of frequency of occurrence as is done in Table 9-8. This table and subsequent tables were prioritized by using numerical values of the frequencies as calculated by the code BETA, although these numbers are suppressed in favor of frequency classes to avoid a misleading impression of precision. Initiators can also be compared by the magnitude or total frequency that the accident sequences associated with them lead to various degrees of plant damage. The Consequence Categories (CCs) are defined in Section 3.3 (also in Table of Definitions.) Table 9-9 orders the initiators according to the frequency that they lead to CC-3 and Table 9-10 according to the frequency that they lead to CC-2. One should note that a different ordering of the initiators is obtained depending on the measure used.

From Table 9-9 it is seen that Loss of Off-Site power is the most important initiator leading to CC-3 with Computer Malfunction next. However, if we consider the next lower category of consequences, namely CC-2, the most important is Turbine Trip followed by Flow Blockage. This ordering change reflects the kind of accidents that the different initiators can produce.

Table 9-8 Order of Initiators by Frequency of Occurrence	
Frequency Category	Initiator
I	Turbine Trip
I	Loss of Off-Site Power
II	Feedwater Transients
III	Computer Malfunction
IV	Steamline Break
IV	Flow Blockage During Refueling
IV	Small Primary Pipe Break
V	Large Primary Pipe Break
V	Severe Seismic Occurrence

Table 9-9 Initiators Ordered by Associated CC-3 Frequency	
Frequency Category	Initiator
IV	Loss of Off-Site Power (LOSP)
V	Computer Malfunction (CM)
V	Flow Blockage (FB)
VI	Feedwater Transient (FWTR)
VI	Small Primary Pipe Break (SPPB)
VI	Large Primary Pipe Break (LPPB)
VI	Steamline Break Inside Containment (SBIC)
VI	Steamline Break Outside Containment (SBOC)
VI	Turbine Trip (TTRIP)
VI	Severe Seismic (SS)

Table 9-10 Initiators Ordered by Associated CC-2 Frequency	
Frequency Class	Initiator
IV	Turbine Trip
IV	Flow Blockage
V	Steamline Break Outside of Containment
V	Feedwater Transients
V	Steamline Break Inside of Containment
V	Large Primary Pipe Break
V	Computer Malfunction
VI	Loss of Off-Site Power
VI	Severe Seismic
VI	Small Primary Pipe Break

9.4 Systems Important to Safety

In Table 9-8 to 9-10, the comparisons were based on the accident initiators themselves. The systems and other top events involved in accident sequences can also be prioritized according to their safety significance. Such prioritization is valuable for resource allocation for design improvement, maintenance, inspection, and other performance improvements to the plant systems. Various importance measures have been devised (see Fullwood and Hall, 1988, for a summary of importance measures). The importance measure used in preparing these tables is non-normalized Fussel-Vesely (FV) importance (Ibid., Table 11.8-1). Physically, FV importance is a measure of the rate of change of risk with respect to variations in the failure probability of the system of interest. Table 9-11 applies FV importance to the sum of plant risks for the systems (and other top events from the ET analysis) identified in the table. If desired to obtain a list of only systems, the other event tree headings (such as CHF) could be dropped, however maintaining these items illustrates the importance of such an assumption or design feature. These systems are ordered by decreasing order of importance. As with initiators, the importance measures are related to the expected consequences of the accident. Table 9-11 shows the system importance leading to CC-3.

Table 9-11 Top Events of ET Analysis Ordered by Importance to Safety (CC-3 Sequences)
Passive Scram (PS)
Active Scram (AS)
Manual Scram (MS)
Emergency Water Makeup (EWM)
Containment (CNM)
Remove Blockage (RB)
Exceeding Critical Heat Flux (CHF)
Passive Pool Cooling (PPC)
Active Pool Cooling (APC)
Steamline Isolation (ISO)
Steam Generator Tube Rupture (SGTR)
Large Primary Pipe Break under Seismic Conditions (1-LPPB)
Exceeding Critical Heat Flux under Seismic Condition (CHF)

9.5 Key Operator Actions

In Section 7, potential accident sequences were identified and event trees were constructed. Various mitigating functions were specified. Many of these could involve some form of dependence on the operator. They could include monitoring or verification of function, and possible direct action to facilitate or restore a mitigation function. In the PIUS PSID, there was no information on possible operator actions pertaining to mitigation or accident management strategies for the plant. Therefore these operator actions were based on PIUS team judgements from the PIUS design, the analyses presented in this report, and plant experiences with current LWR plants.

Table 9-12 is a compilation of postulated operator actions involving mitigation functions for the accident sequences analyzed in Section 7. The acronyms for the mitigation functions and accident types appearing in the table are the same as those used in that section. No time estimates are given in the table for performing the actions, but no tasks were identified that required fast operator response.

One of the first observations is that the operator is not part of the control system, which is why fast (on the order of minutes) operator response is not required. Many of the operator actions are those of verification. In some cases, the action of the operator is that of backing up a failed automatic action - in a sense, adding another redundancy. More serious actions are those of initiation, such as initiation of reactor pool cooling via systems 323/721/724, or controlled venting of the containment. In case of Loss of Off-site Power, the operator initiates load sequencing of the diesels to restore decay heat removal. In case of Turbine Trip, the operators initiate manual operation of the steam dump valves or open the turbine stop valves to the low pressure stages. Also the operators close the main steam isolation valves to initiate blowdown to the suppression pool and increase the pool cooling with system 355. In case of Computer Malfunction, the operators may need to perform manually actions that would

otherwise be controlled by the computer; for example, the initiation of borated water flow to the primary system via system 3123. If blockage of the lower density lock occurs, the blockage must be removed manually, however, the reactor must be disassembled as is done at refueling to gain access to the lower density lock.

The identified operator actions are presented in Table 9-12. Where mitigating functions and/or operator actions are the same as for an earlier event sequence in the Table, they are not repeated. Thus, from Table 9-12 one can see the importance of the operator in many sequences. This tends to contradict the impression left in the PSID that operators and the main control room are not necessary to the PIUS design.

Table 9-12 Operator Response for Performing Mitigation		
Event sequence	Mitigation function	Operator action
LPPB	AS	Verify SCRAM, if not then manual scram.
	MS	Verify SCRAM through main computer, if not, trip primary pump, or trip turbine, or initiate borated water flow to primary system via subsystem 3123.
	PS	Verify SCRAM, if not, borate and remove heat via system 413.
	APC	Initiate reactor pool cooling via system 323 and/or system 721.
	PPC	Initiate reactor pool cooling via system 724.
	EWM	Replenish water in reactor pool using external source via fire system connection.
	CNM	Initiate controlled venting of containment.
SPPB	Same as LPPB	Same as LPPB.
LOSP		Initiate load sequencing of diesels to restore DHR function. Also take actions to restore power.
TTRIP	BYPOP	Initiate manual operation of steam dump valves to condenser or open turbine stop valves and flowpath to LP stages.
	RVSV	Close MSIV initiate blowdown of primary system to suppression pool and maximize pool cooling with system 355.

Table 9-12 Operator Response for Performing Mitigation (Continued)		
Event sequence	Mitigation function	Operator action
SBIC	ISO	Initiate closure of MSIV and/or initiate feedwater reduction.
SBOC	Same as SBIC	Same as SBIC.
Loss of Feedwater		Restore main feedwater or initiate feedwater addition via system 4634.
CM	RR MSIV	Initiate borated water flow to primary system via system 3123. Initiate reopening of MSIV and/or manually trip turbine and open steam dump valves to condenser. Move vital house loads to diesel power supplies.
SS	CNM	Initiate/restore load sequencing of diesels to restore DHR functions. Initiate borated water flow to primary system via system 3123. Verify containment isolation.
FB	RB	Initiate removal of core blockage and/or initiate emergency boration via system 413.

9.6 Summary of Results from FMECA, HAZOP, and Supporting Analyses

The major results from all the previous sections were extracted and then provided to the PIUS team members for review and prioritization. The individual prioritizations of the team members were then arithmetically combined to obtain a final expert opinion ordering, which is presented below in Table 9-13. It must be noted that this summary table presents a condensation of our numerous findings. For more complete information, reference should be made to Sections 4 through 6 for more details and for additional identified items and observations.

Table 9-13 Summary of Major Results from FMECA, HAZOP, and Supporting Deterministic Analyses

1. Lack of diversity in the scram - there is only one way to scram PIUS, that is by activation of the pool loop.*
2. Density lock blockage - a scram reduces the heat removal through the steam generators and activates the pool loop. Core decay heat cannot be removed effectively if pool flow is blocked. Either continuous sensing of the ability of the pool loop to function, if called upon, or the ability to restore flow through the steam generators would mitigate this condition.*
3. Lack of redundancy in the active scram - a scram is performed by removing power from a dedicated pump. No redundancy, such as usually found in LWR scram systems, is indicated.*
4. Power distribution - in the absence of control rods, corrections to flatten the power distribution or to compensate for xenon oscillations do not seem possible.
5. Boron stratification could result in low-borated water entering with the resulting reactivity transient.
6. Common-cause computer failure - PIUS is dependent on digital computer control and information processing. The thyristor variable speed control of the main cooling pumps produces electromagnetic interference, which could common-cause fail the computers.*
7. Failure to maintain adequate boration in the reactor pool - the concern is that the pool boration system will deborate the pool. Protection against this is provided by monitoring the boron in the pool.
8. Seismic collapse of the flow structure - through structure-flow motion coupling may be severe enough to damage the flow structure and to inhibit core cooling.
9. Benchmarking of thermal hydraulic codes against integral test results is needed.
9. Classification of systems mixes safety with non-safety systems.
11. During refueling, the incore instrumentation is removed; no explanation is provided with regard to the approach to criticality.
12. Loss of heat insulation may result in entrained material that could cause loop-flow blockage.
13. Foreign objects that may enter during refueling or maintenance, or loose parts may cause blockage of pool loop flow.
14. Failure of the reactor support structure would drive the gas cap down on the lower density lock and block coolant flow.
15. A number of seismic design issues were identified, as summarized in Section 6.4
16. Materials issues were identified in the following areas: stainless steel pool liner, gas cover lock, wet thermal insulation, reactor internals, and boric acid leakages.
17. Status of Control Room (CCR) - Discussions in PSID seem to indicate low importance of CCR and operators. CCR is designated non-safety grade.
18. Tendon inspection - tendons lose tension in service. If a sufficient number loose tension, a catastrophic PCRV failure is possible. Tendon inspection appears to be difficult.
19. Gas-lock blockage - under degraded core conditions, gas may accumulate in the cap and block loop flow.

Table 9-13 Summary of Major Results from FMECA, HAZOP, and Supporting Deterministic Analyses

20. Accessibility for refueling - refueling PIUS requires flow structure disassembly and grappling fuel from 47 m (154 feet), most of which in water. The boundary fuel elements appear to not be accessible on a straight line.
21. Thermal transients - when PIUS scrams, the rate of temperature change is about 20°C/min. (>2000°F/hr). These will occur several times per year and may lead to cyclic fatigue.*
22. Dynamic loads on systems during blowdown could result in failed internal structures that could contribute to core flow or density lock blockage.
23. Multiple failure scenario was identified with the potential for large power oscillations and some fuel damage. Failures consisted of: reactivity control system failure to inject pure water, inadvertent startup of idle distilled water pump, scram failure, and feedwater trip.
24. The containment isolation valves are shown as motor operated. Such valves close fairly slowly and would release more radioactivity than if they were air-operated valves.
25. No analyses for high winds, tornados, or floods performed to date.

* Comments received from ABB-Atom during the review of the draft report affect these results, generally by providing additional information about the existing design or about a revised design since the issuance of the PSID. This new information may change the observation or lower its ranking. This additional information is discussed in Section 4.

Generally, PIUS embodies many passive safety features, as might be expected of an advanced reactor. Some additional detail to our table entries will be given here:

1) The scram system is not truly diverse. In this analysis, we subdivided the scram into Active, Manual and Passive. Strictly speaking, however, these are ways of initiating the scram and not diverse ways to scram. Regardless of how it is initiated, a scram of PIUS is the same process namely, activation of the pool loop. Activation of the pool loop is a passive process; however, it is not clear that the probability of this feature is less than the probability of two diverse active scram systems failing for current LWRs. (This observation has been mitigated somewhat by the recent PIUS design modification, adding an active scram system.)

2) Lack of redundancy in active scram results from the use of a dedicated scram pump with no mention of redundancy in the system for removing power from this pump. (This observation has been mitigated somewhat by the recent PIUS design modification, adding an active scram system.)

3) Should passive scram fail, there is no way to remove the decay heat except by boiling of the pool. Activating the loop flow through the steam generators, if possible, would be highly desirable under such conditions.

4) Deboration of the reactor pool is hard to envision, since a large amount of boron must be removed. But, if such did happen, the effects could be catastrophic.

5) Concerning incipient failure detection of the pool loop (e.g., low boron concentration or stratification), there does not appear to be a way, to continuously monitor the operating status of the pool loop for both scrambling and for decay heat removal. If such a method were available, the probability presented here for failure of the passive scram and passive heat removal would be considerably reduced.

6) The extensive use of digital computer control and monitoring might raise questions, partly on the basis of lack of operating experience. Failure of the control and monitoring system may be a particular concern in PIUS because of the electromagnetic interference (EMI) produced by the thyristor variable speed pump controllers. The EMI was not discussed in the PSID. Presumably, there will be EMI suppression systems; if they were to fail, the control and monitoring system might fail. (Supplemental information by ABB on the design to avoid EMI mitigates this observation somewhat.)

7) Lack of diversity in the PIUS control and monitoring systems. Digital control and monitoring is used throughout; any common cause failure eliminates the control and status monitoring. Some use of analog control and monitoring would provide diversity; as an example, manual scram now takes place through the main computer rather than by hardwiring the scram to the dedicated scram pump.

8) Refueling of PIUS appears to be difficult. It requires partial disassembly of the flow structure, grappling for fuel from 47 m (154 feet), most of which distance is in water. This is a time at which objects could enter the primary system and block flow.

9) There appear to be inconsistencies in the safety-grade system classification. We find that safety-grade systems are supported by non-safety grade systems (see Appendix E), and vice-versa. (This could be partly a documentation problem in the PSID.) This creates problems in the establishment of normal operating conditions, the initial conditions from which transients begin, and in the actual mitigation of transients and accidents.

10) With the lack of control rods, the operator has no means of compensating for xenon oscillations or flux tilts during normal power operation.

11) The status of the central control room as non-safety related and the treatment of control room operators as essentially unnecessary for transient response is a deviation from current LWR practice.

9.7 Maintenance, Test, and Inspection Observation

Test and maintenance are activities that are important to all reactors and which when not implemented properly have caused problems in the operation of LWRs. The same is potentially true of PIUS. In this section we give the test and maintenance observations that arose during of the integrated PIUS study.

9.7.1 Maintenance Observations

Maintenance functions in nuclear power plants can affect plant availability, radiation exposure (ALARA), and overall plant safety. For this design, the flow diagrams and building layout drawings were reviewed to determine potential maintenance concerns. The review was limited because PIUS is in a preliminary design. Much of the equipment location and sizing is not reflected on the drawings, which is to be expected at this stage of the design phase. However, the review generated some observations that may warrant closer scrutiny.

1. On the drawings, the main reactor building crane appears to operate in one direction only. To support the laydown of the reactor dome, it would also need to move in other directions.
2. A review of the flow diagram showed an absence of check valves in many fluid systems. Maintenance on MOV could be time consuming without isolation capability. Large sections of system piping may have to be drained for work to be performed.
3. Clearance for Main Coolant Pump and motor removal appears inadequate on the drawings. Also, work on the pump will require the respective steam generator to be drained. Pipe sizing to facilitate draining should be done to minimize the drain down time.

4. Any work to be done involving the tendon anchor flange bolts appears difficult to perform because they are not accessible.
5. Since this reactor has to be disassembled and reassembled for refueling, there is concern about the following:
 - a) Disconnecting and reconnecting the primary coolant pipes from the nozzles.
 - b) Disconnecting and reconnecting instrumentation needed for power operations.
 - c) Removal and replacement of the outermost fuel assemblies in the core. It does not appear that these fuel assemblies are reachable in a straight line from above due to reactor skirt geometry.
 - d) Replacement of damaged wet insulation which could occur during disassembly or reassembly of the reactor.
 - e) Sufficient clearance to disconnect the various cylindrical sections of the reactor.
 - f) Rigging requirements as well as lifting lug placement on various sections of the reactor.

9.7.2 Test and Inspection Observations²

In Section 7, certain key mitigation functions were identified. Those functions could become, or form the basis for, the Technical Specification test requirements for PIUS. As such, the systems that support those mitigation functions either directly or indirectly would be included in the testing or Inservice Inspection (ISI) Program.

Section 8 contains the PIUS systems that support each mitigation function identified in Chapter 7. For selected systems deemed to be important to test, Table 9-14 was constructed. The type and frequency of testing noted in Table 9-14 is based on tests performed in current operating LWRs as well as engineering judgement. In particular, column 3, Frequency, presents typical current inspection practice and is not a suggestion for the PIUS reactor. For cases where one train of a system is running during power operations, it is assumed that the alternate train would be tested.

²The PIUS PSID indicates that inservice inspection (ISI) information will be given later. The information contained in this section that arose during the study presents concerns which may be addressed in the PIUS ISI program, when submitted.

Table 9-14 Table of Inservice Inspection Observations			
PIUS System	Type of Test	Typical Frequency	Mitigation Function Supported
Batteries - 673	Verify operability	Weekly	All
Diesels - 651	Start and load	Weekly	AS, MS, PPC
Diesel starting air - 751	Bleed moisture from receivers	Every operator shift	AS, MS, PPC
Compressed air system - 753	Bleed moisture from air receivers	Every operator shift	ISO
Primary pressure safety relief valves - 314	Operability and calibration	Every refueling outage	AS, MS, PS, RVSV
Steam bypass valves - 413	Stroke and hydro	Every refueling outage	BYPOP
Steam line safety relief valve - 411	Operability and calibration	Every refueling outage	AS, MS, PS
Diesel blocked ESW - 712	Functional	Quarterly	AS, MS, APC
Fuel pool cooling - 324	Functional	Quarterly	RB
Containment pool cooling - 355	Functional	Quarterly	CNM
Reactor pool cooling - 721	Functional	Quarterly	APC
Reactor pool line cooling - 722	Functional	Quarterly	APC, PPC
Passive pool cooling - 724	Functional	Quarterly	PPC
Reactor pool water replacement - 325	Functional	Quarterly	APC, PPC
Borated water supply - 3123	Functional	Quarterly	AS, MS
Distilled water supply - 3122	Functional	Quarterly	PS

Table 9-14 Table of Inservice Inspection Observations (Continued)			
PIUS System	Type of Test	Typical Frequency	Mitigation Function Supported
Filtered water supply - 3121	Functional	Quarterly	PS
Seismic monitoring system - 584	Calibration	Monthly	ISO, CNM
SGTR monitoring system - 554	Calibration	Monthly	ISO
Pipe rupture monitoring system - 546	Calibration	Monthly	ISO
Containment atmosphere monitoring system - 583	Calibration	Monthly	ISO, CNM

9.8 Lessons-Learned in the Study and Considerations for Tests

The work plan that was developed did not originally envision the need for event tree analysis. The need for event tree analysis was realized about half-way through the study. This resulted in some inefficient changes to the work plan.

The question arises as to whether an integrated study as conducted is necessary. The FMECA was valuable for the condensed design descriptions and to aid in plant familiarization in addition to identifying potential failure concerns, such as those in Section 4. HAZOP provided the valuable contribution of viewing the plant from a process perspective which contributed to the completeness of the study. The HAZOP would have been improved if the needs for supporting analyses could have been identified earlier and have been available at the time of the HAZOP. The supporting analyses were necessary to assure the correctness and provide information for the accident analyses. The study would have been improved if severe accident analyses had been available rather than using judgment on the outcome of the accident sequences.

Considerations For PIUS Testing and Modelling

Based on the results of studies given in this report, we believe that the following would be fruitful areas for further testing and/or improved modelling of the PIUS conceptual design :

1. The effectiveness of a partially blocked density lock in providing adequate flow from the reactor pool to shut down the plant during accident conditions.
2. Natural circulation flow rates from the reactor pool through the core region for decay heat removal.
3. Pool cooling with the passive heat removal system.
4. The density lock water interface control with the three level temperature measuring system that is included in the current design.

5. Core cooling to prevent fuel damage during simulated depressurization transients. The following transients would be of interest:

Primary system pipe break
Pressurizer steam space break
Break in the steam system piping
Inadvertent relief valve opening
Steam generator tube rupture
Siphon breaker failure injecting cold water into steam space
6. The potential for the accumulation of unborated water at various locations in the primary system (e.g., the riser funnel region).
7. Shaker-table testing of a scale model of PIUS with the pool water represented to investigate the fluid-structure interactions and the possibility of resonance.
8. The possibility of level oscillations in the siphon breaker region that could cause a loss of the hot/cold interface in the density locks.
9. The structural integrity of the density locks for anticipated thermal transients.
10. The noise production from the thyristor pump controllers and the electrical power noise threshold for the plant computers.
11. The capability of the wet insulation to maintain its structural integrity for design transients and over the life of the plant.

9.9 Limitations Imposed by Lack of Engineering Detail

The PSID gives a general overview of PIUS-600, but many details are lacking; some examples follow:

- General lack of control information needed for thermal-hydraulics modeling.
- Tolerance band on the system controlling the interface in the lower density lock, i.e. what are the ranges of process upset before scram takes place.
- Conditions under which the critical heat flux will be exceeded.
- Conditions that could lead to boron and/or thermal stratification in the pool or in the primary loop.
- Criteria for determining a failed digital control system.
- How software QA will be performed.
- The effects of thermal cycling on reactor components.
- Mechanisms for removing boron from the reactor pool.
- How electromagnetic interference from the thyristor speed control of the main pump will be prevented from affecting the control and display systems.

- Conditions under which pressure oscillations may take place.
- Methods of returning condensed steam from condensation pool to reactor pool (see Section 7.3).

References

R. Fullwood and R. Hall, "Probabilistic Risk Assessment in the Nuclear Power Industry," Pergamon Press, Oxford, 1988.

APPENDIX A

SYSTEMS DESCRIPTIONS AND FMECA ANALYSES

- A.1 Pressurizer Systems 211 and 328
- A.2 Reactor Internals, Reactor Fuel and Instruments Systems: 213, 215, 261, 216, 217, 291
- A.3 Reactivity Control System 312
- A.4 Primary Recirculation System (System 313)
- A.5 Pressure Relief and Condensation Systems
- A.6 Containment
- A.7 Control and Monitoring Systems
- A.8 Plant and Turbine Control Systems
- A.9 Electric Power
- A.10 Cooling

APPENDIX A.1 - PRESSURIZER SYSTEMS 211 AND 328

Following the PIUS Preliminary Safety Information Document (PSID) the FMECA analysis begins at the top of the reactor - at the pressure dome. In proceeding through the system listing to perform the FMECA, one will note some overlap in components and subsystems, thus to avoid covering items twice, not every numerical system is covered per se. For example, systems 321-325 are not listed separately since their components were addressed elsewhere. This appendix section presents the dome which forms the upper pressure boundary and closure to the prestressed reactor vessel (PCR.V). It is by removal of this upper structure that access is gained to the interior of the PCR.V and to the reactor proper. Tendons in the concrete serve the function of head bolts in a conventional LWR to hold the head on the reactor under the operating pressure. The section also discusses the pressurizer that receives steam from an external boiler to control the pressure inside of the PCR.V. The pressurizer is rather large because one of its functions is to accept thermal expansion water from the PCR.V.

A.1.1 Structural Aspects of the Upper Pressure Vessel: System 211

System 211 is shown in Figure A.1-1 consists of:

2111 The anchor flange with its underlying platform and connections to its hold down tendons

2113 The cylindrical vessel steel extension with piping connections, upper flange and support brackets

2115 The vessel steel head with the pressurizer steam and water volumes, vessel head internals, and connections for the pressure relief system (314), and some auxiliary systems.

Anchor flange (2111)

The anchor flange consists of a forged and machined ring with a truncated triangular cross-section. It is made of pressure vessel steel, SA 533 B, and clad with stainless steel on its inside. The flange design is shown as Figure A.1-2.

Anchor flange (2111)

The anchor flange is a forged, triangular-cross section, ring of pressure vessel steel with stainless steel on its inside welded to the lower end of the cylindrical vessel steel extension (2113), as shown in Figure A.1-1. The underside of the anchor flange rests on a specially designed, ring-shaped welded and machined platform, that is cast into the concrete at the top of the prestressed concrete vessel. The platform has a width of 1.5 m, and its total contact area with the concrete is about 30 m², which limits the contact pressure on the adjoining concrete to acceptable values. The base of the platform has a thickness of 100 mm with holes for the hold down tendons. On its inside it is ring-shaped with a thickness of 0.2- 0.3m, extending up to a height of about 0.75 m. This inner portion acts as part of the vessel pressure boundary and supports the cavity liner which on its inside is facing the reactor pool water. (The cavity liner is not bonded to the platform, just supported by it.)

Outside this ring, the platform is provided with 130 mm broad fins with a spacing of about 450 mm, for installation of the hold down tendons. The upper side of these fins and of the inner ring is machined and serves as the support surface for the anchor flange proper.

The hold down tendons for the anchor flange penetrate the outer portion of the platform, through the holes between the fins. They are fastened to the underside of the anchor flange by means of transition pieces. One end of these pieces is threaded into holes in the flange and the other around the tendon head. By rotating the transition piece, both ends are disengaged, and the tendon can then be withdrawn. The transition piece can be removed radially outwards.

Figure A.1-1 Reactor Upper Structure

3 of 6

The tendons are basically of the same type as the ordinary prestressing tendons of the concrete vessel proper, with an ultimate stress capacity of at least 10 MN^1 (2.2 Mlb). The hold down tendons are run down to the bottom of the concrete vessel monolith and anchored at their lower ends.

The anchor flange and the platform are cooled by water from the diesel-backed closed cooling system for certain components and pools 722) via a large atmospheric pool at a higher elevation. During normal plant operation the cooling water flows through drilled holes in the flange and platform in forced circulation by means of the pumps in system 722, which have a power supply from the non-safety-grade diesel generators. In a loss of power situation the forced circulation flow stops, but the flange and platform cooling continues by natural circulation to the pool - at reduced capacity.

Vessel Steel extension (2113)

The cylindrical vessel steel extension (Figure A.1-1) has an inner diameter of 6.1 m, and a wall thickness of about 185 mm. It is made from forged rings of ASME SA 508 class 3 material with a stainless steel overlay welded on the inside.

The upper flange is a conventional pressure vessel flange with metallic O-ring seals as used on other reactor pressure vessels. About 3 m below the seal, the four main coolant outlet and four main coolant inlet nozzles for the recirculation system 313) are welded to the cylindrical vessel. The two inlet lines for filtered and distilled water from the reactivity control system 312) are connected at a somewhat higher level.

Just below the main coolant nozzles a number of brackets are welded to the inside of the steel cylinder. These brackets support a ring which in turn serves as support for the uppermost flow guide structure 213). A support skirt on the flow guide structure is pressed against the ring by means of hold down spring beams in the vessel head.

About two meters below the main coolant nozzles there are two DN 250 (250 mm nominal diameter) nozzles for the outlet pipes to the forced circulation reactor pool cooling system 323) coolers.

The pressurizer is supplied with steam from an electrically heated recirculation boiler 328). The connections to this boiler are made via pipe nozzles which are located about 1.5 m above the main coolant nozzles.

Vessel steel head 2115)

The inner diameter of the cylindrical vessel is 5.6 m, and its wall thickness is about 175 mm. The total height of the head is about 6.5 m.

The head contains a riser funnel 2138) and the riser funnel hold down springbeams. When the head is installed on top of the vessel steel extension, the springs will press against the bottom flange of the riser funnel and press it towards its sealing surface against the uppermost flow guide structure 213). This in turn is then pressed against the ring on the support consoles on the vessel steel extension and thus kept in place by means of the spring beams.

Considerations

During normal full power operation, the lower part of system 211 is at a relatively low temperature up to a point just below the support consoles for the uppermost flow guide structure. Then, a vertical temperature gradient extends up to the level of the support consoles. Above the level of the consoles, the temperature is near that of the

¹ The interior diameter of the dome is about 4.34 m dia. giving an area of 14.79 m^2 . At an operating pressure of 9 MPa, the upward force is 133.1 MN.

outgoing riser water (290°C).

After a reactor coolant pump trip, the water temperature outside the thermal insulation in the lower part of the steel extension increases by about 200°C. The water will then be cooled down by the reactor pool cooling System 323). During this period, the anchor flange is thermally protected by the thermal insulation and kept cool by the cooling system inside the flange.

A.1.2 Pressurizer: System 328

System 328 normally controls and maintains reactor vessel pressure and absorbs coolant level fluctuations due to temperature variations in the primary system. During a plant transient it limits pressure variations in the vessel by acting as a manifold for the pressure relief system 314), and provides the third and top connection between the reactor primary system and the reactor pool. During certain abnormal transients, it receives steam formed in the primary system.

The pressurizer vessel (Figure A.1-1) consists of the vessel steel head 2117) of the pressure vessel and the riser funnel in the center of the vessel. The riser funnel cylinder is hydraulically part of the primary system and the opening at the top connects it with the steam volume of the pressurizer vessel compartment.

The pressurizer has two compartments:

- 1) The upper pressurizer vessel compartment has a diameter of 5.6 meters and a height of 6.5 meters. The upper part of the compartment is filled with saturated steam at 9.0 MPa, and the normal volume of this space is about 100 m³ which includes the upper part of the central riser funnel pipe.
- 2) The lower part of the compartment, called the pressurizer pool, is filled with water to a height of about 3.0 meters with volume of about 50 m³ and kept at the saturation temperature by the pressurizer heating system 3283). The pool level must rise to 4.6 meters before water spills over into the riser funnel top opening.

The pressurizer offgassing system 3285) protects against build-up of explosive oxy-hydrogen gas.

Instruments are provided to monitor: pressure, temperature, and liquid level.

Considerations

Under stable plant operation, the pressurizer conditions are stable with little heating required to maintain pressure. Under changing conditions, the primary system volume will vary leading to reactivity control system 312) corrections and the reactor pool water replacement system 325) by adjusting the injection and withdrawal rates. If these adjustments are not sufficient, the pressure control system 538) will order increased pressurizer heater input, or initiation of the relief valves in the pressure relief system 314).

In certain transients, such as a hypothetical loss of feedwater combined with failure to scram, the primary system volume will expand to such an extent that primary coolant will flow over the rim of the riser funnel (4.6 m) and down to the pressurizer pool. To the extent that subcooled water is involved, steam condensation will take place, thus reducing the pressure increase resulting from the primary system volume expansion.

A.1.3 Pressurizer Heater

System 3283 heats the water in the pressurizer pool and produces steam in order to control the pressure in the reactor vessel. It uses an external electrically heated steam boiler inside the reactor containment. It is supplied with water from the pressurizer pool by means of one of two redundant feed pumps with the pump flow rate regulated by a control valve to keep the water level in the boiler constant. Steam is extracted from the steam dome

of the boiler and led back into the pressurizer water pool via a control valve.

To avoid dilution of the boric acid concentration in the boiler water, about 10% of the feedwater flow is continuously recirculated from the boiler to the pressurizer pool by means of a small pipe and a flow regulating orifice. The boiler is heated by 8 groups of 350 kW electrical heaters (2.8 MW). All heaters are used for startup but only one for normal operation.

Each flow line is provided with a motor-actuated shut off valve outside the pressurizer and check valves to avoid flow reversals during plant transients. The boiler is provided with redundant pressure and level measuring, a safety valve and a nitrogen supply pipe to provide an inert environment in the boiler before startup.

The computerized control system builds up and controls the pressurizer pressure at 9 MPa (1305 psi). The boiler pressure is limited to and kept at 1 MPa above the pressurizer pressure. Signals are taken from the pressurizer pressure and the boiler pressure and level measuring devices. These are used to control the feedwater flow to the boiler, the steam flow to the pressurizer, and the power input to the electrical heaters in such a way that the pressure in the pressurizer can be kept constant or varied by the computer program.

Considerations

Before startup, compressed nitrogen is introduced into the boiler to establish a nominal water level in the boiler.

One of the feed pumps is started and the recirculation flow to the pressurizer is established. The control valve in the return line is closed and the pressure rises from steam production while the water level is maintained constant until the boiler pressure reaches 1.1 MPa when, the control valve begins to release steam to the return line maintaining 1 MPa pressure difference between the boiler and the pressurizer until the pressurizer pressure reaches 9 MPa. When the pressure reaches 9 MPa, the steam flow rate is automatically reduced to about 110% and the large return line downstream of the control valve is automatically closed after which, steam is supplied by the small line.

If one heater group fails, a standby unit is automatically activated. If the operating feed pump fails, the standby pump is automatically started. If both pumps stop, the water level in the boiler will drop because of backflow to the reactor via the recirculation line. A signal of low water level in the boiler will automatically disconnect the heaters.

A.1.4 Pressurizer offgassing system (3285)

Radiolysis will produce gaseous hydrogen and oxygen which must be removed from the pressurizer vessel compartment to prevent explosive concentrations.

The system consists of valves, coolers and level monitors. It is idle until the water chemistry control system (533) or laboratory personnel decide that the concentration of permanent gases has become too high (>0.4 MPa) in the pressurizer dome compartment. Then two vent valves are opened to draw off 9.0 MPa/hr dome steam and condense it leaving the non-condensable gases which are cooled and dried for release by their offgas system (341).

A.1.5 FMECA

Tables A.1-1 present the FMECA evaluations of systems 211 and 328 based on the availability of material at the time of preparation.

FMECA Notes for Appendix A-1

a) The lack of a gas-filled accumulator tank means that a LOCA will result in a fairly rapid depressurization of PIUS. However, the fuel power density is low and critical heat flux may not be exceeded during the depressurization transient. Thermal-hydraulic calculations are needed to address this possibility. Eventually core dryout may occur but such will require many days assuming PCR V integrity.

Table A.1-1 PIUS 600 Failure Modes Effects and Criticality Analysis

System: Upper Vessel: 211 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: September 10, 1991								
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
1	Head retention tendon/ retain upper vessel structure against operating pressure.	fracture	corrosion, material defect electrolysis, fatigue,	LOCA - rapid depressurization may exceed critical heat and damage fuel	Low	V	There are multiple tendons holding the head on.	<ul style="list-style-type: none"> • Inspection of tendons of the PCRV not described. Inspection could be by tendon removal but difficult and time-consuming. • PIUS has no accumulator for suppression of depressurization transient. See note a.
2a	Head nuts for tendons /retain upper vessel structure against operating pressure.	fracture	material defect, over-stress	see above	Low	V	see above	Nuts would be inspected at each refueling.
2b	Head bolts/ retain pressure dome	fracture	see above	see above	Low	V	see above	Bolts would be inspected at each refueling.
3	pressure dome/ primary pressure envelope	fracture	material defect fatigue,	see above	Low	V	see above	Head inspection would be similar to conventional reactor vessel inspection if not easier because of lower radiation.
4	steam feed pipes /deliver steam to pressurize the reactor	pipe break	material defect fatigue,	small LOCA - plant shutdown	Low	IV	none identified	<ul style="list-style-type: none"> • Steam pipes are inspectable. • It is believed that the depressurization would be too slow to damage fuel.

Table A.1-1 PIUS 600 Failure Modes Effects and Criticality Analysis

System: Pressurizer: 328 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: September 10, 1991								
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
5	pressure relief valves/ relieve excess pressure	failure to open	corrosion, binding, bad adjustment	failure to relieve pressure could lead to dome failure and large LOCA	Low	III	other valves could relieve the pressure	A systematic mis-calibration error could be a common cause leading to over pressure.
6	pressure relief valves/ relieve excess pressure	failure to close	corrosion, binding, poor maintenance	same as a small LOCA	Low	IV	none	The conditions and blowdown rate are not expected to result in fuel damage
7	water drain pipe/ drain water to control boron concentration	pipe break	see item 4	small LOCA	Low	IV	none identified	water pipes should be inspectable
8	level sensor/ provide indicators to the steam supply	sensor fails	wiring, electronics, calibration	inability to determine and control water level in pressurizer. Likely forced shutdown.	Low	II	none	continued operation may result in filling pressurizer dome to form a "hard" system. If this is followed by a fluid expansion, the pressure could be damaging.
9	temperature sensor/ monitor steam temperature	sensor fails	see item 4	reduced knowledge about pressurizer	Low	II	none	

Table A.1-1 PIUS 600 Failure Modes Effects and Criticality Analysis

System: Pressurizer 328 heater and off gas systems Reference: PIUS PSID

Preparer: R. Fullwood, BNL

Date: September 10, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
10	pressure sensor/ monitor reactor pressure	sensor fails	see above	inability to control pressurizer steam flow and regulate reactor pressure	Low	II	none	on the basis of the information, it is not known whether the pressure would rise or fall in the event of a sensor failure.
11	heaters/heat water to form steam for pressurizer	open circuit	corrosion, fatigue, chemical action	loss of one heater for steam generation	Low	II	switch to one other of the 8 heaters	
12	water for boiler/ water to form steam	failure of supply	control failure, supply failure	350kW heater could overheat and fail boiler vessel. Blow- down through steam supply line.	Low	III	temperature monitors	if the boiler envelope fails from this cause, the blowdown would be through the steam line and not very fast.
13	offgas release valve / to vent non-con- densible gases to the offgassing system (3285)	valve fails to operate or be operated	human error, valve actuator failure, valve failure, control failure, sensor failure	explosion in pressure dome causing large LOCA and fuel damage	Low	V	continuously vent gas from the pressure dome and measure its composition.	continuous monitoring the explosive gas ratio is safer than batch sampling based on human actions and decisions.

APPENDIX A.2 - REACTOR INTERNALS, REACTOR FUEL AND INSTRUMENTS

Systems: 213, 215, 261, 216, 217, 291

This appendix section describes the reactor core, support and flow structures as shown in Figure A.2-1. This figure shows the metal dome, whose structural aspects were discussed, and the volume contained within the PCR.V. Contained within this are the 4 flow structures, all of which must be removed for refueling, the pool passive heat exchangers, the connections to the steam generators and the core.

A.2.1 Reactor Internals System 213

System 213 supports the reactor core and provides structures concerned with coolant flow to the reactor. The PIUS core is cooled by a parallel connection of two coolant loops: one through the steam generator and one through the reactor pool. Which flow path provides core cooling in a given situation, depends on balances between the thermal pumping head due to density differences and the pumping head provided by the primary pumps. System 213 contains all of the components for directing the flow to the loop that includes the reactor pool and some of the components used for directing the flow through the steam generators loop. These alternative flow paths are unique to PIUS.

Another unique PIUS feature is the reactor riser that:

- During normal operation, forms part of the equipment needed to develop the thermal pumping head that is balanced with the mechanical pumping head to prevent reactor pool entry into this core-steam generator loop thereby transferring heat to the steam generator.
- During cold shutdown, forms part of the equipment needed to develop the thermal pumping head to circulate reactor pool water through the core and as such transfer core heat to the reactor pool.
- In transient situations, performs the cold shutdown function which brings borated water into the core and shutdown the reactivity.
- In abnormal situations, conducts steam to the pressurizer region.

Detailed Descriptions

System 213 will be discussed in terms of the following constituent systems:

- 2131 Core envelope structure
- 2132 Core bottom and side support structures
- 2133 Reactor inlet plenum components and the lower density lock
- 2134 Flow guide structure No 1 - upper support - with core upper support
- 2135 Flow guide structure No 2 - with flow shift - over arrangement and upper density lock
- 2136 Flow guide structure No 3
- 2137 Flow guide structure No 4 - with siphon breaker and main coolant nozzle connections
- 2138 Riser funnel

System 2131 Core envelope structure,

System 2131 supports the core, core bottom, side support structures (System 2132), provides reactor inlet plenum components (System 2133), the lower density lock, encloses the recirculation loop (313) downcomer around the core and seals to the lowest flow guide structure (System 2134) Figure A.2-1 presents the scope of the system.

Figure A.2-1. PIUS Reactor Assembly

The core envelope structure is partly double-walled, cylindrical, made of 3 cm thick 304 stainless steel. The annular volume between the two walls, above the flow crossover carries the heated coolant to the steam generator. Below the crossover, coolant is returning from the steam generator to enter the core above the gas lock.

The inner and outer walls are connected together by 12 radially oriented steel sheets of about 3 m height, welded along their edges to the walls. These sheets strengthen the structure and eliminates the risk of collapse of the inner wall in the event of overpressure in the downcomer annulus (relative to the riser), e.g., due to a large break in the upper steel portion of the reactor pressure vessel.

System 2132 Core bottom and side support structures

System 2132 which supports the core vertically and horizontally, consists of the core bottom support plate and the core baffle. This system is depicted in Figure A.2-2. The core bottom support plate, (A) is made of 304 stainless steel with a diameter of 4.2 m and thickness of 400 mm. It is supported by the bottom flange (B) and the inner wall of the core envelope structure - system 2131).

The 213 fuel assemblies of the reactor core (D) stand on the core bottom support plate. Square holes are provided under each fuel assembly position. On each corner of each square opening there is a guide pin which fixes the position of each fuel assembly.

To protect the core support plate against thermal stresses from inflow of cold pool water when the pool loop is activated following an inadvertent tripping of the main coolant pumps (System 313), a thermal sleeve structure is placed in each square opening. This sleeve consists of a square plate, with a central hole to which a short vertical core support hole liner pipe is welded. The length of the liner pipe corresponds to the thickness of the core support plate, and protects it from direct contact with cold water from the reactor pool. Surrounding the core is the core baffle (G).

System 2133 Reactor inlet plenum components

System 2133 consists of the reactor inlet plenum compartment, the pool water inlet baffle and the pool water inlet pipes (density locks) which are bolted together.

The reactor inlet plenum compartment is divided into two by the pool water inlet baffle which forms an inner compartment. The outer compartment for the flow from the steam generator loop is separated from the other flow path by a lower funnel shaped bottom piece which extends from the bottom flange on the core envelope structure (System 2131).

The natural circulation loop inlet plenum encloses the startup gas lock baffle (H) and is connected to the lower density lock (C) with its buffer volume, thermal insulation and piping, located in the central hole of the bottom piece. The system can be unbolted from the core envelope structure and removed to the reactor components storage pool for inspection as a single unit after unloading of the core and removal of the core bottom and side support structures (System 2132) leaving only a core envelope structure. If necessary the different parts of the system can be unbolted from each other in the storage pool.

The upper wall of the pool water inlet baffle is conically shaped and penetrated by holes with the pool water inlet pipes - one for each fuel assembly. The conical shape implies that the available height of the flow path for the return flow to the core decreases successively towards the center to correct for the flow pattern from the pool and assure uniform flow through the core.

Figure A.2-2 Core bottom and Side Support Structures

The pool water inlet pipes are welded at their bottom end to the conical top of the pool water inlet plenum. Welded to their bottom is an elongation that permits water entry but prevents collapse of the inner compartment in case of hypothetical high outer overpressure which could block the natural circulation path through the core.

Thermal stress and fatigue from the rapid temperature change from pool water entry are minimized by three concentric vertical flow baffles welded to the underside of the conical baffle. The water, in the enclosed volume, remains hot during the transients. Also, thermal sleeves are used to avoid thermal weld stresses.

The bottom of the natural circulation loop plenum is shaped to minimize its volume and thermal stresses. It is in contact with but not bolted to the upper side of the flange by which the lower density lock assembly is bolted to the funnel-shaped bottom piece.

The density lock proper (C) consists of a bundle of 55 mm i.d., 1 m long stainless steel pipes welded to a bottom "tube sheet", welded to a stainless steel cylinder and flange for bolting to the funnel shaped bottom piece. Above this tube bundle there is a 10 cm free space above which there is another pipe bundle (I) that serves as a buffer volume to avoid transient-caused boron ingress from an upward surge of the hot/cold interface in the lower density lock. There are a number of small diameter pipes leading from the interspace between the two pipe bundles to the volume between the core baffle and the inner cylindrical wall of the core envelope structure. During plant operation there is a small flow in these pipes bypassing the core to keep the upper tube bundle at the primary loop coolant core inlet temperature and composition. In case of a sudden temporary surge of pool water into the upper pipe bundle, a small but insignificant amount of pool water will enter the primary system through these pipes.

An extreme accident might evaporate sufficient pool water to concentrate boric acid in the core and riser. Before saturation, the increased density will result in discharging this concentrated solution through these pipes and density lock into the bottom of the PCR.V.

The riser consists of the following parts counted from its lower end upward:

- a large diameter plenum (F) above the upper core support plate through which core instrument lead tubes enter and a cylindrical section (J) where flow velocity is increased,
- a transition part (Figure A.2-3 (K)) where the riser flow is shifted from the center inside the downcomer to the periphery,
- three concentric pipes passage which ends below the siphon breaker part,
- uppermost section containing siphon breaker and main coolant nozzles.

The reason for transferring the riser flow from the center to the periphery is to maximize the recovery of kinetic energy of the incoming flow from the cold leg main coolant nozzles, to minimize the potential for flow induced vibrations and to provide a central location for the siphon breaker. This design features long diffusers from the nozzles to the downcomer. Moreover, the cold leg nozzle diameter is reduced to provide a flow constriction to avoid a large underpressure in the downcomer in case of pipe rupture.

Figure A.2-3 Upper Reactor Riser

System 2134 Flow guide structure No. 1 with the core upper support.

There are four flow guide structures for coolant transport from the cold leg coolant pipe inlets to the upper downcomer section of the reactor envelope. These flow guide structures are removed for refueling separately, to a storage pool. Each of the structures has thermal insulation.

No. 1 is the lowest flow guide structure. It is 9.85 m long and consists of two concentric, partly cylindrical, partly conical structures. The flange at the lower end is bolted to the core envelope structure and three similar flanges must be separated for refueling. A typical flange of this type is shown in Figure A.2-4. Notice the thermal insulation shown as corrugation.

The core top support structure is welded to the inner wall and is part of flow guide structure No. 1 which consists of: upper core support plate, upper support ring, and instrument guide tubes. The upper core support plate, with its 213 square openings, engages the top end plates of the fuel assemblies. It is welded to the upper support ring.

Figure A.2-4 Typical Envelope Flange

System 2135 Flow guide structure No. 2 with flow shiftover arrangement and upper density lock.

Just above the flange joining structures 1 and 2, in the 8.4 m guide structure, is the flow shift-over (Figure A.2-3 (K)).

Above and outside the crossover is the insulated, upper density lock (M) which is an annular array of many, 55 mm inner diameter, 1 m high, vertical pipes welded to an annular "tube sheet" which is welded to the outside wall of the riser.

Above the upper density lock is a buffer volume (N) to contain the expansion of the primary system under a transient in which the interface in the lower density lock is held constant. Its volume is large to contain the expansion from a step load increase of 10 % of full power without pool water ingress.

A suction pipe continuously draws water from an annular slot inside the density lock pipe bundle and transports it to the external reactor water cleanup system to prevent the buildup of borated water that could undesirably enter the primary system when the power is increased.

System 2136 Flow guide structure No. 3.

Flow guide structure No. 3 (Figure A.2-3 (L)) consists mostly of three concentric cylindrical pipes with inner diameters of 1.65, 2.6 and 3.2 m respectively, 8.15 m long. The largest pipe has a conical expansion at its lower end and a flange that seals against its counterpart in flow guide structure no 2. The innermost pipe is the downcomer. Its upper end mates, in a sliding joint, with its counterpart in flow guide structure No. 4.

The annular volume between the innermost and middle pipes is part of the riser for the annularly flowing water to the steam generator. The middle pipe mates similarly to the inner pipe with its counterpart in flow guide structure No. 4. The volume between the middle and the outermost pipe connects the upper density lock with the upper end of the riser and is part of the loop through the reactor pool.

In case of trip of the main coolant pumps, the ensuing natural circulation, driven by the core decay heat, will use both paths. Analysis has shown that the flow through the upper main path is large enough so that all low boron primary system water in it and in the external loops is displaced by pool water.

If the water level in the concrete vessel should fall below the upper end of the riser, so that the main natural circulation flow path is broken, the flow through the by-pass holes in the middle pipe will be sufficient to allow the passive removal of all core decay heat to the reactor pool. The passive reactor pool cooling system ensures efficient cooling of the pool and there will be no boiling even at atmospheric pressure.

At the outer pipe, upper end of flow guide structure No. 3 is a bellows to absorb up to 150 mm thermal expansion of the central stainless steel column relative to the cold concrete vessel when the column is heated to operating temperature. In order to avoid risk of stress corrosion cracking the bellows is placed in the cold pool water and is only exposed to hot water on reactor scram or pump trip.

The upper end of the bellows is provided with a flange that connects with the lower end of flow guide structure No. 4. Flow guide structures 1 through 3 are solidly connected to the core envelope and via the latter to the liner near the concrete vessel cavity bottom; flow guide structure no 4 is mounted on the pressure vessel steel extension.

System 2137 Flow guide structure No. 4 with siphon breaker and main coolant nozzle connections (2137)

Flow guide structure No. 4 is shown in Figure A.2-5:

- Constitutes the uppermost part of the riser
- Contains the uppermost part of the downcomer sending cold leg nozzle water downward
- Houses the uppermost part of the natural circulation path from the upper end of the riser down to the upper density lock
 - Houses the siphon breaker
 - Provides with connections to and seals the eight main coolant nozzles
 - Houses the inlet arrangement for filtered and boron free water from the reactivity control system
 - Provides a path for back flow of water through the external loops in case of trip of all the main coolant pumps.

The four cold leg inlet nozzles (M) in the pressure vessel steel extension are connected to the central downcomer plenum via long diffusers (N), the purpose of which is to recover flow kinetic energy and minimize flow induced vibration due to the 90° downward turn of the > 13000 kg/s (206,000 gpm) primary coolant flow. Each of these

diffusers divide into a part that is left in the vessel and a part that belongs to guide structure 4 and is removed when refueling. Figure A.2-6 shows the arrangement that is used. By means of special tools, the springs and the bellows are compressed to disengage the sealing surfaces so guide structure 4 can be removed.

As shown in Figure A.2-5, the flow from the four cold leg nozzles travels the diffusers to the centrally located upper end of the downcomer and then turns 90° downwards. It then enters an annular section with small cross-section area where the velocity is increased, hence the static pressure is reduced to about that of the riser at this level. A long expanding diffuser (O) reduces the velocity before the coolant enters flow guide structure No. 3.

In the center of the high velocity annulus there is a straight 640 mm inner diameter pipe which is open to the pressurizer steam volume. The pipe ends (open) in the high velocity section but 140 mm below it there is another concentric pipe with the same diameter.

During normal operation there is no flow in the upper of the two central pipes. (There is never any flow in the lower pipe.) The static pressure in the high velocity section at the pipe opening will determine the water level in it.

Figure A.2-5 Upper Riser Structure

Figure A.2-6 Expansion Bellows

In case of a main coolant cold leg pipe break leading to extensive loss of reactor pool water inventory, the level in the pool and riser may fall considerably below the main piping nozzles. However, steam flowing from the pressurizer through the central pipe will break the siphon and stop the outflow, leaving enough water inside the concrete vessel cavity to ensure the presence of the natural circulation coolant path through the core which is cooled by external heat exchangers.

Holes, with a total square section of 0.05 m^2 penetrate the wall between the riser and downcomer, at the siphon breaker elevation. These holes also help break a postulated siphon flow but their main purpose is to direct riser water to the main coolant cold leg following a trip of all main coolant pumps. This is also the purpose of the flow baffles in front of the entrance to the hot leg pipes that force the normal outgoing flow to a level that cannot be reached by the cold water during natural circulation. Hence cold water enters the drilled holes just mentioned and natural circulation through the external loops occurs in reverse.

Figure A.2-5 shows the arrangement used for adding pure water to the primary system for normal cleanup and for startup and power increase (part of system 312, reactivity control system). Water enters through two pipes with their nozzles between the main coolant hot leg nozzles (Q). From there it is sent to a balcony (R) overflow arrangement. Because the incoming water is colder than the primary water, it tends to move to the bottom of the balcony. During plant operation it flows from there to the entrance of the hot leg nozzles through a suction pipe to each nozzle; the suction being provided by the underpressure caused by the acceleration of the flow entering the nozzles.

If the flow of boron reactivity control system is accidentally started when the main coolant pumps are idle, cold water will pass over the overflow weir and flow down to the upper density lock and into the large pool. There is suction that can transport the cold water up to the hot leg nozzles under these conditions.

Thus, criticality accidents cannot occur as a result of local pockets of pure water from improper operation of the pure water inlet system with idle main coolant pumps.

System 2138 Riser funnel

The riser funnel is the uppermost of the reactor internals and constitutes the connection between the primary system (riser upper end) and the pressurizer steam volume.

The riser funnel consists of a 6.1 m high, 2.0 m i.d. cylinder (Figure A.2-5 item (S)) welded to a lower 0.6 m high truncated cone (T) with an outside diameter of 4.4 m that seals to a flange at the upper end of the support skirt of flow guide No. 4. During operation elastic beams inside the pressure vessel head press against this flange, providing the sealing force for it and the force that keeps flow guide structure No. 4 pressed against its support brackets on the vessel steel extension.

The riser funnel houses the pipe leading from the pressurizer steam volume to the siphon breaker opening in its center. During normal operation there is no flow in the riser funnel. The level in the funnel will be about 4.4 m above that in the surrounding pool due to different average temperatures hence different densities in the two water columns measured from the lower end of the upper density lock.

Transients, e.g., loss of feedwater without scram will result in an overflow of water to the pool over the upper rim of the riser funnel. If the overflowing water is below saturation temperature some steam condensation will occur. In depressurization transients large two-phase flows will temporarily occur in the riser funnel.

Normal plant operation

During normal plant operation at full reactor power, the reactor outlet water temperature is $\sim 290^{\circ}\text{C}$, and the inlet temperature is $\sim 260^{\circ}\text{C}$. The pressure drop in the inlet throttles (that are part of the fuel assemblies), across the fuel assemblies themselves and the core support structures, as well as in the reactor riser pipe amounts to about 0.05 MPa (7.3 psi) at full reactor power when the core coolant flow rate is 13000 kg/s.

The difference in density between the hot riser water and the colder reactor pool water results in a pressure differential of about 0.05 MPa (7.3 psi) between the lower and upper density locks inside the reactor primary circuit up the riser pipe. This buoyant riser force is the driving force for the full power core coolant flow rate of 13000 kg/s 206,000 gpm. As long as the main coolant pumps (System 313) supply the reactor with a matching water flow rate (which is determined by the pressure differential between the density locks), the pool loop is inactive.

Plant startup

At the beginning of plant startup from a cold shutdown, the temperature in the reactor riser is too low for stable operation of the reactor pool loop. The gas lock (Figure A.2-2, item (V), System 3225) in the lower density lock (System 3221) is filled with a gas bubble preventing inlet flow from the reactor pool until the reactor outlet water temperature has risen to $\sim 100^{\circ}\text{C}$ corresponding to a flow rate of $\sim 20\%$ of full speed of the main coolant pumps (about 100 C).

The plant is normally shutdown by injection of boric acid into the core-heat exchanger loop. Alternatively tripping one of the four main coolant pumps by the trip and interlock system (System 516) will upset the flow balance and cause pool water to enter the core via the lower density lock and scram the reactor. If this is done, the inlet plenum and the pool water inlet pipes will be exposed to a rapid water temperature change from $\sim 260^{\circ}\text{C}$ to $\sim 50^{\circ}\text{C}$ which is ameliorated as described. If all four main coolant pumps trip, will the forced circulation inlet plenum be stressed by being flooded with cold reactor pool water?

Refueling

At refueling, the riser funnel and the top three flow guide structures (System 2134 to 2137) are lifted out of the reactor vessel (the concrete vessel cavity and the vessel steel extension) and placed in a separate storage pool.

Abnormal situations

If the main coolant pumps stop, all the water brought to the top of the reactor riser will not be removed and all the water needed in the reactor inlet plenum will not be provided which will activate the reactor pool loop transferring the residual heat to the reactor pool (System 321).

Certain plant transients will raise the reactor water temperature to boiling which will increase the buoyancy in the riser, upsetting the flow balance and scram the reactor.

Under loss of heat sink accidents, evaporation of the reactor pool water removes the reactor residual heat. If this is prolonged, loss of water will cause the recirculation in the reactor pool loop to cease and boiling will occur. The reactor riser could buckle ("the beer can syndrome") and stop the circulation of the reactor water in both reactor loops. The flow guide structures are designed to prevent buckling.

In a guillotine break in the cold leg of one main coolant pipe, the extremely large break flow will subject the forced-flow inlet to an underpressure transient. For this reason, a small diameter inlet nozzle is included in the cold leg and the components are designed accordingly.

System 215 Reactor Assembly Supports and Restraints

The reactor assembly as defined in this report mainly consists of the thin-walled flow guide structures that direct the coolant flow to and from the reactor core and the core support structures. System 215 supports the reactor assembly and is shown in Figure A.2-2. It consists of system 2151, reactor assembly support and system 2153 reactor assembly seismic restraint.

System 2151 consists of a support skirt - a truncated cone of 30.4 cm thick stainless steel. At the lower end it is welded to cavity liner which is increased in thickness to about 30 mm in this region. The cone has a number of holes for pool water flow. These holes are < 5.5 cm to prevent passage of material in the coolant that could clog the lower density lock tubes. Blockage of a large number of holes is not likely. At the upper rim is a flange for bolting the core envelope structure (system 2131).

System 2153, if found necessary will consist of a number of spokes arranged at the upper end of flow guide structure 2 (system 2135) to restrict motion of the flow guide structure under seismic conditions.

A.2.2 Fuel Assemblies (System 261)

Assembly design

The PIUS fuel assembly is shown in Figure A.2-7. It closely resembles a standard present day PWR fuel assembly except for the shorter length (2.8 m). The assembly is built up of an 18 x 18 square array of rods. The corner rods, and four of the inner rods in an assembly consisting of thick-walled zircaloy tubes and zircaloy spacer grids, to form the assembly skeleton. The other 316 positions are used for fuel rods identical to standard PWR rods except for the length. The four corner rods (tubes) are welded to the four zircaloy spacer grids. In this way protruding corners of the grids that could engage with grids of adjacent fuel assemblies during fuel handling are eliminated. The four zircaloy tubes in the interior of the bundle are welded to the grids and may be used for locating in-core neutron detectors.

The eight, unfueled tubes are closed in their lower end except for a small hole that passes enough water to prevent boiling.

The 316 fuel rods are placed with their lower ends supported by the assembly bottom plate. The bottom and top end plates are connected by the eight thick-walled zircaloy tubes. The upper end plate can be removed after picking out inserts placed in the ends of these tubes. Several rods in each fuel assembly are provided with machined elongations of their upper end plugs. The pellets in these rods contain gadolinia burnable absorbers. After the first irradiation cycle, these rods can be removed (all at the same time) by means of a special tool and replaced by new gadolinia containing fuel rods. These new rods then remain in the assembly for the rest of the assembly life.

Fuel rods

The fuel rod cladding is a Zircaloy tube with an external diameter of 9.50 mm which is closed with welded end plugs. The fission gas plenum length is about 100 mm and the total fuel rod length is 2.625 m containing uranium dioxide pellets having a diameter at room temperature of 8.20 mm.

Other assembly components

Egg-crate spacer grids made of Inconel X-750 are mechanically fixed to the lower assembly end plate. The other grids are welded to the corner tubes. The assembly bottom end plate vertically supports the fuel rods and the assembly. When the assembly is placed on the core support grid, engage two guide pins on the support grid to define the position of the assembly. The assembly upper end plate consists of a grid of beams holding the sides together. The latter are elongated with holes drilled in the elongation to engage with the fuel grapple in fuel handling. The upper end plate is fixed to the upper end of the eight thick-walled zircaloy tubes which carry the weight of the assembly during handling.

The top end plate is not designed to prevent upward movement of fuel rods because of PWR experience where the lifting force due to the coolant flow is an order of magnitude higher than the PIUS design. Table A.2-1 summarizes the assembly data.

Table A.2-1 Fuel Assembly Data	
Fuel pellet diameter (mm)	8.20
Cladding outer diameter (mm)	9.50
Nominal diametrical assembly clearance (mm)	0.16
Uranium dioxide weight/assembly (kg)	428.6
Total assembly weight (kg)	546.7
Fuel rod total length (m)	2.625
Fuel column length (m)	2.500
Number of fuel rods	316
Number of unfueled (water) positions	8
Number of inconel grids	1
Number of zircalloy grids	4
Assembly total height (m)	2.820
Assembly square width (m)	0.2267

Operating situations

Some normal operation pertinent data of PIUS fuel rods as compared to that of a modern PWR (Doel 4) are shown in Table A.2-2.

Load increases in zircaloy clad UO_2 fuel rods is known to cause cladding cracking in current LWR cores due to pellet clad interaction (PCI) under certain conditions. However, this occurs only below about 30 kW/m linear heat load which is well above PIUS (11.9 kW/m). During a main coolant pump trip, the core rapidly shuts down with the coolant temperature dropping 200°C. This could lead to stretching (with plastic deformation) of cooled zircaloy cladding over a hot and uncontracted pellet. However, ABB-Atom calculations show that the pellet contracts fast enough so that the gap actually increases during the cooldown transient.

Table A.2-2 Comparason of PIUS and PWR Fuel Rod Operating Data		
Criterion	PIUS	Doel 4
Average fuel rod linear heat load (kw/m)	11.9	16.5
Average cladding heat flux (kw/m ²)	399	567
Average coolant temperature (°C)	275	313
Core water flow velocity (m/s)	2.8	4.6
Core pressure drop (kPa)	36	167
Core power density (kW/l)	72	96.5

A.2.3 Vessel Internal Instruments Supports - System 216

Subsystems in system 216 are:

- 2161 Support equipment for the neutron flux monitoring system (531) detectors
- 2162 Support equipment for the reactor outlet and inlet water temperature monitoring system (5361)
- 2163 Support equipment for the fuel assembly out-let water temperature monitoring system (5363)
- 2164 Support equipment for the reactor pool temperature monitoring system (5364)
- 2165 Support equipment for the temperature monitoring system in the lower density lock (System 3221) 2166 Support equipment for the temperature monitoring system in the upper density lock (System 3222) 2167 Support equipment for the concrete vessel temperature monitoring system (5366)
- 2168 Support equipment for the primary loop flow rate monitoring system (5367)

Herein are described only subsystems 2161 and 2163. These subsystems (Figure A.2-8) consist of guide tubes that lead from a location inside or directly above the core to a special upward elongation of the drywell where the instrument cables pass out through the primary system pressure boundary in the seal table. The 2161 guide tubes for the incore neutron detectors are divided into five parts. The upper ends of the two lower parts are funnel-shaped to ease the insertion of the instrument probes from above. The lowermost part is a 10 mm i.d. zircaloy instrument guide tube that is an integral part of the fuel assembly skeleton. There are four such tubes where neutron detectors can be inserted in each fuel assembly (in addition there are the four thickwalled corner tubes). The next section mounted in the 2134 lowermost flow guide structure, is fixed to the upper core support plate and ends directly above the corresponding zircaloy guide tube in a fuel assembly. From there it passes up through the support plate and into the lowermost part of the riser. It is then bent outwards and passes through the lower part of the inner wall of the lowermost flow guide structure to pass through the downcomer annulus between the 2134 flow guide structure and the upper part of the outer wall of the reactor envelope structure (2131) and end with their funnel shaped ends immediately adjacent to the reactor envelope structure. The sections of the guide tubes passing through the riser are held together by lateral beams to minimize flow induced vibrations. In their passage across the higher velocity flow in the downcomer annulus, the guide tubes now coming together in four bundles) are protected by an upstream flow deflector mounted on the flow guide structure. The next section of the guide tubes (in bundles) have their ends welded to flanges bolted to the outside of the reactor envelope structure so that each penetration in the flange directly faces a guide tube in the section below. The bundles then pass up through the pool inside an envelope pipe in order to prevent movements due to flow in pool water. Their upper ends are welded to another multi-tube flange that is located immediately outside in the pool side under the liner insulation. The guide tube bundle(s) continue from this flange via the liner through a stainless steel pipe cast into the concrete. At the upper face of the concrete this pipe ends with an inward flange. To this flange is bolted another multi-tube flange for the guide tubes going through the stainless steel pipe.

The pressure inside the guide tube and in the outer envelope pipe is about the same (of the order of 9 MPa) but the guide tube contains low boron primary system water whereas the outer tube contains high boron pool water. This is the reason for all the sealing arrangements with the other, internal flanges. Another multitube flange faces the pressure boundary flange just mentioned. In this is primary system pressure that continues upward with air on the outside with missile impact protection. This uppermost part of the guide tubes ends at the seal table, where the electrical instrument leads pass out of primary system region. The seal table is located in a special upward extension of the drywell at a level high enough to allow opening of the seals and removal of instrument probes while the reactor is in the refueling stage. Thus the 216 instrument guide tubes

Figure A.2-8. Internal Instrument Supports

consist of five successive sections:

- in the fuel assemblies,
- in the lowermost flow guide structure,
- in the pool, in a pipe through the concrete and finally
- in air up to the seal table.

Their total length is about 40 m. In the three uppermost sections they are handled together in four bundles. If necessary the various sections can be removed from the concrete vessel. This is done routinely in refueling for the two lowermost sections. Removal of the other three sections is not expected to be necessary but if it is, the section inside the concrete is removed downwards. The 2163 guide tubes are identical to those for system 2161 except that they end in the upper core support plate. The 216 guide tubes make a 180° turn at the bottom of the cavity before entering the lower density locks; their passage through the pool and concrete vessel is the same as for the 2161 guide tubes. The other guide tubes for the upper density lock will probably be similar.

A.2.4 Startup neutron sources 217

This non-safety grade system has the task of providing initial neutron flux during plant startup. The system consists of four neutron sources per reactor core, each incorporated in a fuel rod (System 261). The sources are located fairly close to the SRM detectors. During the first reactor cycle, a neutron source consists of about 100 µg of Cf-252, encapsulated in double welded cans of Zircaloy. During later cycles antimony and beryllium (Sb-Be) sources supply the startup flux. The system also encompasses a shielded container for the transport of the Cf source rods. The Cf sources have a half-life of 2.6 years. As they decay, the activation of the Sb-Be sources becomes saturated and these become the neutron sources. The Cf sources are permanently installed into the fuel assemblies. The 217 sources are left in the fuel in the spent fuel storage racks (System 243).

A.2.5 Wet thermal insulation - Reactor primary system 291

The PIUS Metalisol insulation system limits heat losses from the reactor primary loop to the reactor pool while maintaining acceptable conditions from corrosion and reactor chemistry is supplied by Compagnie de Ateliers et Forges de la Loire of France. It has been used in nuclear reactor gaseous atmospheres in France and large scale tested in water by Studsvik Energiteknik in Sweden. The basic element of the Metalisol insulation is a "mattress" consisting of a pile of screens, woven from stainless steel wire mesh and held together by stainless steel clips. Each mattress is reinforced along its upper edge by a stainless steel U-profile metal. The Metalisol insulation system consists of a number of mattresses separated by thin stainless steel sheets and protected by stainless steel cover plates. They are displaced with respect to each other in such a way as to minimize edge effect losses. The whole system package is held together by stainless steel rivets.

For PIUS, the Metalisol type insulation has been selected, which was tested in water at Studsvik. This system is characterized by 0.2 mm wire diameter, a pitch of 0.55 mm with 20 screens per mattress having a thickness of 5 mm separated by 0.2 mm sheets with a cover plate of 5 mm thickness for a total thickness of 25 mm. The Studsvik tests were performed at temperatures which are close to PIUS temperatures and can thus be used directly without any resort to recalculations. The losses were on the average 10 kW/m² on vertical surfaces.

During plant operation and hot standby, the expected losses of heat through the insulation will be around 15 MW. The wire may be subjected to general corrosion and to some extent to local pitting corrosion. Experience with wire mesh mattresses in demisters shows that some stress corrosion may also occur due to stresses left in the wires from the weaving operation. Wires may break in the insulation but not to the extent that system integrity is threatened. Water will be essentially stagnant inside the insulation system, and the system is not expected to affect the reactor pool chemistry or corrosion product concentration to any important degree.

A.2.6 Failure Modes Effects and Criticality Analysis

Tables A.2-3 provide failure modes, effects and criticality analyses for the equipment discussed in this section.

FMECA Notes for Appendix A-2

a. Blocking the lower density lock has many ramifications regarding safely shutting the reactor down. First it would inhibit scram because the poisoned pool water would be denied entry. Shutdown would have to be achieved by boron injection into the primary loop flow.

Forced flow with the primary pumps is possible with the lower density lock blocked. Such a flow mechanism would have to be maintained until the residual heat drops to the point that no flow through the fuel is necessary.

b. See note a as well as the following comments: Details on the control of gas and its release from the gas lock have not been found in the PSID. It is understood that the gas cap is designed to "leak" a continuous outflow into the primary coolant to presumably collect in the pressurizer. The gas lock is actuated by flowing gas in faster than it leaks out. If the opening(s) through which the gas leaks become blocked, the gas cannot be released hence the pool loop cannot actuate. A high leak rate would make such an obstruction highly improbable with the penalty of a heavy gas load to be removed from the coolant.

c. See notes a and b as well as: Under a severe accident, gas may be evolving from the overheated core. It would rise vertically and collect in the pressurizer but gas will be entrained and returned to the core by the primary cooling stream. Gas entry into the gas cap (see Figure A.2-2) is inhibited by the fact that the primary flow is injected above the flow equalizer grid (V) and it is difficult to see how the gas could migrate downward to get under the cap and be trapped.

Nevertheless it would be desirable to have no way that pool flow could be blocked short of physically blocking the flow path.

d. If this were to occur, note a. would also be applicable as well as: Blockage in the tubes carrying pool water into the core entry at the core support plate due to the accumulation of boron, insulation debris or whatever, would seem to occur slowly and with time. It is apt to be an asymmetric effect that is detectable by temperature sensing if it is done at different radial and azimuthal positions (Section A.2.3 is not sufficiently specific). General indications may be indicated by comparing the lower density lock water temperature with the core inlet and outlet temperatures when operating under pool cooling.

e. The scenario that is being considered is: The reactor is operating under full power. The inner annulus flow guide structure below the crossover, cracks, letting a considerable amount of primary flow to bypass the core and be returned to the steam generator. This flow would add to the thermal-induced core flow such that pool flow may not be actuated. The core will overheat with local steam formation. The pressure increase will result in flow into the pressurizer. The core outlet coolant temperature is sensed to cause a scram.

f. Under degraded core conditions, debris may be carried by the flow stream upward to be deposited on the upper entrance to the upper density lock. This would cutoff flow to the degraded core and result in further degradation.

g. The PIUS core, by necessity, has low flow impedance. For this reason, it is believed that changes in local flow impedances will have more effect on the flow distribution throughout the core than would be the case with a higher impedance core.

Table A.2-3 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Reactor Internals: 213 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: December 16, 1991								
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
1	Lower density lock/admit pool water, provide stagnant pool-reactor water interface	geometry change blocking flow	loose part/object, unrecognized damage during maintenance/refueling, earthquake, corrosion, metal fatigue, boron accumulation, radiation embrittlement	Prevent scram, prevent pool cooling - no emergency cooling - difficulties in shutting down reactor.	Low - see note a.	IV	Loose part monitors, inspection, water chemistry	Considering the diameter and wall thicknesses of the tubes comprising the lower density lock, this event seems highly improbable
2	Gas lock/prevent pool water entry during startup	Gas release/accumulation rate exceeding gas release rate	Foreign material, loose material blocking gas release holes, boron accumulation in holes, failure of gas injection control.	same as above	Medium - see note b	III	Monitor on gas flow to gas lock, inspection	There is no direct way of sensing the status of the gas lock.

Table A.2-3 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Reactor Internals: 213 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: December 16, 1991								
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
2	(continued)	Gas production rate exceeding gas leak rate	Core degradation and overheating	same as above	Low - see note c	IV	None	A positive means of preventing blockage of the lower density lock is desirable. A better alternative is to eliminate the gas cap. This would require that the primary pump flow be controllable to zero flow.
3	Tubes supplying pool water from the flow equalizer plate to the core support plate/ inject pool water into the core for emergency and shutdown cooling	Blockage	Boron accumulation, foreign material such as insulation debris	same as above	Medium - see note d	III	Inspection, water quality control	A mechanism for direct sensing core cooling flow patterns under pool and loop flow conditions would be desirable.
4	Core support/ supports the reactor core	Fracture	Vibration, metal fatigue, flaw in manufacture, radiation embrittlement, seismic	core drop, cooling channel blockage, core geometry change	Low	V	Design margins, inspection, vibration detection, metal samples for damage detection	Such an accident has only occurred as the result of overheating and core damage. Pius operates with low power density therefore no strenuous requirements on core support design. No anticipated possibility of recriticality from core geometry change.

Table A.2-3 PIUS 600 Failure Modes and Effects and Criticality Analysis								
System: Reactor Internals: 213			Reference: PIUS PSID		Preparer: R. Fullwood, BNL		Date: December 16, 1991	
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
5	Lower inner core flow annulus/ guide primary flow to core inlet	fracture	flow vibration, fatigue, manufacturing flaw	bypass primary flow from core cooling	L	IV	Inspection of flow structures, vibration monitoring	See note e
6	Upper density lock/ return flow for pool flow	Flow blockage	Weld break, material fracture, thermal, vibration fatigue	Failure to scram, inability to secure adequate shutdown cooling	M	IV	Inspection, vibration monitoring	Blocking the upper density lock has the same effect on pool flow as blocking the lower density lock, including inhibiting. See notes a and b.
			Debris from degraded core	See above	M	V	None	See note f

Table A.2-3 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Fuel Assemblies: 261

Reference: PIUS PSID

Preparer: R. Fullwood, BNL

Date: December 18, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
1	Fuel elements/ provides power for the reactor	damage of cladding	Thermal shock - differential expansion, fuel clad interactions, radiation embrittlement, seismic	Contamination of primary water	Low	II	Design fuel for rapid cooling	When the reactor is scrammed, the coolant temperature drops very rapidly as the pool water enters the core region.
2	Core cooling chan- nels/passage of coo- ling water to remove core heat	partial or full blockage	accumulation of boron or other foreign matter, bowing, fuel swelling, seismic	local fuel over- heating possibly propagating to other rods. See note e.	M	IV	Design, quality control, inspection, rod support and constraint	This effect is not expected to be significant in less time than a fuel change. Incipient effects would be detected at that time.

Table A.2-3 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Vessel Internal Instrumentation 216

Reference: PIUS PSID

Preparer: R. Fullwood, BNL

Date: December 18, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
1	Core instrumentation/monitor core and coolant status	Instrument failure	Sensor failure, Instrumentation failure	Reduction in information regarding operating conditions	Low	II	Redundant, multiple instrumentation	Single instrument loss is expected, even if all core instrumentation were lost, scram is not inhibited. However, the requirements of RG 1.97 regarding severe accident monitoring may not be implemented.
2	Instrument mount	Break loose	Fatigue, vibration,	Loose part in system	Low	III	Loose part detection	

APPENDIX A.3 - REACTIVITY CONTROL SYSTEM 312

A.3.1 Discussion

PWRs control reactivity by varying the boron concentration in the coolant and by control rods. PIUS does all of the reactivity control by varying the boron concentration, either by boron addition or dilution of the coolant or by causing the highly borated reactor pool water to enter the core by upsetting the flow balance by a pump trip.

System Purposes

- Start the reactor from hot or cold conditions by diluting the boron content in the primary coolant with clean water
 - Control the power level and compensate for the gradual burn-up of the fuel during normal operation
 - Shutdown the reactor to hot or cold conditions
 - Supply the primary loop with cleaned makeup water and with chemicals for water chemistry control •
- Control the hot/cold water interface level in the upper density lock

System 312 Composition

The non safety grade, System 312 is made 304 stainless steel, except for some components in the injection pumps, the pressure pulse compensators, and some valves. It interfaces with the following subsystems which will be described before the system:

- 3121 Filtered water supply system
- 3122 Distilled water supply system
- 3123 Borated water supply system
- 3124 Inlet-outlet system
- 3125 Normal withdrawal system
- 3126 Temporary discharge system

System 3121 Filtered water supply

System 3121 supplies the reactor primary system with clean water having the boron content required for the current operating mode. In addition it: replaces the water withdrawn by the cleanup system 3125, adjusts the concentration of lithium hydroxide and hydrogen in the primary systems, controls the hot/cold water interface level in the upper density lock.

Figure A.3-1 consists of:

- Two redundant 100% reciprocating pumps (A, P11 and P12). Each pump delivers 8.5 kg/s (135 gpm) using a 90 kW motor supplied from system 643. They take water from the reactor water cleanup system (System 3311) and deliver it to the inlet-outlet system (3124) at about 9.2 MPa.
- Pump service valves (B) and a check valve (C) through which lithium hydroxide (LiOH) solution and hydrogen (H2) flow to the pumps.
- A pressure pulse compensator (D, T11)
- A flow meter (E, K311)

Figure A.3-1. Filtered Water Supply System

The operating pump is controlled by the reactor power control system (535). The supply to the primary system matches the varying demands in the reactor primary system.

System 3122 Distilled water supply system

System 3122 supplies water to the primary system with less than 10 ppm boron to: start the reactor by diluting the primary coolant with clean water, control the power level, compensate for the fuel burn-up.

System 3122 shown in Figure A.3-2 is identical in configuration with that shown in Figure A.3-1.

Figure A.3-2 Distilled Water Supply System

The system consists of the following main parts:

- Two reciprocating pumps (P21 and P22) rated at 30 kg/s (476 gpm) drawing 350 kW from system 642. They take suction from the distilled water tank (331 T22) in the reactor water cleanup system (system 3312) or LiOH and H₂ and deliver it to the inlet-outlet system (3124) at about 9.2 MPa
- Pump service valves and a check valve to system 3312.
- A flow meter (K321)
- A pressure pulse compensator (T21) after the pumps.

The pumps are 100% redundant for power operation of the plant but are used in parallel during plant startup. They are controlled by the reactor power control system (System 535).

System 3123 Borated water supply system

System 3123 supplies the primary with 7000 ppm borated water to adjust power level, and shut down the reactor. The system 3123 design shown in Figure A.3-3 is similar to the previous systems. It consists of:

- Two reciprocating pumps (P31 and P32) rated at 6.0 kg/s (95 gpm) and 65 kW from system 643 which take suction from the borated water tank (331 T23) in the reactor water cleanup system (System 3312) and LiOH solution and H₂ delivering to the inlet-outlet system (3124) at about 10.5 MPa.
- Pump service valves and a check valve to system 3312
- A flow meter (312 K331)
- A pressure pulse compensator (312 T31) after the pumps

Figure A.3-3. Borated Water Supply System

The pumps are 100% redundant for power operation of the plant but are used in parallel when necessary for rapid reduction of reactivity. They are controlled by the reactor power control system (System 535).

System 3124 Inlet-out system

System 3124: supplies and preheats the water from systems 3121, 3122 and 3123 for delivery to the primary system, withdraws and cools water from the primary system for delivery to the reactor water cleanup system. System 3124 shown in Figure A.3-4 consists of:

- The supply pipe connections from the pump systems, systems 3121 (F), 3122 (G) and 3123 (H).
 - One inlet pipe (F) for borated water (from system 3123), with
 - Two containment isolation valves (V101 and V102)
 - One flow meter (KA342)
 - One regenerative heat exchanger (E1)
 - Four motor-operated inlet valves (V103 to V106), one for each steam generator loop (cold leg) in the recirculation system (system 313)
 - One inlet pipe for filtered water (G, from system 3121) and one for distilled water (item H from System 3122) with
 - Four containment isolation valves (V201 to V204)
 - Two flow meters (KA 343 and KA344))
 - One regenerative heat exchanger (E2), the two pipes unite upstream the heat exchanger
 - Two motor-operated inlet valves (V205 and V206), one for each inlet header, (item I, TAI and TBI) -
- Suction lines from the inlet headers (TAI and TBI) to the external recirculation loop hot leg outlet nozzles (System 3131)
- A normal operation water outlet pipe system with four outlet pipes from the four cold legs (immediately downstream of the main coolant pumps), each equipped with shut-off valves (V107 to V110)

Upstream heat exchanger E1 the four pipes unite, and the common pipe leads to the regenerative heat exchangers E1 and E2, and from it to system 3125.

System 3124 is capable of a maximum flow of 70 kg/s (1110 gpm) consisting of: borated water, 0-12 kg/s, filtered water, 0-17 kg/s and/or distilled water, 0-60 kg/s.

System 3125 Normal withdrawal system

System 3125: controls the primary water withdrawal rate via system 3124, cools the withdrawn water to temperatures acceptable to the reactor water cleanup system (system 331), depressurizes the withdrawn water to pressures suitable for system 331, continuously measures the boron content of the withdrawn water.

System 3125 shown in Figure A.3-4 (dashed block labeled (J)) consists of:

- One cooler (E3)
 - One temperature measurement (KA551)
 - One flow meter (KA351)
 - Outlets to the sampling system (System 336) and the water chemistry control system (System 533) -
- Fixed throttles (VA531 to 533)
- Motor-operated shut-off valves (V511 to 513)

The fixed throttles (VA531 to 533) reduce the pressure and control the flow rate. Shut-off valves (V511 to V513) are individually controlled by the reactor power control system (System 535). When large withdrawal rates are required all valves are opened with all outlet pipe trains operating in parallel.

System 3126 Temporary discharge system

System 3126 has the following tasks: support the normal withdrawal system (System 3125) by withdrawing primary water during certain plant transients, and discharge primary water to the condensation pool of the reactor containment (system 316).

System 3126 is shown in Figure A.3-4 in the dashed area (K) consisting of three parallel water withdrawal trains, each comprising:

- One 25 kg/s (400 gpm) water blowdown valve (V601 to V603)
- One discharge throttle (V701 to V703)
- One discharge pipe equipped with vacuum breaking check valve (V801 to V803)

A3.2 Operation of System 312

Normal Power Operation

During normal steady state plant operation, the withdrawal system (System 3125, Figure A.3-4) draws off about 8.5 kg/s (134 gpm) through the regenerative heat exchangers in the inlet-outlet system (System 3124). The water is filtered in the reactor water cleanup system (System 331) and most of it is normally returned to the primary system again via the filtered water supply system (3121). Water from the distilled water supply system (3122) may be injected or borated water from system 3123, according to the reactivity needs. The withdrawal rate is set by the fixed outlet throttles in the withdrawal system (System 3125). The inlet rate is controlled by temperature sensors in the upper density lock system (3222) to keep the hot/cold water interface level constant in the density lock.

The core outlet temperature is adjusted during normal operation by adjusting the core power level. Distilled water from system 3122 or borated water from system 3123 is pumped into the primary system according to the needs for power regulation. The filtered water supply rate from the filtered water supply system (system 3121) automatically compensates for this change in primary water volume.

Power Ascension

Start up is performed by pumping clean water into the primary system by the distilled water supply system (3122) while removing an equal amount of the primary water by the withdrawal system (3125). The position of the hot/cold water interface level in the upper density lock is sensed by temperature which is used to control the flow from the filtered water supply system (item (G), 3121).

Power Reduction and Shutdown

When rapid step load reductions are required with rapid inflow of borated water, additional primary water is discharged through system 3126 to the condensation pool (System 316) to minimize the outflow of primary system water through the upper density lock (3222).

Normal plant shutdown is performed by pumping borated water at full capacity into the primary system by the borated water supply system (3123). Attaining boron concentrations corresponding to hot standby (decay heat level) takes about 12 or 30 minutes at the end or beginning of a core load cycle, respectively. Attaining boron concentrations corresponding to cold shutdown takes about two or five hours at the end or beginning of a core load cycle respectively.

Figure A.3-4 Inlet and Outlet System (HORIZONTAL - FULL PAGE)

Abnormal System Behavior

If the balance between the system 312 inputs and outputs to the primary loop is not maintained, one of two things will happen. Either primary loop water and energy will be lost through the upper density lock (system 3222) to the reactor pool with unnecessary contamination. Or the upper density lock will be made inoperable because of erratic composition and temperature of the water in the pipe bundle. Erratic boron concentration in system 3121, 3122 or 3123 will be reflected by erratic reactor power. If the reactivity swings are sufficient, the interlock system and the reactor power control system (systems 516 and 535) will scram the reactor. Uncontrolled injection of distilled water into the primary circuit will lead to a reactivity excursion and automatic scram. If this were to occur during plant shutdown, the cold non-borated water will flow out into the concrete vessel pool via the system 312 inlet headers (TAI and TBI) without entering into the primary system.

A3.3 Failure Modes Effects and Criticality Analysis

Table A.3-1 provides failure modes, effects and criticality analyses for the equipment for system 213.

FMECA Notes for Appendix A-3

Note a The scenario visualizes full power operation with 100% pump rotational speed. It is assumed that the primary rotational inertia is in the motor rotor and shaft. Abrupt pump seizure will result in a high inertial impulse tending to tear the pump from its mount and sever the connecting pipes. If this were to occur, a back flow would take place through the supply pipe from the 9 MPa reactor pressure. These pipes are small so it would be a small LOCA.

Note b The inability to inject boron disables the ability to control reactor power; it would prevent a normal shutdown of the reactor and necessitate a scram.

Note c A small LOCA associated with flow back into systems 3121, 3122, or 3123 requires a very high order of coincidence. A failure of the check valve in one of these lines must be accompanied by pump failure in the line having the failed check valve, by the doubly redundant pumps in the line and by the check valve in that line. Furthermore closing the containment isolation valves would also prevent the back flow.

Note d The differential pressure across the cooler is minimized by the throttling valve (V107-110) controlling the letdown rate into the cleanup system. The minimum is the pressure needed for the cleanup system. If this is lower than the secondary pressure, the leak is from secondary to primary and no radioactive water will be released to the environment. If the other way, there will be a release in proportion to the pressure differential.

Table A.3-1 PIUS 600 Failure Modes Effects and Criticality Analysis

System: Coolant Process Systems: Filtered Water 3121 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: December 18, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
1	Reciprocating pump/ supply filtered water to primary coolant	pump failure	shaft lock at power	Potential for small LOCA	L	IV	Place check valve after pump dis- charge or use shear pin	See note a.
			pump seal failure, motor failure, power failure, coupling fail- ure, rotor loose from shaft	Loss of filtered water supply. Loss can be compensated by 100 % capacity backup	L	III	Redundant pumps in design.	Losing a pump is expected.
		Excessive flow	Pump speed electronic control failure	Flow into pool, dilution of pool poison	L	III	QA	If not corrected fairly promp- tly, it would be necessary to shut the reactor down. It will cause flow from the upper density lock into the pool.
2	Fluid envelope / convey filtered water to primary	envelope failure	cyclic fatigue stress corro- sion, seismic	Small LOCA	L	V	Inspection, Conservative design	Any envelope failure after the check valves will result in a small LOCA.

Table A.3-1 PIUS 600 Failure Modes Effects and Criticality Analysis

System: Coolant Process Systems: Filtered Water 3121 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: December 18, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
3	Check valve/ prevent back flow to the cleanup system	valve failure and failure of both pumps	sticking, foreign matter, loss of flapper	Small LOCA	L	IV	Inspection	A check valve status indicator would warn of this potential accident
4	Pressure pulse compensator/ minimize pressure pulses	fails to smooth pulses	diaphragm leak, loss of gas	increased pressure pulses in the water supply system	L	III	Inspection	Failure detectable by dynamic pressure monitoring of pressure pulses
5	Flow sensor/ measure the rate of filtered water flow	fails to work	clogging of sensing orifices, failure of pressure differential instrumentation	Inability to know the rate at which filtered water is being supplied. May require reactor shutdown	L	II	QA	

Table A.3-1 PIUS 600 Failure Modes Effects and Criticality Analysis

System: Coolant Process Systems: Distilled Water 3122 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: December 18, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
1	Reciprocating pump/ supply distilled water to primary coolant	pump failure	shaft lock at power	Potential for small LOCA	L	III	Place check valve after pump dis- charge or use shear pin	See note a.
			impeller failure of primary boundary	see above	L	III	Inspection, pump casing of suffi- cient strength to withstand impeller failure also see above.	The scenario visualizes sudden rotor failure that penetrates the pump casing causing a small LOCA
			pump seal failure, motor failure, power failure, coupling fail- ure, rotor loose from shaft	Loss of distilled water supply. Loss can be compensated by 100% capacity backup	L	II	Redundant pumps in design.	Loosing a pump is expected.
		Excessive flow Design	Control fail-ure	Reactivity excu- sion	M	III	Design	Rapid injection of distilled wa- ter would cause a reactivity transient that would activate the pool loop to scram the re- actor. It will also cause flow from the upper density lock.
2	Fluid envelope / convey filtered water to primary	envelope failure, seismic	cyclic fatigue stress corro- sion	Small LOCA	L	V	Inspection, Conservative design	Any envelope failure after the check valves will result in a small LOCA.

Table A.3-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Coolant Process Systems: Distilled Water 3122 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: December 18, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
3	Check valve/ prevent back flow to the cleanup system	valve failure and failure of both pumps	sticking, foreign matter, loss of flapper	Leak back into filtered water tank	L	IV	Inspection	The check valve in system 3124 would prevent a small LOCA if this happened.
4	Pressure pulse compensator/ minimize pressure pulses	fails to smooth pulses	diaphragm leak, loss of gas	increased pressure pulses in the water supply system	L	III	Inspection	Failure detectable by dynamic pressure monitoring of pressure pulses
5	Flow sensor/ measure the rate of distilled water flow	fails to work	clogging of sensing orifices, failure of pressure differential instrumentation	Inability to know the rate at which distilled water is being supplied. May require reactor shutdown	L	II	QA	

Table A.3-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Coolant Process Systems: Borated Water 3123 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: December 18, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
1	Reciprocating pump/ supply borated water to primary coolant	pump failure	shaft lock at power	Potential for small LOCA	L	III	Place check valve after pump dis- charge or use shear pin	See note a. See note b.
			impeller failure of primary boundary	see above	L	III	Inspection, pump casing of suffi- cient strength to withstand impeller failure also see above.	The scenario visualizes sudden rotor failure that penetrates the pump casing causing a small LOCA
			pump seal failure, motor failure, power failure, coupling fail- ure, rotor loose from shaft	Loss of borated water supply. Loss can be compensated by 100% capacity backup	L	II	Redundant pumps in design.	Loosing a pump is expected.
		Excessive flow	Control fail-ur- e	Shut the reactor down	L	III	Design	Rapid injection of borated wa- ter will cause a rapid reactor shutdown. It will cause flow from the upper density lock into the pool.
2	Fluid envelope / convey filtered water to primary	envelope failure	cyclic fatigue stress corro- sion, seismic	Small LOCA	L	V	Inspection. Conservative design	Any envelope failure after the check valves will result in a small LOCA.

Table A.3-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Coolant Process Systems: Borated Water 3123 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: December 18, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
3	Check valve/ prevent back flow to the cleanup system	valve failure and failure of both pumps	sticking, foreign matter, loss of flapper	Small LOCA	L	IV	Inspection	The check valve in system 3124 would prevent a small LOCA if this happened
4	Pressure pulse compensator/ minimize pressure pulses	fails to smooth pulses	diaphragm leak, loss of gas	increased pressure pulses in the water supply system	L	III	Inspection	Failure detectable by dynamic pressure monitoring of pressure pulses
5	Flow sensor/ measure the rate of borated water flow	fails to work	clogging of sensing orifices, failure of pressure differential instrumentation	Inability to know the rate at which borated water is being supplied. May require reactor shutdown	L	II	QA	

Table A.3-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Coolant Process Systems: Inlet-Outlet 3124 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: December 18, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
1	Piping to check valves/ carry filtered, borated and distilled water to reactor	Pipe failure	Fatigue fracture, bad weld, vibration, stress-corrosion cracking, seismic	Small LOCA	L	IV	Inspection, QA	Pipe rupture in these lines would depressurize the reactor and allow reactor water to be released outside of the PCRV but inside of containment.
2	Check valves V102, 202, 204/ prevent back flow and reactor depressurization	Valve failure to prevent reverse flow and failure	Sticking, foreign matter, loss of flapper	Small LOCA	L	V	Inspection, QA, Maintenance	See note c
3	Regenerative heat exchanger (E1)/ heat borated water before injection into the primary	Primary to secondary leak	Pitting, stress corrosion, bad welds, vibration, seismic	None	L	III	Inspection	
4	Motor operated inlet valves/ control flow from water supply systems	valve failure to operate	valve binding, control failure, motor failure	inability to control reactivity	L	III	Maintenance	This failure would probably force a scram
5	Containment isolation valves	See above	See above	inability to isolate containment given that there is a failure of the connecting system	L	III	Maintenance	Presumably these valves could be manually closed. See note c

Table A.3-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Coolant Process Systems: Inlet-Outlet 3124 **Reference:** PIUS PSID **Preparer:** R. Fullwood, BNL **Date:** December 18, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
6	Flow meters/ monitor	Failure to measure flow	Clogging of sensing orifices, failure of pressure dif- ferential in- strumentation	Inability to control reactivity	L	II	maintenance	

Table A.3-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Coolant Process System: Normal Withdrawal 3125 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: December 19, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
1	Cooler/ cool primary water to a temperature for cleanup	Leak from primary to secondary	pitting, vibration, abrasion, welding, differential expansion, seismic	Leakage of primary water into secondary water with environmental release	M	III	Monitor secondary coolant from radiation	See note d
2	Temperature measurement/ determine the water temperature going to cleanup	Sensor or Instrumentation failure and failure of cooling	Sensor failure, electronics failure	Inability to monitor the water temperature going to cleanup	L	IV	Inspection	
3	Flow measurement/ determine letdown flow rate	Sensor or Instrumentation failure and failure of flow control	Sensor failure, electronics failure	Inability to monitor the water flow rate going to cleanup	L	IV	Inspection	
4	Fixed throttles/ measure flow by differential pressure	Clogging or erosion	Foreign matter, boron deposition, corrosion	Error in knowing flow rate or blockage of letdown flow	L	III	Maintenance	
5	Motor-operated shutoff valves	Valve failure to close	Valve sticking, control failure	Inability to stop flow to cleanup system	L	II	Maintenance	
		Valve failure to open	see above	No flow to clean-up system	M	III	Maintenance	This would prevent the ability to control the reactor because water must be released to inject controlling water.

Table A.3-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Coolant Process System: Temporary Discharge 3126 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: December 19, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
1	Discharge throttles/limit the transient discharge rate	Clogging or erosion	Foreign matter, boron deposition, corrosion	Limited or excessive discharge rate.	L	III	Maintenance	It seems that there are wide tolerances to the transient discharge rate
2	Motor-operated shutoff valves	Valve failure to close	Valve sticking, control failure	Inability to control transient discharge rate	L	II	Maintenance	
		Valve failure to open	see above	No flow to transient discharge system	L	III	Maintenance	This would prevent result in excessive transient flow and discharge to the pool through the upper density lock
3	Envelop/ guide flow discharge	pipe failure	fatigue, stress corrosion, welding, unrelieved strain, seismic	Discharge inside of containment	L	IV	Inspection	

APPENDIX A.4 - PRIMARY RECIRCULATION SYSTEM (SYSTEM 313)

In current LWR plants, the reactor coolant pumps supply the pressure difference that forces the reactor coolant through the reactor core, the hot and cold leg piping and the steam generator. However, in the PIUS design, the reactor core and riser pressure drop is supplied by the density difference between a hot and a cold water column. The main coolant pump takes the water from the top of the natural circulation loop and transfers it to the bottom via the steam generator(s) thus keeping the natural circulation reactor pool loop inactive. The pump has to provide the same pressure differential as that caused by the density difference through the core to establish a pressure balance across the lower density lock (System 3221).

System 313 Purpose

The purpose of System 313 includes the following:

1. Providing sufficient primary loop flow to ensure that the reactor pool loop is not activated during normal plant operation;
2. Providing suction pressure for the reactivity control system inlet pipes (System 3124);
3. Allowing high boron content pool water to enter the primary system by natural circulation following a main coolant pump trip;
4. Ensuring that the pool water enters the primary system from below the steam generator and thus, minimize the thermal stresses imposed on the components (primarily on the steam generator tubes).

System 313 Composition

The recirculation system, shown in Figure A.4-1, consists of the following subsystems:

- Hot and cold leg piping (System 3131)
- Steam generators (System 3132)
- Main coolant pumps (System 3133)

Hot and cold leg piping (System 3131)

The diameters of the hot and cold leg pipes are 665 mm and 635 mm, respectively. The pipes are made of forged and machined sections of 304 stainless steel that are welded together. The cold leg inlet nozzles are elongated to provide room for a nozzle with reduced diameter section to serve as a break flow limiter. A potential failure of the break flow limiter could cause an increase in the size of loss of coolant accidents that should be considered in the design analysis.

Steam generators (System 3132)

The steam generators are a once-through, straight tube design. The steam generator and the associated primary system piping is arranged such that low boron water can be drained in the event of a trip of all main coolant pumps. The steam generators are mounted on supports that are fixed to the concrete vessel and are provided with snubbers to prevent excessive movement in the event of an earthquake.

Figure A.4-1 Recirculation System

The thermal expansion of the hot leg pipes is relieved by pipe bends; however, there will be some displacement of the steam generators through sliding in the bearings against the supports. The steam generators have the following main parts:

- cylindrical shell;
- primary water inlet plenum;
- steam outlet plenum;
- tube bundle;
- feedwater inlet plenum;
- primary water outlet plenum with the main coolant pump impeller.

The cylindrical envelope, or the shell, has an inner diameter of 2.35 m with a wall thickness of about 35 mm. The shell is thicker (55 mm) both at the upper end and the lower end at the location of the feedwater inlet and steam outlet nozzles.

The inlet plenum, or upper head, has a hemispherical chamber of 2.1 m inner diameter with a wall thickness of 100 mm. The inlet nozzle is located at the top of this chamber; and the upper tube sheet is the bottom of the chamber.

The steam outlet plenum chamber makes up most of the annular space between the shell and the tube bundle compartment wall. The inner diameter of the shell and the outer diameter of the wrapper are 2.35 m and 2.12 m respectively; the height is about 13 m. The steam outlet nozzle has a diameter of 900 mm.

The tube bundle compartment is made up by the wrapper (a cylindrical steel plate with inner diameter of 2.1 m and thickness 10 mm) surrounding the tube bundle along its length. The tube bundle is made up of 7000 tubes of 15 m length. The outer diameter of the tubes is 15.875 mm with a thickness of 0.86 mm. The tubes are arranged in a triangular pitch of 22 mm. The tube sheets at the upper and lower ends are about 300 mm thick, at the location where the tubes are expanded and welded to the sheets. Tube support plates are provided at regular intervals to minimize tube vibrations.

The feedwater inlet plenum chamber is the lower annular space between the shell and the wrapper with a height of 2 m; the feedwater inlet nozzle has a diameter of 350 mm.

The primary water outlet plenum has an inner diameter of about 2.1 m and a height of about 1.7 m. The lower tube sheet is at the upper end of the plenum and the main coolant pump impeller is inserted from the bottom into the cone-diffuser structure. The pump header outlet nozzle that connects to the cold leg piping has a diameter of 700 mm.

The steam generator shell is made of SA 533, Class 2, Grade B material; the tube sheets and the upper and lower heads are fabricated from SA 508, Class 3 material. Other materials in the steam generator design include:

- wrapper: SA 285, Grade C
- tube support plates: SA 240 Type 405 SS
- tubes: Inconel 600.

Main coolant pumps (System 3133)

The main coolant pumps are similar to the wet motor recirculation pumps utilized in the ABB Atom BWR plant designs. The pumps have the following design data:

Speed	1450-1500 rpm
Head	80 mWG
Motor power	3.5 MW
Motor voltage	3000 V
Pump capacity	4.5 m ³ /s

The pump motors are water-filled asynchronous squirrel cage motors, supplied with power from the adjustable speed drive converters of the main coolant pump power supply system (System 649). The power is supplied at a variable voltage and frequency. (The ratio between voltage and frequency is kept constant to achieve constant electromagnetic conditions in the motors.) The pumps are supplied with vibration dampers and are cooled by water from the non-safety related closed cooling system (System 723). The cooling is accomplished with heat exchangers 313 EA31 through EB32. The motors are supplied with a purge flow from the high pressure distilled water system (System 734) to prevent ingress of primary reactor water.

The speed of the pump motors is controlled by the main coolant pump control system (System 534) and the main coolant pump power supply system (System 649). The maximum pump speed is 105 % of the nominal speed. This maximum speed is also referred to as "full pump speed".

Normal power operation

During normal power operation the main coolant pumps are operated at a speed that is a few percent lower than the full pump speed. This mode of pump operation is used when the reactor is operating at full power and when the plant is at part load. The pump speed is controlled by the main coolant pump control system (System 534) and the main coolant pump power supply system (System 649). As described in system chapters 322 and 534, pump speed control is required to keep the reactor pool loop inactive. The pressure head of the pumps is designed to overcome the pressure drop in the steam generators and the recirculation system piping with the siphon breakers, and the density induced pressure differences in the pool loop inactivation system (system 322) and the downcomer. This will total 10.53 MPa (5.3 bar). As discussed above, the pressure drop across the reactor core and the riser is not overcome by the main coolant pumps but balanced by the density induced pressure difference between the hot riser water column and the cold pool water column acting on the density lock interfaces.

The following data are representative of full power operation:

Main coolant pump flow rate:	13200 kg/s (2E5 gpm) 16.8 m ³ /s
Power transferred:	2000 MWt

Abnormal operating conditions

The reactor pool loop will be activated within a few seconds following a malfunction in System 313 or the main coolant pump power supply system (System 649) that causes the main coolant pumps to stop. The reactor will be shut down and residual heat will be removed by the reactor pool.

During normal operation, the main coolant pump speed is controlled to keep the hot/cold fluid interface in the lower density lock (System 3221) at the desired level. Pump malfunctions that cause overspeed conditions will force primary coolant to leave the reactor loop through the lower density lock. Reactor pool water will enter the primary loop through the upper density lock (System 3222) and eventually cause the reactor to shutdown.

The transients imposed on the plant during a main coolant pump malfunction mainly affect the reactor internals and are discussed in system 213.

Seizure of the main coolant pump (e.g., locked rotor) or pump shaft failure can be serious transients in current PWRs and may result in core damage. However, in the PIUS design, these transients will only result in a plant shutdown in the same way as a pump trip.

Malfunctions in the plant that result in transients in the primary system water temperature i.e., power grid or feedwater system failures will influence the thermal interface system. The main coolant pump control and power supply systems (System 534 and 649) will respond to these transients and change the speed of the pumps to counteract the motion of the density lock interface.

The transient analysis of the plant behavior takes the response of the main coolant pumps into account. This analysis shows that the pump response does not affect the inherent safety features of the plant.

A.4.3 Failure Modes Effects and Criticality Analysis

Table A.4-1 presents a failure modes effects and criticality analyses for the equipment in system 313 based on the available information.

FMECA Notes for Appendix A-4

Note a A large LOCA (fracture of the largest primary cooling pipe) would result in discharge of primary coolant inside of containment to the condensation pool. The drop in level of coolant is limited at the height of the siphon breaker for reverse flow in the cold leg. In the hot leg, the drop in level of coolant is limited at the height of item R Figure A.2-6 which is a bit above the siphon breaker level. It is impossible to uncover the core by this mechanism. Core uncover due to pipe LOCA can only occur by a break followed by sustained steam generation without appreciable condensate return.

The steam pressure in the upper reactor dome, pressurizes both the pool and primary water. A loss of pressure on the primary water will, to some extent, result in pool water being driven into the reactor through both density locks. The amount of this effect depends on the inertia of the masses being driven and the flow impedances.

Note b

When PIUS scrams, the cold pool water fills the flow guides resulting in approximately a 200°C change in temperature in 90 seconds. If primary pumps continue operation, the cold water will circulate through the primary loop with significant thermal shock.

Note c

This scenario assumes that abrupt stopping of a main cooling pump will result in such torque that the pump may fracture its piping. For the PIUS pumps, it is not clear that there is sufficient stored energy because there is no mention of fly-wheels to extend the coastdown time. The inertia of the rotating pump assemblies may not be sufficient to cause such a break. Furthermore, it depends on the strength of the pump mounts to withstand the force.

Note d

Figure A.4-1 does not show any check valves in the primary pump lines. This suggests that all 4 pumps must be on at the same time. Additionally, if one pump fails, some flow will bypass the core and go through the failed pump in reverse. If this impedance is small compared with the impedance through the core, very considerable cooling will be bypassed and activate the pool loop.

Note e

Pump overspeed will result in discharge of primary coolant from the lower density lock and intake of pool water through the upper density lock. When the entrained pool water travels through the steam generator back to the core, it will tend to shutdown the core although a portion of the pool water will be discharged through the lower density lock. However the poison that enters the core will reduce reactivity, and reduce thermal pumping which will result in an increase in the rate of pool water entering through the upper lock. When the reactor is shutdown, the thermal pump will be zero and the relative amounts of flow through the core and through the pool will be in the inverse ratio of their impedances.

If it is conceivable that a pump could considerably overspeed ($> 105\%$) to fracture the impellor, or other rotating parts, then the scenario becomes that of 3a and note c applies.

Table A.4-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Primary Recirculation: Piping 3131 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: December 20, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
1	Piping (4 loops)/ transport the hot and cold coolant from the reactor upper structure to the steam generators and back to the reactor	pipe break	crack growth from cyclic fa- tigue, stress corrosion cracking, unrelieved strains, seismic loading	Large LOCA	M	V	Periodic NDE inspection	See note a
			thermal fatigue	see above	M	IV	see above	See note b

Table A.4-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Primary Recirculation: Steam generators 3132 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: December 27, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
2a	Steam generators (4)/ produce steam powering the tur- bines.	tube leak	crack growth from cyclic fa- tigue, stress corrosion cracking, unrelieved strains, denting	primary to sec- ondary leak	L	II	Periodic NDE inspection	Radioactive gases and particles from a primary to secondary steam generator tube leak will be contained within system 441. Non-condensable gases are processed by system 442 for filtration and delayed release.
2b		tube(s) break	over pressure with or with- out: crack growth from cyclic fatigue, stress corro- sion cracking, unrelieved strains, denting, tube/ tube sheet weld cracking	SLOCA	M	IV	see above	Primary will discharge and overpressure secondary causing failure of condenser and release of radioactivity to atmosphere.

Table A.4-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Primary Recirculation: Main Coolant Pumps 3133 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: December 18, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
3a	Main coolant pumps (4)/ (1) Circulate primary water through reactor and steam generator and (2) match the reactor thermal head to prevent pool loop activation.	Mechanical impeller lock-up	loose part, foreign object, impeller fracture, bearing seizure	large LOCA	M	V	Inspection, close control on foreign object entry, loose part monitoring, bearing temperature sensing, vibration sensing	See note c
b		Motor failure	open winding, short between windings or to ground	loss of flow from one pump followed by back flow and bypass of core cooling	L	III	Inspection, electrical detection of insulation breakdown	See note d
c		Drive coupling failure	fracture	see above	L	IV	Inspection, vibration sensing	See note d
d		Pump power supply failure	loss of off-site power, supply breaker trip	see above	L	II	Supply redundancy, switchgear inspection	See note d
e		Pump speed control failure - under or no speed	electronics failure	see above	L	III	Redundant control, inspection	See note d

Table A.4-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Primary Recirculation: Main Coolant Pumps 3133 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: December 18, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
3	Main coolant pumps (4)/ (1) Circulate primary water through reactor and steam generator and (2) match the reactor thermal head to prevent pool loop activation.	Pump speed control failure - overspeed	Electronics failure	Activate pool loop - shutdown the reactor	L	IV	Design, redun- dancy	See note e

APPENDIX A.5 - PRESSURE RELIEF AND CONDENSATION SYSTEMS

A.5.1 Pressure Relief System 314

Safety grade system 314 discharges steam from the pressurizer steam dome to the condensation pool to: prevent vessel overpressurization during plant transient and abnormal operating conditions, force blow-down of the reactor pressure during certain plant abnormal conditions, manually actuate reactor pressure blowdown following a long period of residual heat removal by reactor pool water boiloff.

System 314 is diagramed in Figure A.5-1. It is fabricated from 304 stainless steel and consists of: safety valves, relief valves, outlet system, and emergency blowdown valves. Safety (B) and relief valves (C), each 100 mm i.d. are installed directly on the top of the pressurizer dome (A). The emergency blowdown valves (E), also 100 mm i.d., are installed at the wall of the containment and connected to the pressurizer offgassing system (system 3285).

Safety Valves

The safety valves prevent overpressurization of the reactor vessel during plant transient and abnormal operating situations. The four safety valves (314 V11 to V14) will each discharge 25 kg/s (396 gpm) at full operating pressure. They are operated by pressure actuated pilot valves (V111 to V114) via servo valves (V121 to V124). The set opening pressure is 10.0 ± 0.2 MPa for two of the valves and 10.2 ± 0.2 for the two others. The nominal valve opening pressures are: 9.9, 10.0, 10.1, and 10.3 MPa

Relief Valves

The relief valves blow-down the pressure vessel during certain plant abnormal conditions (such as station blackout for more than two hours). Valves V21 and V22 are operated by servo valves (V221 and V222) which in turn are operated by solenoid valves (V211 and V212). The solenoid valves actuate the relief valves only on manual order. The capacity of each relief valve is 25 kg/s (396 gpm) at full operating pressure - the same as the safety valves.

Discharge Pipe System

Each of the safety and relief valves has a discharge pipe to the condensation pool (F), (System 316), a vacuum-breaking check valve on the discharge pipe (item (D), V31 to V34), a temperature monitor on the valve outlet (KS11 to KS14) to monitor valve leakage, a pressure monitor (K111 to K114) on the valve outlet, both monitors have central indication and alarm. The diameter of the discharge pipes is 150 mm i.d.

Emergency Blowdown Valves

The emergency blowdown valves enable depressurization of the reactor system if all the reactor pool cooling systems (the forced circulation pool cooling system, system 323, and the passive pool cooling system, system 724) have been lost. By doing so, water may be replaced from external sources by means of normally available low pressure pumps. This is affected by two valves (V41 and V42), each with a capacity of 25 kg/s (396 gpm) at full reactor pressure. The emergency blowdown valves are installed at the wall of the containment with their control shafts extended outside the containment. The valves connect to the pressurizer steam dome by 100 mm i.d. pipe of the pressurizer offgassing system (System 3285), and outlets to the relief valve discharge pipes.

A.4.2 Operating Situations

Normal Plant Operation

During normal plant operation steam or water leaking through the valves will pass to the condensation pool (System 316). Leaks are detected and measured by temperature monitors.

Figure A.5-1 Pressure Relief System

Overpressure Situations

Safety valves handle all overpressure situations. If a safety valve opens, the steam discharges into the condensation pool. The safety valves are expected to operate very infrequently or not at all during the life of the plant. They are primarily for protection in ATWS transients.

Abnormal Situations

Failures will involve valve leakage or inadvertent valve opening; failure of the safety valves to open is a potential safety issue affecting the concrete vessel integrity. Additional protection is provided by elongating the bolts connecting the head of the vessel steel extension (System 211) to relieve the pressure long before the ultimate strength of the tendons is exceeded.

Emergency Blowdown Situation

If the active and passive reactor pool cooling functions have been lost for several days (a week), water will have to be supplied to the reactor pool in order to avoid an eventual core degradation.

This water replenishment may be by fire truck, but to do this, it is necessary to be sure that the reactor pressure is reduced to a level where normally available hoses and pumps can be used.

A.5.2 Condensation System 316

System 316 is like the condensation systems in BWR plant containments of pressure suppression type. It receives water from the temporary reactor water discharge system (System 3126), and receives steam from the safety valves on the secondary side of the steam generators.

After an accident, the system receives and condenses steam from the reactor pressure relief system (System 314) and from ruptures in the reactor primary system. Water in the condensation pool is used for filling the reactor service pool with water. When the reactor service pool is drained, water is sent to the condensation pool, thus serving as water storage.

Figure A.5-2 shows the condensation system (316). It is located in the wetwall of the reactor containment consisting of: a pool of water (G), a gas compression volume (H), blowdown pipes (L) from the drywell (J) into the pool, reverse flow (vacuum breaking) valves, (V101 to V116) between the drywell and wetwell to let non-condensable gases flow back to the drywell, temperature (K502) and water level (K401) instrumentation in the pool, and temperature (K501) and pressure (K101) instrumentation in the compression volume.

The blowdown pipes from the reactor safety and relief valves (System 314) are located in six of the blowdown pipes in system 316.

System 316 has: pool water volume of 2100 m³ (554,800 gal.), compression volume of 21500 m³ (5.68E6 gal.), 120 blowdown pipes, of 600 mm diameter, 100, 250 mm vacuum breaking valves.

Normal Plant Operation

During normal operation the pool may be heated by blowdown of hot reactor water from the primary loop via the temporary reactor water discharge system (System 3126) or by blowdown of steam from the safety valves on the secondary side of the steam generators.

The pool water temperature is kept below 40°C by water being circulated and cooled by the containment pool cooling system (System 355). The boron concentration in the pool is about 2000 ppm. Adjustments of the boron concentration and the total water inventory is done by transfer with the fuel pool cooling and cleanup system (System 324).

Figure A.5-2 Condensation System

Pipe Rupture Inside Reactor Containment

If a pipe break occurs in the reactor primary system or in the steam generator secondary side inside the reactor containment, a mixture of water/condensate will flow from the drywell (J) down into the condensation system (G). Steam flowing to the pool, will contain entrained air and non-condensable gases. These gases will rise from the pool into the compression volume and pass through the vacuum breakers (I) into the drywell.

The pressure in the containment vessel and the water temperature in the condensation pool depends on the type and size of the pipe rupture. The maximum pressure occurring after a double-ended cold leg break is about 2.6 bar and the maximum water temperature about 70°C. In certain cases the pool water temperature may reach 95°C.

A.5.3 Failure Modes Effects and Criticality Analysis

Table A.5-1 presents a failure modes effects and criticality analysis for the equipment in system 314 based on the available information.

FMECA Notes for Appendix A.5

Note a

It might seem that a blowdown of the primary through the pressure relief valves would have the effect of automatic injection of pool water into the reactor with scrambling. However this is not the case because the pool pressure and the reactor pressure communicate in the upper head.

Note b

As the valve(s) remain open, water will be discharged and collected in the condensation pool as it is supplied by the system 312. Stopping the feedwater will stop the discharge but the reactor cannot be started until the stuck open safety valve is closed and repaired. The core cannot uncover by this mechanism because the release point is much above the core and pool water will enter to make up boil-off from the central region.

Note d

Overpressurizing the PCRV requires the failure of four safety valves and failure of manual actuation of the relief valves or manual actuation and failure of four relief valves. If this occurred, the pressure would stretch the tendons and cause a blow-down leak.

Note e

Inability to refill the reactor from an external water supply requires the failure of the relief valves and the manually-operated shaft-driven valves.

Table A.5-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Pressure Relief System 314 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: December 27, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
1a	Safety valves/ prevent overpresurization of the reactor vessel	Failure to open	Sticking of safety valve proper, sticking of pressure sensing valve or of servo valve	Overpressurization and fracture of PCR/V, vessel steel head, tendons, or primary piping.	M	IV	Testing of safety valves, stretching of tendons to blowdown at head seal, relief valves	Diversity in relief valves would be desirable. It seems that the three stage arrangement (sensor-servo-valve) is complex. See notes a and d.
1b		Failure to close after opening	See above	Pool water will flood reactor region and shut-down reactor	L	II	Testing and maintenance	See note b.
1c		Opening at less than the set point	Miscalibration, sensor failure	Inadvertent blow-down	L	III	Testing and maintenance, redundant design	This would result in an unnecessary transient stress cycle on the reactor components. Otherwise no effect.
2a	Relief valves/ manual relief of pressure for certain transients and external filling	Failure to open	Sticking of relief valve proper, sticking of solenoid valve, Human error in failing to actuate	Inability to relieve pressure under abnormal conditions including pressure relief for external addition of water <u>given</u> the inability to open the emergency blow-down valves.	M	III	Testing of relief valves, operator training, safety valves, tendon stretching	If the relief valves failed to open as required, the safety valves should automatically open to limit the maximum pressure.

Table A.5-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Pressure Relief System 314 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: December 30, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
2b	Relief valves/ manual relief of pressure for certain transients and external filling	Failure to close after opening	Sticking of relief valve proper, sticking of solenoid valve, Human error in failing to close	Pool water will flood reactor region and shut-down reactor	L	II	Testing and maintenance	Failure to reclose would prevent startup until the valve(s) are closed and repaired. See note b.
2c		Inadvertent opening	Human error	Inadvertent blow-down	L	II	Training	See above
3	Discharge pipe system/ transfer water from safety or relief valves to the condensation system	Pipe break	Stress cycles, vibration, unrelieved strain, stress corrosion, seismic	Pressurization of containment	M	IV	Inspection, design, stress relief	<u>No pressure-relieving vents for the containment have been identified.</u> Apparently failure to condense the steam could over-pressurize the containment.
4	Emergency blow-down valves/ pressure relief for emergency external water addition	Failure to open valves	Stuck valve, misalignment of valve actuating shaft, failure of shaft-to-valve coupling, failure to take action	Inability to relieve pressure sufficiently to inject water from a fire truck	M	IV	Inspection, testing, training	The inability to depressurize requires a failure of both the relief valve system <u>and</u> the emergency blowdown valves.

APPENDIX A.6 - CONTAINMENT

The purposes are: containment of radioactivity after accidents, contribution to the residual heat removal function, and cleanup of containment atmosphere.

The PIUS pressure-suppression containment resembles similar containment used in U.S. BWRs but the amount of radioactivity that can be released to the containment from an accident in PIUS is believed to be considerably less than prescribed by U.S. licensing practices as specified by USNRC Reg. Guides 1.3 (BWR plants) and 1.4 (PWR plants). The amount of radioactive matter which could be released to the containment after a pipe break inside the containment in a PIUS plant, can be estimated on the basis that a leakage corresponding to about 1 % damaged fuel in normal operation contaminating the primary coolant, but with no additional fuel damage occurring due to the accident.

An important element in the tightness of the containment is the closure of the isolation valves in lines which penetrate the containment. Isolation valves have the task of closing in the event of a pipe rupture as a part of the leak-tight containment, however the leakage requirements for PIUS containment is less stringent since the amount of radioactive matter which credibly can be released to the containment atmosphere is significantly lower than that specified for a present day U.S. LWR plant. For a current LWR, the safety-related task in case of a pipe rupture outside of the containment is to shut off the loss of reactor coolant to prevent core uncover, and to limit the atmospheric release of radioactivity. In the case of PIUS, the core will be left with enough water for passive long-term residual heat removal with no isolation. Limiting the release of radioactivity is less important than for a present day LWR plant because there is less radioactivity released to limit.

Residual Heat Removal

The residual heat from the core can be dissipated in three ways in a PIUS plant: 1) normal heat removal system, 2) reactor pool cooling system (System 323) and the non-safety diesel-backed cooling systems (Systems 721 and 712) to the ultimate heat sink, and reactor pool coolers immersed in the reactor pool. In the event of loss of AC power, reactor pool coolers dissipate the heat to the ambient air via the passive closed cooling system (System 724) and dry natural draught cooling towers (System 714). These systems are safety-related. No dedicated post-accident residual heat removal system is required for the containment. The initial release of steam and hot water from the primary system following a pipe break inside the containment is absorbed in the condensation pool of the containment. The reactor and the reactor pool are kept at a temperature below 100°C. Should all residual heat removal systems be lost, the reactor pool will still contain sufficient water for several days of residual heat removal by evaporation.

Cleanup of the Containment Atmosphere

Cleanup of the containment atmosphere in a conventional LWR plant is used to reduce the requirements for containment leakage and to recombine combustible gases. According to ABB Atom, in a PIUS plant, the cladding temperature will be too low after a pipe break to produce significant amounts of hydrogen generated by metal-water reactions. In a present day LWR plant, the usual way of reducing the amount of radioactivity in the gas volume is to spray the containment (PWR) or by a combination of spraying and scrubbing in the condensation pool (BWR) thereby substantially reducing condensable radioactive matter. In a PIUS plant, the condensation pool provides a passive means for reducing the radioactive matter in the containment atmosphere (except for noble gases and organic iodine). The condensation pool is very efficient during the blow-down phase after a pipe break. In the long-term phase, there will be no release of steam or water from the reactor system to the containment because the reactor and the reactor pool will be below 100°C rendering a containment spray unnecessary.

A.6.1 Reactor containment system

The reactor containment system (System 152) consists of:

- 1521 Containment vessel,
- 1522 Elevated cooling tower compartments,
- 1523 Reactor service pool and fuel pool structure,
- 1524 Containment vessel access penetrations
- 1525 Containment vessel pipe penetrations

System 1521 Containment Vessel

The containment structure consists, of all the main structural parts of the reactor building outside the PCRV, except the upper and outermost parts enclosing the reactor service and transport compartments. The base mat below the containment is part of the containment structure.

Main Structure

The layout of the containment is shown in Figure A.6-1 which displays a vertical cut through the containment structure. The reactor containment (H) is made of reinforced concrete. In the Figure, (A) identifies the PCRV, (B) is the air-filled space underneath the reactor for access to the tendons, (C) is the drywell volume into which the pressure relief valves discharge, (D) is drywell, (E) is a wetwell and (F) is a compression volume in addition to that in the wetwell. The main load-bearing part is the base mat (G), extending from the center to periphery of the containment's two concentric cylindrical walls. The outer wall encloses the suppression pool compartments, rising to 10 m above ground, and the inner wall encloses the PCRV. The primary systems rise to the containment roof level. The roof structure is comprised of a circular roof slab, structurally connected with the wall and the containment steel dome. The main load-bearing parts are reinforced concrete structures, with the exception of the removable steel dome. The primary leak-tight environmental protection barrier is an inside surface liner attached to the concrete wall and slab structures.

Figure A.6-2 shows a more detailed cut through the reactor building showing the spent fuel storage pool and the reactor internals storage pool used during refueling. (For scaling purposes, the distance from the bottom of the basemat to the top of the containment is 72.5 m.)

In the pool compartments the liner material is made of stainless steel, otherwise it is carbon steel. The thickness and the internal strength of the outermost walls and slabs of the containment is determined by requirements for protection of the primary systems against large external missile impacts. Structurally, the containment and the PCRV are connected at the base mat.

System 152.2 Elevated cooling towers compartment system

The system consists of four compartments placed on the roof structure of the containment. Each compartment contains one cooling tower for the natural circulation reactor pool cooling system (System 714).

System 1523 Reactor Service Pool and Fuel Pool Structure

The pool and structure is divided into four separate partitions placed on the containment roof slab. The reactor service pool with the main reactor internals storage pool and the spent fuel storage pool are placed perpendicular to the reactor service room. Two pools communicate through a gate in the partition wall. The other two partitions are placed on opposite sides of the reactor service pool in the reactor. One compartment for the second reactor internals storage pool with the spent fuel transport cask position and one compartment for the two cooling pools of the diesel-backed closed cooling system for component and pool cooling (System 722) for the concrete vessel liner cooling system (System 157).

Figure A.6-1. Cross Section Through the Containment

Figure A.6-2. Detailed Cut Through the Reactor Building

Systems 1524 and 1525 Penetrations

The containment is penetrated by pipes and electric cables as well as access penetrations and an opening in the containment roof above the reactor. The penetration pipes are anchored to the concrete structure to which the steel liner is encased and serve as fixed points for the pipes and designed to withstand transferred loads from a pipe rupture either inside or outside the containment without damaging the containment's pressure retaining and leak-tight functions. Each of the two doors of an access lock is designed to withstand the containment design pressure. The steel dome covering the opening above the reactor is bolted to an upper flange of a vertical steel cylinder. The flange is anchored to the concrete roof structure and welded to the steel liner.

Compartments and Installations

The containment vessel volume is mainly divided into two separate spaces, i.e. the drywell and the wetwell. The drywell (25,000 m³) contains the high pressure portions of the reactor primary system, e.g. the whole reactor recirculation system with main coolant pumps and steam generators (System 313), reactor pressure relief system (System 314) the reactor pool cooling system (System 323), parts of the reactivity control system (System and the reactor pool water replacement system (System 325). It also contains the main steam and feedwater piping to the steam generators including the safety valves for the main steam line system (System 411). Access to the drywell is possible through the access air locks. The wetwell is divided into a condensation pool (volume = 2000 m³) and compression chamber (volume = - 22,000 m³). Access to the wetwell is possible from the drywell and leads to the compression chamber through openings in the partition slab. The blowdown channels of the condensation system (System 316) and the outlet pipes of the reactor pressure relief system (System 314) and the safety valves for the main steam line system (System 411) lead to the condensation pool. Vacuum breakers are provided to prevent overpressure in the wetwell against the drywell.

Containment Design

The PIUS containment has a relatively large volume with a condensation pool inside it to condense outflowing steam from a pipe break etc resulting in a design pressure of 2.8 bars absolute.

The basis for the containment design is that the containment shall be able to withstand design pressure and temperature, without exceeding a design leakage rate. These tightness requirements shall be demonstrated by periodic integrated leakage rate testing. These requirements for the PIUS containment are the same as for a containment in a present day LWR plant as specified by 10 CFR 50, Appendix J.

For the calculation of off-site consequences, the leakage rate of gas is specified by 10 CFR 50 Appendix J, to be the design leakage rate divided by 0.75. For PIUS, a design leakage rate of 1% of the enclosed gas volume per day at maximum pressure has been chosen. In the calculation of environmental consequences for the FSAR, chapter 15, a leakage rate of 1.33% per day has been used according to 10 CFR 50 Appendix J requirements.

The design basis for isolation valves can be divided into three parts:

- 1) General arrangement of isolation valves, i.e. whether there should be outer and inner isolation valves or whether an outer isolation valve is sufficient,
- 2) The tightness requirements for isolation valves, individually as well as collectively, and
- 3) The requirements on closing times for isolation valves.

An important difference between PIUS and conventional PWRs is that even if a pipe ruptures outside the containment and the break is not isolated by means of an isolation valve, the core integrity will still be protected by the inherent passive, self-protective features. The core will be covered with water for a number of days without the function of active systems or operator actions.

The tightness requirements are set by the acceptance criteria for the integral periodic leakage test of the containment. In 10 CFR 50 Appendix J, type C tests are used for isolation valves, and type B tests are for air locks, penetrations etc. The type A test is the integral leakage test for the containment including air locks, isolation valves etc. Leakage during Type B and C tests, according to Appendix J, is 60% of the Type A allowed design leakage rate. ABB Atom's practice is that the allowed leakage during Type C tests is 50% of the Type A allowed design leakage rate. In accordance with subchapter 152.3.3 this leads to an allowed leakage through all isolation valves of 0.5% per day of the enclosed gas volume at the calculated maximum pressure.

The requirements on closing times for isolation valves will be based on the allowed environmental doses from release of the radioac set according to the pipe diameter in question.

A.6.2 Failure Modes Effects and Criticality Analysis

Table A.6-1 presents a failure modes effects and criticality analysis for the equipment in system 152 base on the available information.

FMECA Notes for Appendix A.6

Note a

Failure of containment, to be significant, must be concurrent with a reactor accident. It is conceivable that an earthquake could cause failure of the PCRV resulting in a large LOCA accident and fail the containment. Failing the containment without a release of radioactivity by itself would not constitute a safety hazard. Also it might be possible to sustain a core-damaging accident that resulted in over-pressurization of the containment and cause the containment to rupture, since there is no pressure relief mechanism for the containment. However external missiles such as tornado-driven or air crash must not only fail the containment but the missile must also result in a core-disruptive accident such as rupture and draining of the PCRV.

Table A.6-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Containment System: 152 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: December 30, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
1a	Containment vessel 1521/ contain radioactivity in the event of an accidental release of radioactivity	rupture of concrete walls and containment liner	earthquake, air crash, tornado, missile, overpressurization from reactor accident	Release of radioactivity to environment	M	V	design, inspection, vented containment	See note a
1b			overpressurization from reactor accident	Release of radioactivity to environment	M	V	pressure relief, vented containment	See note a
2	Elevated cooling tower compartments 1522/ protect cooling towers from possible air crash	Collapse of cooling tower compartments	earthquake, air crash, tornado, missile, poor design and construction	Remove passive cooling tower cooling from service - possible pipe rupture	L	V	design, inspection	The cooling towers are only needed if there is a loss of active cooling. Pipe break in the passive cooling loop would result in the loss of the affected loop
3	Reactor service pool and fuel pool structure 1523/ storage of radioactive components and fuel	Rupture of pool boundaries including liners.	see above	Overheating of spent fuel and/or release of particulate from contaminated components	M	IV	see above	The service and spent fuel pool are thinner and more elevated for impact and earthquake motion amplification than the PCRV but have less internal pressure.

Table A.6-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Containment System: Containment Vessel 1521 **Reference:** PIUS PSID **Preparer:** R. Fullwood, BNL **Date:** December 30, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
4	Containment vessel penetrations 1524/ penetrations of the containment for installation and service access	Failure of access penetration	excess pressure, metal fatigue, cracking, stress corrosion, improper installation	Containment leak in excess of regulations	L	IV	design, inspection, training, containment leak testing	Such a failure is safety-significant only if there is a concurrent accident.
5	Containment vessel pipe and electrical penetrations	Leak or blow-out	excessive pressure, improper material selection, fire, improper installation	Leak in excess of regulations	L	III	inspection, training, design, leak testing of containment	See above

APPENDIX A.7 - CONTROL AND MONITORING SYSTEMS

A.7.1 Trip and interlock system 516

System 516 differs from present day nuclear power plants in that it is not responsible for ensuring protection of the core integrity because of the inherent and passive protection functions thereby providing design simplifications.

System purposes

System 516 does the following: initiates actions to protect against significant releases of radioactive products from the plant and damages to plant equipment, initiates actions which minimize the number of times that the inherent and passive protection functions for core integrity protection are activated. It is a safety-grade system.

System design and operation

System 516 consists of the following subsystems:

- 5161 Reactor scram trip system
- 5162 Hot reactor shutdown trip system
- 5163 Isolation trip system
- 5165 Startup interlock system

All four 516 subsystems consist of what in principle is a four channel arrangement with a two-out-of-four logic.

Reactor scram trip system (5161)

System 5161 shuts the reactor down automatically after incidents which make the plant unable to sustain further operation above the residual power level and where a normal runback or shutdown of the reactor is not considered sufficient.

The system is tripped by some of the events which trip the hot reactor shutdown trip system (System 5162), but the analysis of the plant transient behavior does not yet allow a selection among them. When tripped, system 5161 will initiate the following actions:

- Trip one of the main reactor coolant pumps (System 313) via the main coolant pump power supply system (System 649),
- Interrupt the power supply to the filtered and distilled water injection pumps in the reactivity control system (Systems 3121 and 3122),
- Close the shut-off valves in the inlet lines,
- Trip a scram alarm in the alarm display system (System 517)
- Trip a scram alarm survey by the safety parameter display system (System 527)

Hot reactor shutdown trip system (5162)

System 5162 has the task of shutting down the reactor automatically after incidents which briefly make the plant unable to sustain further operation above the residual power level. The following events trip system 5162:

- Manual trip in the control room,
- Excessive reactor power during power operation as indicated by the neutron flux monitoring system (System 531),
- Excessive reactor power rate of increase during reactor startup indicated by signals from the measuring channels K711-714 and K715-719 of the neutron flux monitoring system,

- Excessive reactor outlet temperature indicated by signals from the measuring channels K501-504 of the reactor, reactor pool and concrete vessel monitoring systems (System 536)
- Excessive PCRVR pressure signals from the measuring channels K101-104 in the pressurizer system (System 328)
- Excessive pressurizer pool level, signals from channels K401-404 in the reactor pool water replacement system (System 325),
- Tripping of the turbine (System 421) causing the turbine protection system (System 592) to block bypassing of steam (System 413) to the turbine condenser (System 441),
- Tripping of the isolation trip system (System 5163)

System 5162 initiates the following actions when tripped:

- Initiates rapid reduction of the feedwater flow rate to the steam generators by throttling of the steam supply to the feed pump turbine drive (System 463),
- Interrupts the power supply to the filtered and distilled water injection pumps in the reactivity control system (Systems 3121 and 3122),
- Initiates injection of borated water into the primary loop by the pumps in the reactivity control system (System 3123)
- Trips the hot reactor shutdown alarm in the alarm display system (System 517)
- Trips a plant reactor shutdown survey by the safety parameter display system (System 527)

Isolation trip system (5163)

In case of pipe rupture in the plant, system 5163 limits the loss of reactor pool water and releases to the environment by closing isolation valves.

The system is tripped by the following:

- Low-low reactor pool level signals from the measuring channels K401-404 in the reactor pool water replacement system (System 325),
- Excessive high radiation level in the half-hour delay offgas system (System 3412),
- Excessive pressure in the main steam lines (411) as indicated by signals from the measuring channels KX111-112 (KA, KB, etc),
- Excessive high radiation in the main steam lines indicated by signals from the steam generator tube rupture monitoring system (System 554), although this may eventually trip isolation of the line involved.
- Excessive temperature or pressure in the reactor containment indicated by signals from the containment monitoring systems (System 583),
- Excessive water level, temperature, or pressure in reactor building rooms, or
- Signals from pipe break monitoring systems (System 546) measuring channels

System 5163 initiates the following actions when tripped:

- Closure of isolation valves in the reactivity control, the reactor pool water replacement, the pressurizer, the steam line, and other systems (Systems 312, 325, 328, 411 etc),
- Trips a subsystem in the turbine protection system (System 592),
- Trips the hot reactor shutdown trip system (System 5162), or the reactor system (System 5161),
- Trips an isolation alarm in the alarm display and alarm systems (Systems 517 and 843), and
- Trips a plant isolation survey by the safety parameter display system (System 527)

Startup interlock system (5165)

System 5165 prevents reactor criticality during certain stages of plant refueling and service shutdowns. It is tripped by:

- Manual trip in the main control room, or
- Low pressure in the reactor vessel as indicated by signals from the measuring channels K101-104 in the pressurizer system (System 328).

POD: Plant Operation Desk
 ROD: Reactor Operation Desk
 TOO: Turbine Operation Desk
 AOD: Auxiliary Operation Desk
 OIP: Overview Indication Panels

Figure A.7-1. Alarm Display System

System 5165 initiates the following actions, when tripped:

- Interrupts the power supply to one main coolant pump (System 313/649),
- Interrupts the power supply to the filtered and distilled water injection pumps in the reactivity control system (Systems 3121 and 3122)

A.7.2 Alarm Display System

System 517 in PIUS plants is similar to the alarm annunciator panels of older nuclear plants. The PIUS system is a pure software system which selects signals for display on the video display units (VDUs) of the control room man-machine interface system (System 523).

System 517 (non-safety grade) displays visual and acoustical alarm signals for alarm-tripped process variables.

System design and operation

The signals which trip alarms in the plant, come from contacts associated with various plant process systems. Signal processing and display are performed by the plant computer systems (System 520). All alarms are assigned to a specific system in a specific part of the plant having a specific priority. As indicated by Figure A.7-1, Alarms are displayed automatically on:

- Several VDU units of the man-machine interface system (System 523)
- In the plant control room (System 511),
- On the shift supervisor desk,
- On the relevant plant operator desk, and possibly
- On the overall display panel.

Normally, a group alarm is first indicated on a VDU dedicated to this purpose, and then the operator can request detailed information. All alarms are printed out by typewriter in chronological order or in alarm lists arranged into system groups or plant operation modes.

A.7.3 Emergency Control Room System 518

Safety-grade system 518 facilitates monitoring of the conditions inside the reactor vessel and the reactor containment if the buildings around the reactor building have been destroyed, or when the control building and the central control equipment have been made inoperable.

It consists of the equipment in the emergency control room and cables from the containment and the reactor vessel to the ECR. The equipment in the ECR consists of a scram button that trips one of the main reactor coolant pumps, instruments for indication of temperatures and pressures in the reactor vessel and the containment, and the water level in the reactor vessel. The emergency control room is housed in a separate compartment at the bottom of the reactor building (building 121), beneath the condensation pool (System 316) of the reactor containment (System 152). It is thus physically well protected against external impacts, and since the whole reactor building is designed to withstand seismic events, it is also earthquake proof. The compartment contains half of the safety-related parts of the safety-grade control equipment, e.g. the reactor trip and interlock system (System 516), the pipe rupture monitoring system (System 546), the containment monitoring system (System 583), and the control equipment for different isolation valves. The other half is located in a separate compartment at the other side of the reactor building, diametrically opposite to the emergency control room. Adjacent to these rooms, supply systems are provided in the form of a safety-grade 110 VDC battery-backed power supply system (System 673). The well protected cables, and pipe connections, to system 518 originate with sensors in the reactor pool, the pressurizer part of the reactor primary system, the drywell, and the wetwell of the reactor containment.

A.7.4 Main Process Computer System 521

System 521 performs all the tasks that the corresponding systems do in other modern nuclear plants, however the system goes a step further by handling a number of tasks which are not computerized in current nuclear plants because the tasks are safety-related. In PIUS, the self-protective process features allow the relegation to lower protection levels.

Non-safety grade system 521 provides data processing, data storage, and information about plant conditions. It consists of:

- Processing equipment,
- Storage equipment, and
- Maintenance equipment

as shown in Figure A.7-2. The data processing equipment consists of two redundant sets of mini-computers, each with a front end computer and a process computer. The central unit is the process computer, which processes data from the plant and performs calculations. It communicates directly with the control room man-machine interface and the core management computer systems (Systems 523 and 525), and with the FP control processors in the control processors system (System 524) and various other systems in the plant.

The storage equipment consists of a number of auxiliaries to the two sets of mini-computers such as:

- Floppy disk memories,
- Hard disk memories,
- Magnetic tape memory equipment, and
- Printers which have recording tasks.

The maintenance equipment consists of service stations with keyboards, alpha-numeric display units and some computer capability.

Operating situations

During normal plant operation, both computers of system 521 receive information from the plant control processors system (System 524). However, only one of the two sets is operative in the sense that it delivers information to the plant systems and to the control room man-machine interface system (System 523). If the operating computer set fails, the stand-by unit takes over its tasks automatically. During normal plant operation, system 521 provides the following kind of information:

- General directives for the plant process systems
- System 523, input for displays in the man-machine interface system
- Printouts as ordered by the plant operators or by routines in system 521.

After abnormal occurrences in the plant, system 521 provides the following additional information: Event recording print-outs Input for the safety parameter display system (System 523)

A.7.5 Control room man-machine interface system - 523

System 523 is similar to the corresponding systems in other LWR plants. It transfers exchanging information between the plant systems and the plant operators by:

- Displaying fault alarms automatically, and on request,
- Displays process information to the operators
- Transmits directives from the operators to the main computer system (System 521)
- Enables process control and supervision of systems and objects via the control processors system (System 524).

Figure A.7-2. Main Process Computers Block Diagram

It consists of:

- Overall display panel, described in the main control room chapter (System 511),
- VDU screens, as outlined for the main control room and for the alarm display system (Systems 511 and 517),
- Operator keyboards and tracker ball equipment,
- Printers, which may belong to the main computer system (System 521), and possibly plotters.

The keyboards enable the plant operator to select different functions and pictures for display on VDUs. The tracker ball controls a cursor on a VDU so the operator can select computer functions on the main computer system (System 521), or a process object for manual actions by positioning the cursor on the process display picture (System 524).

A.7.6 Control processors - system 524

Specifies the process control modules which are part of a number of systems in the plant and which act as intermediaries between control room man-machine interface equipment (System 523) and process systems as well as signal processors for process control in a number of plant systems.

The control processors, called functional processor (FP), consist of the following main parts:

- Tandem microcomputer system with floppy disk mass memory,
- Serial data communication links between systems served by the FP, the main computer system (System 521), and the control room man-machine interface system (System 523),
- Process interface units (PIUs), and
- Software for the computer and the interface systems.

The use of FP and PIU units for process control and as links between the process and the man-machine interface equipment as well as the main computers is described under system 542. The FP units can each handle more than 100 process objects, always have a redundant unit as standby which automatically takes over all tasks. The process interface units (PIUs) between the FP units and the plant process systems belong to the respective process system.

A.7.7 Core management computer system - System 525

Is similar to the corresponding system in the most modern BWR plants and some PWR plants. It does the following:

- Supervises the core operation online, and
- Performs fuel management calculations offline.

System 525 shown in block form in Figure A.7-3, consists of two redundant minicomputers with the required memory, display, software, printers and offline data processing.

The two tasks of the system are to receive information from neutron flux monitoring, reactor, reactor pool and concrete vessel monitoring systems (Systems 531 and 5363) via the main computer system (System 521) to calculate:

- Core power distribution,
- Core dryout margin,
- Fuel burnup distribution,
- Isotopic content,
- Optimum reloading schedule, and
- Optimum replacement fuel enrichment.

It uses a three-dimensional model of the core that is currently the most advanced system of its kind in the world.

Figure A.7-3. Core Management Computer System Block Diagram

A.7.7 Safety parameter display system - system 527

This non-safety grade system is standard, original equipment and is an extension of the plant computer system. Its purpose is to assist the plant operators during abnormal plant operations in evaluating the plant safety status for nuclear safety and protection of major equipment. The system can also serve as a plant training simulator.

It consists of is a shared micro-computer with display on VDUs in the main control room (System 511), and possibly in the plant security center (Systems 858). The software is modeled on the software of the more extensive SPD systems for ABB Atom's BWR plants with a larger computer, the system can be used to train plant operators in handling plant emergencies.

Operating situations

During normal plant operation, system 527 collects information; when something abnormal occurs, of safety or plant protection significance, the system automatically assumes its SPDS function. It synthesizes plant and system status vectors for display on VDUs in the control room (System 511) to present:

- Safety status of the whole plant
- Operating status of plant systems which are important to fuel integrity and the safety of the public and the plant capital investment.
- Trends in plant and system behavior and forecasts of future situations.

A.7.8 Reactor and reactor pool control and monitoring systems - System 530

System 530 consists of:

- 531 Neutron flux monitoring systems
 - 5311 SRM system
 - 5312 IRM system
 - 5313 PRM system
 - 5314 TIP system
- 533 Water chemistry control systems
- 534 Main coolant pump control system
- 535 Reactor power control system
- 536 Reactor, reactor pool and concrete vessel monitoring system
- 5361 Reactor outlet water temperature monitoring system
- 5362 Reactor inlet water temperature monitoring system
- 5363 Fuel assembly outlet water temperature monitoring system
- 5364 Reactor pool temperature monitoring system
- 5366 Concrete vessel temperature monitoring system
- 5367 Primary loop flow monitoring system

Neutron flux monitoring systems - 531

Safety-grade System 531 corresponds to the neutron flux monitoring system in other LWR plants. The monitoring of the radial power distribution in the reactor core is supplemented by reactor outlet water temperature measurements in the reactor, reactor pool and concrete vessel monitoring systems (Systems 536). It measures the neutron flux in the reactor core and supplies the core management computer system (Systems 525) with information about core usage, gives alarm and scram signals, if the flux or the flux rate of change passes certain limits, and possibly provides input to the reactor power control during plant startup. It consists of four subsystems;

- SRM system (5311),
- IRM system (5312),
- PRM system (5313), and
- TIP system (5314).

The startup and intermediate range monitors (SRM and IRM) systems have four measuring channels each (531 K711-K719) with combined detectors, and the PRM (the power range monitors) system has 16 measuring channels (531 K721-K737) with detectors of their own (LPRM - local power range monitors). The detectors are mounted in wet guide tubes, belonging to the vessel internal instrumentation support equipment (216). They are located in the four quadrants of the reactor core. The traversing incore probe (TIP) system consists of four measuring channels (531 K742- K746) with movable detectors, mounted in dry guide tubes (belonging to system 216), one in each core quadrant. The detectors are of the beta-type, and the measuring channels get their power from the battery-backed a.c. power supply system (System 677). The output from the channels goes to the computer systems (System 520), which handle the protective alarm and trip functions as well as the process and display functions of the system. In addition to these operations-related monitors there will probably be two to four out-of-core detectors (531 K751-K752 (K754)) which provide information to the emergency monitoring station (Systems 518) so that the subcriticality of the core can be verified after postulated severe accidents.

In the startup range, the SRM system measures the neutron flux but will scram the reactor if the neutron flux increases too rapidly as indicated by 2 out of 4 channels. The IRM system performs the same function intermediate power range. The PRM system measures the neutron flux at all times. The LPRM detector signals are grouped into four average PRM (APRM) channels (K738-K741) which provide input to the trip and interlock system (Systems 516) and other systems.

System 516 takes the following actions:

- The reactivity control system (312) gets a correction signal if any two channels show a flux which is 10% above the full power operation,
- The reactor is scrammed if two out of four channels show a flux which is 20 to 25% above full power operation.

Water chemistry control system - 533

Non-safety grade System 533:

- Performs online analysis
- Supervises to assure that prescribed limits are not exceeded
- Controls the lithium hydroxide system (System 3333)

It consists of the following:

- 5331 Boron concentration monitoring system
- 5333 pH control system
- 5335 Hydrogen control system
- 5337 Conductivity monitoring system

System 5331:

- Monitors the reactor pool boron concentration during all plant operating situations,
- Monitor the boron concentration in the primary system, especially during approach to criticality.

System 5331 consists of a vessel, a neutron source and a neutron detector. The boron content of sampling system (System 336) water flows through the vessel. The attenuation of neutrons by the solution is indicative of the boron content of the water, and an alarm is given when the pool boron concentration drops below 2000 ppm.

The safety-grade system has two ranges:

Range	A	B
Limits	0-2500	0-500
Accuracy	± 50 ppm	± 10 ppm

During plant startup, system 5331 monitors the water from the withdrawal line of the reactivity control system (System 3125), first on range A and later on range B.

During normal plant operation, system 5331 monitors alternately the reactivity control system (System 3125) and the pool water replacement system (System 3253) every other day. The monitoring of the pool water replacement system (System 3253) on range A will give assurance that the reactor pool does indeed contain boron in high enough concentration to shut down the reactor.

Tasks requiring a higher accuracy of measurement than system 533 is capable of giving, such as the monitoring of the reactor core burnup, will be handled by the plant laboratory.

pH control system (5333)

System 5333:

- Supervises the pH in the primary system and sounds an alarm, if it exceeds limits.
- Controls the lithium hydroxide system (System 3333)

The pH control of the reactor pool is not performed automatically, but is based on sampling (System 336) and plant laboratory analyses.

System 5333 consists of the following:

- Two redundant on-line pH measuring systems, and
- A lithium hydroxide dosage control system.

The pH measuring systems measure the pH of water withdrawn by the sampling system (System 336) from the reactivity control system (System 3125) using standard equipment, but at low temperature, therefore the control system must calculate the primary system pH using temperature information provided by local temperature monitors that are part of the system. The control system takes information from the following sources: pH measuring system, reactivity control system (System 312) pump flow rate measurements, the plant power overall control system (System 591) mode selector and adjusts the pH by controlling the lithium hydroxide system (System 3333) pumps.

Concentration control system (5335)

Performs the following tasks:

- Supervise the hydrogen concentration in the primary system and sounds the alarm if it passes limits,
- Controls the injection of hydrogen, when needed.

The hydrogen concentration of the reactor pool water is not controlled automatically but is adjusted based on sampling and plant laboratory analyses.

System 5335 consists of the following main parts:

- Two redundant on-line hydrogen measuring systems, and
- Hydrogen injection control system.

The hydrogen measuring system uses sampling system (System 336) water from the reactivity control system (System 3125). The control system takes information from the following sources:

- The hydrogen measuring system
 - The reactivity control system (System 312) pump flow rate measurements, and
 - The plant power overall control system (System 591) mode selector switch
- to control valves in the hydrogen injection lines.

Conductivity monitoring system (5337)

Has the task of monitoring the conductivity of the water in some systems, particularly after the filters in the condensate cleanup system (System 467). It consists of a number of conductivity measuring cells, located in hoods in the sampling system (System 336). Whether the system will shut the plant down or just give a warning has not been determined.

A.7.8 Main coolant pump control system (534)

Non-safety grade System 534 is similar to the corresponding system in ABB Atom's BWR plants, since the pumps are very similar. However, the control function is quite different. The system controls the speed of the main coolant pumps of the recirculation system (System 313) via the main coolant pump power supply system (System 649) in such a way that the reactor pool loop is not activated during plant operation. System 534, shown in Figure A.7-4 consists of:

- An FP microcomputer processor (System 524), which it may share with some other system,
 - A set value generator for the location of the hot-cold interface in the lower density lock (System 3221),
- and
- A set value generator for determining pump speed as a function of reactor outlet temperature.

System 534 gets inputs from:

- The plant power overall control system (System 591) which gives the operating mode selection orders,
- and
- The temperature interface monitors 322 K501 through 503 in the lower density lock system (3221) of the reactor, and
 - Current mixed reactor outlet temperature data from the detectors 536 K 501 through 504 in the reactor, reactor pool and concrete vessel monitoring system (System 536).

System 534 has three operating modes:

Mode A, plant operation. System 534 uses the hot-cold interface set value generator and controls the pump speed to maintain a constant interface position.

Mode D, plant startup from cold shutdown. System 534 controls the pumps at constant low speed (the minimum speed level). When the reactor outlet temperature has increased to about 100°C, the pump control system switches to mode E.

Mode E, hot standby mode. System 534 utilizes the pump speed set value generator and the hot-cold interface set value generator to control the pump speed and maintain a constant interface position.

Plant startup from cold shutdown

During plant startup from cold shutdown, the pumps are brought to 20% of full speed on manual order, and the gas lock (System 3225) is established in the lower density lock region. System 534 then takes over the coolant pump control and operates in mode D. The boron concentration in the primary loop is reduced by dilution to reactor criticality and subsequent nuclear heating. A small continuous flow of gas to the gas lock keeps the lock active and the reactor pool loop inactive.

When the reactor riser temperature has reached about 100°C (corresponding to 20% speed of the main coolant pumps), system 534 is switched to mode E and the pump speed is increased, to keep the reactor pool loop inactive, as the reactor riser temperature increases. The pump speed set value generator transforms the riser temperature measurements into a pump speed control signal, and the hot-cold interface set value generator provides a corrective signal for pump speed adjustment so that the interface position will be maintained within close limits.

When normal operating temperature has been attained in the reactor riser, System 534 switches to mode A and the pump speed control signal is supplied by the hot-cold interface set value generator.

Figure A.7-4. Main Coolant Pump Control System

Plant steady state operating conditions

System 534 operates in mode A under the following plant conditions:

- Full power operation (plant overall mode A)
- Part load operation (plant overall mode A), and
- Changing power requirements (plant overall modes B and C)

This means that the coolant pumps are operating at almost full speed during all these conditions. At hot standby or other situations when the primary loop operates at reduced temperature, system 534 will operate in mode E (plant overall mode E) and the pump speed is adjusted to the actual reactor riser temperature by the pump speed set value generator. The hot-cold interface set value generator performs the fine control function to maintain the interface at the correct position during both mode A and E operating conditions. During cold shutdown, system 534 is shut off manually.

Plant transients

During non-shutdown transients, the pool loop inactivations system acts as a buffer to prevent borated pool water from entering the primary loop. System 534 remains in mode A, but, in certain situations, in mode E resulting in the coolant pumps trying to counteract the transient effect on the lower density lock.

It is important to plant safety that this action does not prevent activation of the pool loop when such is required. The RIGEL code analysis including this effect has yet to find a detrimental influence.

Malfunction

Usually a failure in System 534 result in activation of the pool loop. However, under sabotage, during a positive power transient, System 534 must not command faster than the limit value of the pump speeds or the transient could lead to local core dry-out. It is presently envisaged that such a malfunction shall be prevented by having System 534 give a "negative" control signal to the pump power supply system (System 649), which is zero at a pump speed which is a few percent higher than the normal operating point.

A.7.9 Reactor power control system - System 535

System 535 is equivalent to the corresponding system in other LWR plants, although the operating modes and specific tasks in various plant operating conditions differ. System 535 controls the reactivity control system (System 312), and:

- During plant power operation and hot standby controls the reactor power output,
- During plant startup controls the reactor power rate of change, and
- During plant power operation, hot standby and hot shutdown, controls the hot-cold water interface level in the upper density lock (System 3222).

A schematic logic diagram for system 535 is shown in Figure A.7-5. The hardware consists of a shared FP (System 524) micro-computer processor. It receives the following inputs:

- Operating mode selection orders from the plant power overall control system (System 591),
- Mixed reactor outlet temperature set value from system 591,
- Current mixed reactor outlet temperature data from the reactor, reactor pool and concrete vessel monitoring system (System 536) measuring channels K501-505,
- Reactor inlet temperature data from the reactor, reactor pool and concrete vessel monitoring system (System 536) measuring channels K511-512,
- Reactor neutron flux data from the neutron flux monitoring system (System 531)
- Upper density lock temperature data from the pool loop inactivation system (System 322) measuring channels K521-522,
- Reactivity control system flow rate feedback data from the reactivity control system (System 312) (K 311, K 321, K 331 - 332, K 343 - 344)

System 535 provides the following outputs:

- Flow rate control to the injection pumps in the reactivity control system (System 312), and the control valves in the system, and
- Withdrawal rate orders to the valves in the withdrawal lines of the reactivity control system (System 3125)

Operating modes

The operating modes are:

- Power operation mode

In mode A, system 535 keeps the reactor outlet temperature constant at 290°C. It receives the set value from the plant overall control system (System 591) and the present value from the reactor, reactor pool and concrete vessel monitoring system (System 536) measuring channels K501-505. The reactivity control system (System 312) is ordered to pump a constant flow rate of filtered water from the filtered water buffer tank in the reactor water cleanup system (331 T11) with small additions of distilled or borated water from the storage tanks for such water (from 331 T22 or T23) to compensate for core burnup and inleakage of pool water to the primary system. The withdrawal system (System 3125) is ordered to withdraw an amount equal to that supplied by the water supply systems (Systems 3121 - 3123) via the inlet-outlet system (System 3124). The set values for supply and withdrawal rates are set manually by the plant operator to meet the cleanup requirements of the reactor water.

- Power decrease mode

When the plant power is to be decreased, the feedwater flow rate is decreased, resulting in a reduced steam generation rate. The reactor water inlet temperature increases as does the reactor outlet temperature. The average primary loop water temperature thus rises which in turn reduces the reactor power due to the negative moderator temperature reactivity coefficient, and results in an expansion of the water volume. In mode B, System 535 orders the reactivity control system (System 312), to stop pumping filtered water (subsystem 3121) into the primary loop, and orders subsystem 3125 to withdraw water at maximum rate which in turn calls for switching to the withdrawal line II in the inlet-outlet and withdrawal systems (Systems 3124/3125), or possibly to activate a temporary discharge to the condensation pool (System 316).

In the event of power changes beyond the 30% control range, subsystem 3123 is ordered to pump borated water into the primary loop at high rate to restore the reactor outlet temperature, i.e., reducing the reactor power to balance at a lower temperature. The mode is terminated automatically by the plant power overall control system (System 591) when the reactor outlet temperature is close to normal again and or the temperature monitors K 531 - 540, etc. in the upper density lock (System 3222) signal that the location of the upper density lock hot-cold interface is high.

- Power increase mode

When the plant power is to be increased, the feedwater flow rate is increased, resulting in decreased reactor water inlet temperature and reduced reactor outlet temperature resulting in a shrinkage of the water volume. The average primary loop water temperature thus is reduced which in turn increases the reactor power due to the negative moderator temperature reactivity coefficient.

In mode C, subsystem 3121 of the reactivity control system (System 312) continues pumping filtered water into the primary loop, and the inlet-outlet system (subsystem 3124) is switched to inlet line II. Simultaneously, subsystem 3125 is ordered to stop withdrawing water. In the event of power increase beyond the 30% control range, subsystem 3122 is ordered to pump distilled water into the primary loop at high rate to restore the reactor outlet temperature, i.e., increasing the reactor power to balance at a higher temperature. The mode is terminated automatically by the plant power overall control system (System 591) when the reactor outlet temperature is close to normal again, and/or the temperature monitors K 531-540 in the upper density lock (System 3222) signal that the location of the upper density lock hot-cold interface is low.

· Plant startup modes

These modes are for the time being assumed to be manually controlled and GOBLIN calculations indicate that there is no problem during the passing of criticality. If the heating of the primary system is too rapid for manual control, automatic modes may replace the manual mode. Such modes could for instance be specified as follows:

- In the precritical mode, D1, system 535 could control the rate of change of the reactor neutron flux. The system would then utilize a neutron flux period set value generator which would belong to system 535 and receive present period values from the neutron flux monitoring system (System 531).

- In the system heating mode, D2, system 535 could use heating rate information from the mixed reactor outlet temperature measuring channels K501-505 of the reactor, reactor pool and concrete vessel monitoring system (System 536) and control the reactor primary loop water temperature rate of change. The plant power overall control system (System 591) will switch to mode A when the reactor outlet temperature approaches the operating region, leaving system 535 in operation mode A.

- In hot standby mode, E, system 535 controls the reactor power to maintain a constant reactor outlet temperature in the same way as in mode A. The reactor power is kept at a very low level (only about 1 %), determined by the heat losses to the reactor pool and in the steam generator loops.

- In shutting down mode, mode F, system 535 initially orders the reactivity control system (System 312) to pump borated water at full flow from subsystem 3123. The primary loop temperature decreases, i.e. the water volume shrinks, and subsystem 3125 is ordered not to withdraw any water.

Upper density lock (System 3222) restoration mode

In mode R, system 535 will order the withdrawal subsystem (System 3125) of the reactivity control system to stop withdrawing water temporarily. When the hot-cold interface position has reached the very low level, the mode is terminated and subsystem 3125 withdraws water at high rate to reestablish the normal position.

Steady state power operation

During normal steady state plant operation, system 535 operates in mode A and controls pump rates and control valves in the reactivity control system (System 312) in such a way as to keep the reactor outlet temperature constant at 290°C. If a detailed analysis should indicate a need for special measures to control the high - low boron interface in the upper density lock (System 3222), the mode R outlined above may be implemented. At intervals, system 535 may then switch to mode R, i.e. ordering a lowering of the hot-cold interface, and a subsequent restoration of the interface level. This will take about 10 minutes, and after that, system 535 returns to mode A.

The mixed reactor outlet temperature set value for mode A is currently 290°C. It may be necessary for the reactor outlet temperature set value to vary during the cycle. The basic criterion is that the core hot channel outlet temperature shall be at saturation for the 9.0 MPa pressure of the concrete vessel i.e. there shall be no net boiling at the core outlet.

Changing power requirements

When the plant power requirements change, the feedwater control system (System 598) changes the feedwater flow rate. This causes:

· The steam generator outlet steam pressure to change which is handled by the turbine control system (System 597).

· The reactor inlet temperature to change, and in order to keep the reactor outlet temperature close to 290°C, system 535 has to change the reactor power. System 535 goes into mode B or C.

For small normal change rates, system 535 can remain in mode A and the composition of the make-up water changes somewhat. The flow and withdrawal orders to the reactivity control system (System 312) during power changes are caused by the changes of primary loop water volume that accompanies the changes in temperature drop across the reactor as discussed for system 537.

Plant startup

During plant startup from cold shutdown, system 535 operates in manual mode D. Limitations are imposed by the neutron flux period information from the neutron flux monitoring system (System 531) and possibly by the reactor outlet temperature time constant information from the reactor, reactor pool and concrete vessel monitoring system (System 536), subsystem 5361, reactor outlet water temperature monitoring system.

If it is found that these changes are too rapid for manual operation, the instruments which give the limiting factors may instead be made into actual value generators for automatic operation. Mode D may then be split up as indicated under "System operating modes".

Shutting down, shutdown, and hot standby

When the reactor is to be shut down but not scrammed, the plant power overall control system (System 591) puts system 535 in mode B, with borated water being pumped into the reactor primary loop at full speed, reducing the reactor power to residual level after 15 to 30 minutes, depending on the operating situation. If the reactor is to remain at hot standby, system 591 switches system 535 to mode E when the hot standby power level has been reached. If the reactor is to be shut down to the residual power level, the mode switching is instead made to mode F, when the neutron flux monitoring system (System 531) shows that the reactor is sufficiently sub-critical.

At hot standby, system 535 has the task of making the reactor produce enough power to maintain the pressure in the steam line at 4.0 MPa.

Pool loop activation

When the reactor is scrammed or the pool loop is activated, system 535 goes into mode F. This makes the reactivity control system (System 312) pump water at full speed into the primary system, in order to make up for the shrinking of the volume of the total water mass in the concrete vessel. No water is withdrawn by the subsystem 3125.

A.7.10 Reactor, reactor pool, and concrete vessel monitoring systems - System 536

System 536 differs from corresponding systems in present day nuclear plants in the following ways:

- The reactor pool adds some extra tasks to the system,
- Core power distribution monitoring tasks are added to one system tasks, and
- Only parts of the systems are safety-related

System 536 measures and monitors several variables in the reactor primary and pool systems and the concrete vessel, which are not easily assigned to process systems.

Safety-related system parts

For systems 5361, 5362, 5364, and 5366 the following parts are treated as being safety-related:

- The parts inside the concrete vessel,
- The parts which supply the emergency control room (System 518) with information,

Non safety-related system parts:

- The parts that supply the main control room (System 511) with information, and
- The main control room displays etc.

Other common system design features

In order that pipe ruptures outside the concrete vessel shall not be able to drain the reactor pool, the following requirements are imposed:

- The concrete vessel penetrations are located in the top of the vessel above the level where the embedded membrane ends.
- The guide tube interior communicates with the 9.0 MPa water somewhere in the reactor pool above the level defined above.
- Instrument probes are designed in such a way that they cannot drain water at 9.0 MPa.

The guide tubes for detectors inside the reactor core envelope structure enter through penetrations in the concrete vessel upper part and pass along to the measuring position. Those guide tubes for detectors in the lower density lock pass down the concrete vessel cavity liner across the concrete vessel bottom and up to the measuring position and is part of the internal instrumentation support equipment system (System 216). The temperature detectors are resistance thermometers where this is possible, otherwise they are thermo-couples. The output from all 536 instruments is via the control processors (System 524) to be input to the main process computer system (System 521) for storage and possibly logged and typed out. Some instrument outputs are used for process system control and for trip functions etc. The measurement channels are treated in separate control processors, and those related to the reactor pool and the concrete vessel in other processor units.

Reactor outlet water temperature monitoring system (5361)

Subsystem 5361 consists of five detectors: (536 K501-505) located in the reactor riser high enough above the reactor to provide good mixing of the outlet water from the fuel assemblies in the reactor core. Four detectors give scram signal information to the trip and interlock system (System 516). They also give the input signal to the main coolant pump and the reactor power control systems (Systems 534 and 535). One detector is a spare.

Reactor inlet water temperature monitoring system (5362)

Subsystem 5362 consists of two detectors: (536 K511-512) detectors are located in the reactor lower plenum to give inputs to the reactor power control system (System 535) control functions and to the computer system (System group 520) for various trip functions.

Fuel assembly outlet water temperature monitoring system (5363)

Subsystem 5363 consists of ten to twenty detectors: (536 K521-530 to 540) located at the outlet of coolant flow guide tubes from selected fuel assemblies in the reactor core. The probes are entered from above through the reactor core envelope structure. The 5363 detectors monitor the margin to net boiling in the reactor core. Their output is compared by the core management computer system (System 525) with the calculated temperature at the detector locations, and this comparison is utilized to establish:

- The current radial power distribution in the core,
- The current margin to net boiling in the hot channel

Reactor pool temperature monitoring system (5364)

Subsystem 5364 consists of ten detectors (536 K 541 - 550) located in such a way that they can monitor:

- Hot conditions in the pool upper parts
- Layering effects in the reactor pool
- Malfunction of the reactor pool cooling system (System 323)

Increased temperature in the reactor pool upper parts will cause automatic startup of the standby pump in system 323 and possibly the standby pumps in the diesel-backed cooling systems (Systems 721 and 712), if this is needed to keep the temperature down in these systems.

Concrete vessel temperature monitoring system (5366)

System 5366 has not been designed but is expected to consist of about ten detectors (536 K 561 - 570) located behind the vessel liner in the upper vessel parts where hot water will collect when the reactor pool loop is activated.

Primary loop flow monitoring system (5367)

System 5367 has the task of measuring the flow rate in the reactor primary loop for information purposes. System 5367 consists of a venturi measuring system located in the cold leg inlet nozzles of the reactor recirculation system (System 313). Four differential pressure transmitters are connected to these venturis, one at each nozzle.

A.7.11 Pressurizer pool level control system 537

Non-safety grade System 537 is similar to corresponding systems in present day PWR plants. Its purpose is the controlling the pressurizer pool (System 328) water level and hence the reactor pool water level, via the reactor pool water replacement system (System 325). System 537 consists of a shared FP (System 524) micro-computer processor and associated software.

The system receives information from:

- The plant power overall control system (System 591) about plant operating modes, and
- The instruments in the reactor pool water replacement system (System 325) and possibly in other plant systems.

System 537 controls the pumps P1 and P2, the control valves V33 and V34, and the valves V37 and V38 in the reactor pool water replacement system (System 325). The system may also control the reactivity control system (System 312) via the reactor power control system (System 535) in those cases, where the pool water replacement system (System 325) needs help from the reactivity control system (System 312) to counteract rapid changes in the pressurizer pool surface level.

System operating modes

System 537 and thus the reactor pool water replacement system (System 325) has the following operating modes:

- Normal plant operation

This mode is ordered by the plant power overall control system (System 591), when this system is in mode A. Pipe A is used in the reactor pool water replacement system (System 325). The flow rate within the pipe A range is ordered by the plant operator on grounds of pool cleanup needs. One pump operates in the system and controls the pressurizer pool level.

- Power decrease

This mode may be ordered by the plant power overall control system (System 591) in mode B, but more likely by the reactor pool water replacement system (System 325) instrumentation. Pipe B is used to counteract swelling of the reactor pool volume, if the reactivity control system (System 312) cannot prevent primary water spills over into the reactor pool. The pumps in the reactor pool water replacement system may be stopped.

- Power increase and reactor scram

This mode may be ordered by the plant power overall control system (System 591) in mode C but more likely by the reactor pool water replacement system (System 325) instrumentation. The withdrawal of pool water is stopped and both pumps may operate briefly in order to counteract the decrease in volume of the total reactor water mass. Mode C in system 537 is also ordered by the trip and interlock system (System 516) or the plant power overall control system (System 591) at reactor scram. The withdrawal of reactor pool water is stopped and the water injection subsystem will operate with both pumps at full speed for some time to

compensate for the considerable contraction of the reactor pool water volume. F Cold shutdown Mode F is a manual mode for system 537.

A.7.12 Reactor pressure control system 538

Non-safety grade System 538 is equivalent to the corresponding systems in present day PWR plants. Its purpose is to keep the pressurizer steam dome (System 3281) pressure constant. The system also controls the blow-down of the pressurizer offgassing system (System 3285), when the steam dome is purged of noncondensable gases.

System 538 consists of a shared FP micro-computer processor (System 524), and the associated software. It receives information from the following sources:

- The plant power overall control system (System 591) about plant operating modes, and
- Instruments in the reactor pool water replacement system (System 325) and the pressurizer systems (System 328), and possibly other plant systems.

System 538 controls the following systems and components:

- The switchgear in the pressurizer heater power supply system (System 648),
- The control valve V513 and shut-off valves V511, V521 and V522 in the pressurizer offgassing system (System 3285), and
- Control and shut-off valves for a pressurizer spray system.

A.7.13 Process control and measurements systems 540

Systems 540 consists of the following subsystems:

- 541 Process measurement principles
- 542 Process control principles
- 543 Process operation principles
- 545 Leak detection principles
- 546 Pipe rupture monitoring systems

Process measurement principles 541

System 541 is similar to the corresponding system in ABB BWR plants. This section establishes the principles, but does not specify the performance and scope of measuring equipment in a process system. These principles apply to reactor plant process measurements involving limit switches and the measurement of quantities such as pressure, level, flow, temperature, etc., which are used for status indication, control purposes, alarms or initiation of automatic operations in process systems of the plant. Measuring a process variable and converting the measured value to the required form requires:

1. A transducer which converts the process variable (e.g. temperature) to a quantity of an electrical character (e.g. resistance),
2. A transmitter which converts the output signal of the transducer to a standardized electrical signal (e.g. 4-20 mA).
3. An analog input circuit board for connection to programmable control equipment for signal processing.

Equipment consisting of such components for measuring a process variable is called a measuring channel. All signal processing (such as limit checking) in the measuring channels is performed with programmable equipment.

The process values are presented on CRT display units in the central control room. By means of portable plug-in units, the values may be displayed locally at the electronics cubicles.

Important process parameters are recorded and stored in the computer system. On request, or continuously, the trend of such parameters can be shown on the CRTs and the values printed. The measuring equipment is standardized both electrically and mechanically. The following methods are normally used for measuring different process parameters:

- Liquid levels are mostly measured according to the hydrostatic method. Level monitors are mostly floating

4 of 6

or ultrasonic type switches. Differential pressure and bubble methods may also be used.

- Flow rates are usually measured as differential pressure across orifices, venturis, annubars or by rotameters and inductive sensors.

- Differential pressures are measured by means of differential pressure cells of diaphragm or bellows type.

- Pressures are measured by means of Bourdon tube, diaphragm or bellows type pressure transmitters and switches.

- Temperatures are usually measured by means of resistance thermometers, i.e. resistance elements of the Pt 100 type. Thermocouples replace the resistance thermometers, when the conditions so require.

- Conductivity in water is measured with common types of conductivity sensors.

- Pump or fan speeds are measured by tachometer generators or pulse frequency measuring devices.

The transducers, transmitters, switches etc. are located in special rooms or protected areas, separated from areas with high potential for radiation, in order to shield them from erroneous signals and make them easily accessible for maintenance, testing and calibration. The output from transducers and transmitters will normally be 4-20 mA d.c., but 0-20 mA or 0-10 VDC may also occur. These analog signals are transmitted to an electrical apparatus room, using twisted and shielded cables.

Process control principles - 542

The control systems structure has five operating levels as shown in Figure A.7-5:

- Signal conditioning equipment
- Process interface equipment
- Signal processing equipment
- Supervising equipment
- Operator interface equipment

The equipment for communications between these five levels is subdivided into:

- Process communications equipment
- Inter-computer communications equipment

Signal conditioning equipment

The signal conditioning equipment transforms the output from the various instrument systems to a standard signal, for instance, 0-10 V d.c. These signals are then inputs to process interface equipment, PIUs. PIUs also described in the control processors system chapter (System 524), is based on standardized electronic plug-in units. These PIUs contain control logics and signal input and output circuits, which are designed to interface with contactors, actuators, transmitters and other control systems and measuring system hardware via signal conditioning units. One PIU can handle either one complete control function for a process object such as a pump drive or a number of analog or digital inputs and outputs, for instance, measurements channels. The PIUs are mounted in racks and cabinets and are distributed in electrical rooms throughout the plant. Each control PIU has the ability to serve as an independent control station and does not necessarily need connections to a higher control level. Manual control of a pump or a valve can be carried out locally from its PIU.

Signal processing equipment

The signal processing level control equipment is mostly non-standard. Examples are interlock and automatic control systems and generation of process pictures to be used for display and manual control via the operator interface systems. Automatic operations, closed-loop control, manual control and information display are

Figure A.7-5. Reactor Power Control System

all carried out by control processors (FPs). Automatic sequences by systems that have importance for plant safety or for protection of major equipment, are initiated through these FPs which are described in the control processors system chapter (System 524).

Supervising equipment

The main objective of the supervising level is to assist the plant operators by providing information related to the plant operation. Current and historical measurements can be presented on demand, or as a trend. The alarm display system functions (System 5171) and generation of various reports are also performed at the supervising level.

Operator interface equipment

The operator interface equipment for the main control room has been described in system 523.

Process communication equipment

The exchange of information between the FPs and the PIUs is performed by means of serial communication links using conventional cables, each link serving several PIUs. All PIUs are connected to both of their two redundant FPs, which means that a single link failure will not lead to loss of the control.

Inter-computer communications equipment

The communication between FPs and between FPs and computers in the main computer system (System 521) is performed by means of serial links based on the HLDC (High Level Data Control) standard.

Leak detection principles

PIUS 600 has out-of-vessel coolant loops with steam generators and coolant pumps, as well as other system pipes carrying warm, pressurized primary system water, similar to present day PWR plants. The integrity of these parts of the reactor coolant pressure boundary is not as important from the core protection point of view as in other PWR plants consequently the requirements on leak detection can be eased compared with those of USNRC Regulatory Guide 1.45. Leak detection detects and monitors leakage of reactor coolant into the containment area, and other leakages that may affect the protection of the reactor core against overheating and damage, or affect the environment.

The PIUS plant does not have a specific leak detection system, but these functions are performed by other systems. Leakage in the concrete vessel liner is monitored by the concrete vessel venting system (System 159). The vent lines are small pipes thus limiting the outflow. Leaks in the piping and fittings of the main coolant loops (System 313), the reactor pool cooling loops (System 323), the pressure relief system (System 314), the warm portions of the reactivity control system (System 312) and the pool water replacement system (System 325) are detected by the containment monitoring systems (System 583). Leaks in the coolers and heat exchangers of the reactor pool cooling system are detected by equipment in the diesel-backed closed cooling system for the reactor pool (System 721). Leaks in the steam generators are detected by monitoring of the radioactivity level on the secondary side by the steam generator tube rupture monitoring system (System 554), by the half-hour delay offgas system inlet radiation monitor (System 552) and by the stack radiation monitors. Leaks in the cold portions of control system (System 312), the pool water replacement system (System 325), and the reactor water cleanup system, as well as other systems carrying primary system water, are detected and monitored by the pipe rupture monitoring system (System 546) or the other process monitoring system (System 547). Leaks in stuffing boxes for valve stems etc. are detected and monitored by the controlled drains system (System 338).

Pipe rupture monitoring systems - 546

Safety-grade System 546 is similar to the corresponding system in ABB Atom's BWR plants, but considerably smaller, since PIUS has fewer and smaller systems with primary water outside the primary volume of the containment than ABB Atom's BWRs. System 546 has the task of detecting pipe fractures and ruptures in systems with primary system water and to initiate their isolation. The system monitors the steam and feedwater lines compartment outside the reactor containment as well as the turbine building proper to detect fractures and ruptures in the steam lines (System 411) and the feedwater lines (System 463) and to initiate reactor scram via the trip and

interlock system (System 516). It monitors rooms containing hot water by detecting excessive temperature, temperature and/or pressure change. Rooms containing cold pressurized water are monitored by means of pressure or pressure change measurements and/or water level transducers located in the monitored floor drain system (System 345). The following sets of measuring channels are assumed for the system:

- Four pressure monitors (KA101, KB101, KA102, KB102) which are located in the rooms containing the pumps of the reactivity control system (System 312) and pool water replacement system (System 325), as well as the inlet pipes to the reactor water cleanup system (System 331) and its evaporators,
- Four temperature monitors (KA501, KA502, KB501, KB502), located with the above pressure monitors,
- Two pressure monitors (KA103, KB103) and temperature monitors (KA503, KB503) in the steam/feedwater line compartment outside the containment,
- Two pressure monitors (KA104, KB104) and temperature monitors (KA504, KB504) inside the turbine building at a high location, and
- Eight flow monitors (KA301-KA304, KB301-KB304) in the main portions of the monitored floor drains system (System 345).

A.7.14 Radiation monitoring systems - 550

System 550 consists of the following subsystems:

- 551 Extended delay offgas system input radiation monitor
- 552 Half hour delay offgas system input radiation monitor
- 553 Stack radiation monitors
 - 5531 Sampling systems
 - 5532 Stack noble gas monitor
 - 5533 Stack aerosol monitor
 - 5536 Emergency stack monitors
- 554 Steam generator tube rupture monitoring system
- 555 Room radiation monitors
- 556 Portable radiation monitors
- 557 Environmental radiation monitors

Extended delay offgas system input radiation monitor - 551

The PIUS plant offgas systems are more similar to BWR offgas systems monitoring the input to the offgas system, where longer delay times than the statutory 30 minutes (in Sweden) may be needed. Non-safety grade System 551 measures and records the input to the delay chamber of radioactive matter in the extended delay offgas system (System 3411) and gives alarms if certain levels are exceeded. It consists of a gamma-sensitive scintillation detector along the pipe from the reactor water cleanup system (System 331) to the sand tank of the offgas delay system (341 T11) to measure the nuclide Xe-133, but can be used for gamma spectrometry although no local multi-channel analyzer is included. The output is displayed in the main control room and printed. System 551 operates continuously to provide prewarning and release prevention. Samples can also be drawn by the sampling system (System 336) for analyses in the plant laboratory.

Half hour delay offgas system input radiation monitor - 552

Non-safety grade System 552 in PIUS plants is analogous to the system which measures the radioactivity of the turbine condenser gas removal system, in plants where 30 minutes or less of delay of offgas is used. System 552 measures and records the input to the delay line of radioactive matter in the half hour delay offgas system. It consists of a gamma-sensitive detector along the pipe from the condenser gas removal system (System 442) to the delay volume of the offgas delay system (341 T21) to measure Xe-133, Kr-87 or Kr-88. It can be used for gamma spectrometry. The output is displayed in the main control room and

printed. System 552 operates continuously to warn the plant operator at least 30 minutes to act before any radioactive products leave the delay volume (341 T21) and enter the stack.

Stack radiation monitors - 553

Non-safety grade System 553 has the task of measuring and recording the release of air-borne radioactive matter to the plant environment via the stack (System 144). It consists of the following subsystems:

- 5531 Sampling system
- 5532 Stack noble gas monitor
- 5533 Stack aerosol monitor
- 5536 Emergency stack monitors

All of system 553 is located in a room at the main plant stack (System 144).

Subsystem 5531 consists of a fan duct system with a flow rate of about 1 m³/s taking air isokinetically at several locations across the stack cross section and bypassing this air from the stack (System 144) to the noble gas and aerosol monitor systems (5532 and 5533). It is designed to prevent loss of particles in the sampling.

Noble gas and aerosol monitors (5532, 5533)

Subsystem 5532 has a gamma-sensitive detector, which normally measures Xe-133, but can be used for gamma spectrometry with a collocated multi-channel analyzer; it also has an iodine and particulate filter. The 5532 output will be displayed in the main control room and printed by the main computer system (System 521).

Emergency monitors (5536)

System 5536 has two self-powered gamma detectors and two ionization chambers which are inserted into the stack. They measure dose rates in the intervals 10⁵ to 10⁸ rad/h and 1 to 10⁵ rad/h, respectively.

A.7.15 Steam generator tube rupture monitoring system

Safety-grade System 554 gives alarm and optionally trips the isolation of the steam lines (System 411) when steam generators are subjected to tube ruptures or large tube leaks. It consists of three detectors per steam generator, which measure N-16 gamma radiation. The detectors monitor the steam lines (System 411) close to the steam generators and may trip closure of the corresponding isolation valve in a 2/3 arrangement if counting rates exceed certain limits.

A.7.16 Room radiation monitors - 555

System 555 has the task of monitoring radiation dose rates in plant rooms to trip alarms or cause automatic actions. System 555 is mostly not safety-grade but there are some safety-related parts. It consists of four gamma detectors in the reactor service room. They trip the following actions in a 2/4 arrangement after a postulated refueling accident:

- Alarm systems (Systems 517 and 843)
- Startup of the safety parameter display system (System 527)
- Shutting down of the reactor building ventilation system (System 742)
- Startup of the emergency ventilation system (System 749)

There may be some other areas in the plant which will have to have radiation alarm systems for personnel protection.

A.7.17 Containment monitoring systems - 583

System 583 is similar but smaller than the corresponding system in ABB Atom's BWR plants. It monitors the atmosphere in the reactor containment to check the performance of various auxiliary systems during normal operational situations and to detect the occurrence of minor leakage in process equipment, and to initiate safety measures during or after accidents within the containment and to supervise the function of the safety measures. System 583 is generally not a safety-grade system, but there are safety-related parts. It consists of measuring channels divided into four separate and independent parts. The signals from the measuring systems are connected to control processors located in local rooms (in the reactor building (System 121) or in a bunker in its vicinity). These control processors initiate and supervise safety measures and communicate with the computer systems (System 520) for indication in the main control room, for recording and for alarm annunciation, as well as providing inputs

to the trip and interlock system (System 516). Sampling pipes passing through the containment wall are provided with shut-off valves. Depending on the functions of the measuring channels, these valves are operated manually, either locally or remotely, or automatically, -- via the control processors or via the trip and interlock system (System 516). The system consists of the following measuring channels:

- One channel for monitoring the pressure during normal operation (K101),
- Four channels for monitoring the pressure to indicate pipe fracture inside the containment primary volume (KX102) (X denotes A, B, C and D),
- Two channels for monitoring the pressure after an accident inside the containment (KA103, KB103) (indication in the emergency monitoring station (System 518)),
- One channel for monitoring unidentified leakage by measurement of sump drainage (K321),
- One channel for monitoring unidentified leakage by monitoring of drainage flow from the ventilation coolers (K311) in the containment atmosphere cooling system (System 7415)
- Four channels for monitoring the temperature of the atmosphere, for indicating pipe fracture inside the containment primary volume (KX501),
- Six channels for monitoring the temperature of the atmosphere, for supervising the performance of the cooling equipment and for locating small leaks (KA502-KA504, KB502-KB504),
- One channel for measuring the temperature in the space surrounding the top of the steel vessel extension (K511),
- Two channels for monitoring the total gamma activity in the containment after an accident (KA721, KB721)

In addition, it is possible to take samples of the containment atmosphere for analysis in the laboratory.

During normal operation, the pressure and temperature of the atmosphere in the reactor containment is monitored to check that the cooling arrangements are functioning, and that no large primary system leakage has occurred. The functioning of redundant channels is checked automatically by comparison of signals. In the event of a pipe failure in the containment primary volume, safety measures are initiated by the monitors for pressure (KX102) or temperature (KX501). The shut-off valves in the sampling pipes will be closed automatically, except for the pipes to the pressure measuring channels. In the event of a large rupture (very high pressure and/or temperature increase) the containment spray function is initiated. After an accident, the containment atmosphere is monitored by channels KA103 and KB103 for pressure, channels KA501 and KB501 for temperature, and channels KA721 and KB721 for radioactivity. The signals from these channels are recorded by a special logging program in the computer system (System group 520) and by recorders in the emergency monitoring station (System 518). The unidentified leakage detection system comprises sump and cooler drainage monitoring, which is recorded by the computer system.

A.7.18 Failure Modes Effects and Criticality Analysis

Table A.7-1 presents a failure mode effects and criticality analysis for the equipment covered in this appendix based on the level of information available.

FMECA Notes for Appendix A.7

Note a

The PSID states that the reactor scram trip system trips one of main reactor coolant pumps via its power supply (649). It is not clear that there is a separate scram channel for each pump i.e. all four pumps can be used to scram or if there is a designated scram pump. If the latter, it would seem that the redundancy of scrambling is greatly reduced.

Note b

It is not clear if there is redundancy in the manual scram i.e. a scram button that actuates redundant switches and or redundant scram buttons.

Note c

In addition to tripping a main cooling pump, system 5161:

- 1) interrupts the power to the filtered and distilled water injection pumps in the reactivity control systems (3121 and 3122),
- 2) Closes the shut-off valves in the inlet lines,
- 3) Displays a scram alarm in the alarm display system,
- 4) Trips a scram alarm on the safety parameter display system.

Note d

System 5162 initiates the following actions when tripped:

- 1) Rapid reduction of feedwater flow to the steam generators by throttling the steam supply to the feed pump drive turbine,
- 2) Drops the power to the filtered and distilled water injection pumps in the reactivity control system,
- 3) Initiates injection of borated water into the primary loop by the pumps in the reactivity control systems,
- 4) Trips a reactor hot shutdown alarm in the alarm display system,
- 5) Trips a plant reactor shutdown survey on the safety parameter display system.

Note e

System 5163 isolates the containment. A trip of system 5163 results in:

- 1) closure of isolation valves in reactivity control, reactor pool water replacement, pressurizer, steam lines and other systems,
- 2) turbine trip,
- 3) trips the hot reactor shutdown (5161),
- 4) trips alarm on alarm panel, and
- 5) trips isolation indication on the safety parameter display system

Note f

System 5165 when initiated:

- 1) Trips the power to one main coolant pump
- 2) Trips the power to the filtered and distilled water injection pumps

Note g

The safety grade emergency control room facilitates monitoring conditions inside of the reactor vessel and the reactor containment if the buildings around the reactor are destroyed or their equipment is inoperable.

The emergency control room is located in the bottom of the reactor building. Its close proximity to the reactor raises questions of its survivability or flooding vulnerability.

Note h

Both process computers will operate at the same time each accepting the same data and performing the same data processing, but only one is sending information for display and control. If the active computer fails, the backup is switched over to presenting data and control signals.

Note i

Because the PSID describes system 521 as providing safety-related computations, display and control, it is assumed that the technical specifications (Tech Specs) will require that both computers be operative before startup. It is assumed that limited operation may continue with one computer failed.

Note j

It is assumed that the computers and plant are operating thus the programs necessary to make the computers work have been loaded. A storage medium now fails. If the medium had been failed before startup, it may not have been possible to start unless it is possible to make transfers between the two computers.

Note k

The pump speed (see Figure A.7-4) is controlled by comparing averaged temperature measurements at 3 heights in the lower density lock with the average of the outlet temperatures. This may be viewed as using the outlet temperature as a reference and comparing with an average temperature in the lock. If the hot-cold interface is centered in the lock a certain ΔT will be measured. If the interface moves down, the average temperature in the lower lock rises and the ΔT decreases which is used as a signal for less pump speed and conversely if the interface moves in the other direction.

The gist of this is that the interface location is not directly sensed but inferred from the two measurements.

Note l

System 535 provides the flow rate and withdrawal rate control signals to the injection pumps and to the control valves, respectively, in the reactivity control system.

Note m

The pressurizer level control computer accepts mode and reactor pool replacement signals to control pumps P1 and P2 and valves V33, 34, 37 and 38 in the reactor pool replacement system 325 to introduce and release water from the pressurizer.

Note n

The reactor pressure control computer gets information on the mode and signals from the reactor pool, pressurizer and other systems to control the heaters, pressurizer offgassing valves and pressurizer spray valves to control the reactor pressure.

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Trip and Interlock System 516

Reference: PIUS PSID Preparer: R. Fullwood, BNL

Date: December 30, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
1	Reactor scram trip system 5161/ scram the reactor	failure of manual scram	a) human error to initiate manual scram, b) failure of electronics c) failure of controls	Possible core damage	H	IV	Redundancy, independence diversity, and testing	Subsystem uses 2-out-of-4 logic. See notes a,b and c
2		failure to trip on neutron power flux monitors	a) mis-calibration, b) set-point drift, c) electronics failure	see above	M	IV	see above	Subsystem uses 2-out-of-4 logic. See notes a and c.
3		failure to trip on excessive rate of power increase	see above	see above	M	IV	see above	
4		failure to trip on excessive reactor outlet temperature	see above	see above	M	IV	see above	see above
5		failure to trip on excessive PCRV temperature	see above	see above	M	IV	see above	see above

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Trip and Interlock System 516

Reference: PIUS PSID Preparer: R. Fullwood, BNL

Date: December 30, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
6	Reactor scram trip system 5161/ scram the reactor (cont.)	failure to trip on excessive pressurizer pool level	a) mis-calibration, b) set-point drift, c) electronics failure	Possible core damage	M	IV	see above	Subsystem uses 2-out-of-4 logic. See notes a and b.
7		failure to trip on turbine trip	a) electronics failure, b) welded contacts	see above	M	IV	see above	see above
8		failure to trip on isolation trip	a) electronics failure, b) welded contacts	see above	M	IV	see above	see above

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Trip and Interlock System 516

Reference: PIUS PSID Preparer: R. Fullwood, BNL

Date: December 30, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
9	Hot shutdown trip system 5162/ shut-down the reactor	failure of manual scram	a) human error to initiate manual scram, b) failure of electronics c) failure of controls	Possible core damage	H	IV	Redundancy, independence diversity, and testing	Subsystem uses 2-out-of-4 logic. See note d
10		failure to trip on neutron power flux monitors	a) mis-calibration, b) set-point drift, c) electronics failure	see above	M	IV	see above	see above
11		failure to trip on excessive rate of power increase	see above	see above	M	IV	see above	see above
12		failure to trip on excessive reactor outlet temperature	see above	see above	M	IV	see above	see above
13		failure to trip on excessive PCR V temperature	see above	see above	M	IV	see above	see above

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Trip and Interlock System 516

Reference: PIUS PSID Preparer: R. Fullwood, BNL

Date: December 30, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
14	Hot shutdown trip system 5162/ shut- down the reactor (cont.)	failure to trip on excessive pressurizer pool level	a) mis-calibra- tion, b) set-point drift, c) electronics failure	Possible core damage	M	IV	see above	Subsystem uses 2-out-of-4 logic. See note d
15		failure to trip on turbine trip	a) electronics failure, b) welded con- tacts	see above	M	IV	see above	see above
16		failure to trip on isolation trip	a) electronics failure, b) welded con- tacts	see above	M	IV	see above	see above

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Trip and Interlock System 516

Reference: PIUS PSID Preparer: R. Fullwood, BNL

Date: January 2, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
17	Isolation trip system 5163/ isolate the reactor from the environment	failure of low reactor pool signals and or electronics to trip containment isolation	a) mis-calibration, b) set-point drift, c) electronics failure d) sensor failure	Failure to correct low reactor pool water preceding core uncover and isolate containment	M	V	Test, inspection, redundancy	Safety impact requires the concurrent occurrence of a radiological release inside of the containment. See note e.
18		failure to detect and or respond to high radiation in the half-hour delay off-gas system and isolate the containment	see above	Failure to detect core damage and act to prevent release to the environment	M	IV	see above	Core damage must have occurred before this failure would have safety impact. See note e.
19		failure to detect and or respond to excessive pressure in main steam lines and isolate the containment	see above	Failure to correct excessive steam pressure that might precede rupture of a steam line	M	IV	see above	Excessive steam pressure may result from a reactivity transient or improper operating conditions. See note e.

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Trip and Interlock System 516 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: December 30, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
20	Isolation trip system 5163/ isolate the reactor from the environment	failure to detect and or respond to high radiation in the steam lines	a) mis-calibration, b) set-point drift, c) electronics failure d) sensor failure	Failure to detect steam generator tube failures and act to prevent release to the environment.	M	III	Test, inspection, redundancy	The primary is at higher pressure than the steam loop thus allowing contaminated primary water into the secondary system and release. See note e.
21		Above in conjunction with core damage	see above	Failed steam generator tubes in conjunction with core damage could result in a severe environmental release	M	V	See above	See above
22		failure to detect and or respond to excessive temperature and pressure in the containment	see above	Failure of the containment due to excessive pressure or due to excessive temperature weakening of concrete	M	V	See above	For this to be safety significant, it must occur concurrently with an accident in containment. The design does not include a pressure relief for the containment. See note e.

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis								
Trip and Interlock System 516 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 2, 1992								
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
23	Isolation trip system 5163/ isolate the reactor from the environment	failure to detect and or respond to excessive water level, temperature, and or pressure in the reactor building	a) mis-calibration, b) set-point drift, c) electronics failure d) sensor failure	Flooding, and excessive temperature can result in common mode failures of equipment in reactor buildings. Excessive pressure may rupture pressure boundaries. These are accident initiators and should be responded to immediately	M	III	Test, inspection, redundancy	See note e.
24		failure to detect and or respond to signals from the pipe break monitoring system	see above	Flooding can result in common mode failures of equipment in parts of the plant that are flooded. These are accident initiators and should be responded to immediately	M	IV	see above	See note e.

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis								
Alarm Display System 517 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: December 30, 1991								
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
25	Isolation trip system 5163/ isolate the reactor from the environment	failure to detect and or respond to excessive water level, temperature, and or pressure in the reactor building	a) mis-calibration, b) set-point drift, c) electronics failure d) sensor failure	Flooding, and excessive temperature can result in common mode failures of equipment in reactor buildings. Excessive pressure may rupture pressure boundaries. These are accident initiators and should be responded to immediately	M	III	Test, inspection, redundancy	See note e.
26		failure to detect and or respond to signals from the pipe break monitoring system	see above	Flooding can result in common mode failures of equipment in parts of the plant that are flooded. These are accident initiators and should be responded to immediately	M	IV	see above	See note e.

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Trip and Interlock System 516

Reference: PIUS PSID Preparer: R. Fullwood, BNL

Date: January 2, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
27	Startup interlock system 5165/ prevent criticality during certain stages of plant refueling and service shutdown	failure of manual scram	a) human error to initiate manual b) scram, failure of electronics c) failure of controls	Possible core damage, exposure of personnel to radiation	H	III	Redundancy, independence diversity, and testing	The PSID indicates a power trip to one pump. If this trip is unsuccessful, the scram will fail. Additional redundancy is needed. See note f.
28		failure to detect and or respond to low pressure signals from the reactor vessel	a) mis-calibration, b) set-point drift, c) electronics failure d) sensor failure	Allows criticality while at low reactor vessel pressure	M	IV	the above and training	This failure alone would not result in criticality. It would also be necessary to either block the flow or match the thermal flow with the loop flow and then remove poison from the water passing through the core. See note f.

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Alarm Display System 517								
Reference: PIUS PSID Preparer: R. Fullwood, BNL					Date: January 2, 1992			
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
29	VDU's in the plant control room/ provide operators with plant status information	failure of VDU's, failure of signal conditioning and transmission, failure of contacts	failure from: electronics, improper environment,	inadequate information on plant condition.	H	IV	Redundancy, independence diversity, and testing. Training for operation in absence of VDU's	Single VDU failures are anticipated. Common mode VDU failure would require reliance on display and alarm panels. Human factors shock should be considered after reliance on VDU's.
30	VDU on the shift supervisor's desk/ provide shift supervisor with plant status information	see above	see above	see above	M	IV	see above	shift supervisor would loose all information
31	VDU on the plant operator's desk/ provide plant operators with plant status information	see above	see above	see above	L	IV	see above	plant operator would loose all information
32	VDU on the overall display panel/ provide plant status information	see above	see above	see above	L	IV	see above	loss of information displayed on the overall panel

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis								
Emergency Control Room System 518			Reference: PIUS PSID Preparer: R. Fullwood, BNL			Date: January 2, 1992		
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
33	scram button/ trips one of the main cooling pumps	fails to open	shorted, shorted wiring, or pump trip equipment inoperable	Inability to scram the reactor from the ECR	H	V	Redundancy	See note g. A scenario that destroyed the buildings for normal reactor scram would likely scram the reactor without the necessity for ECR scram. Such massive destruction might short the scram wires. Again there is concern in the lack of redundancy in tripping a main cooling pump.
34	Instrumentation display/ provide remote indication of temperature, and pressure in the reactor vessel	failure of indication	wiring broken or shorted, electronics failure	Inability to know the core status	H	V	Redundancy, testing, design for survival	See note g. The lack of the core status information will affect emergency planning probably necessitating the adoption of worst case scenarios.
35	Instrumentation display/ provide remote indication of the water level in the reactor vessel	failure of indication	wiring broken or shorted, electronics failure	Inability to know the core uncover status	H	V	Redundancy, testing, design for survival	see above
36	Instrumentation display/ provide remote indication of temperature, and pressure in the containment	failure of indication	wiring broken or shorted, electronics failure	Inability to know the containment status	H	V	Redundancy, testing, design for survival	see above

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Main Process Computer 521 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 3, 1992								
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
37	Central processing unit/ accept information, perform operations on the data and provide temporary storage	stops working	operator error electronic component failure, fire, earthquake, flood, hazardous control room	loss of both computers would force a plant shutdown.	L	II	Redundancy, training	See notes h and i. The PSID provides no information on the Tech. Specs. as to whether or not PIUS can be started or operate with only one process computer operative.
38		sends out erroneous safety-related data	see above	initiate an reactivity transient	M	III	Self checking and cross process computer checking algorithm	See notes h and i. The PSID does not provide information on cross checking between computers or what determines a computer error that would require a switch to the backup computer.
39	Storage media/ store data, programs and operating history	stops working	see above	inability to store data in a preferred form	L	II	Redundancy	See note j. It is presumed that the recording is highly redundant and the loss of a fast storage medium is backed by a slower medium.
40	Printing media/ printouts as required by plant operators	stops working	see above	inability to present hardcopy to operators	L	II	Redundancy	Failure of a printer may be ameliorated by magnetic storage for a later printout or for an off-line printout.

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Control Room Man-Machine Interface 523

Reference: PIUS PSID Preparer: R. Fullwood, BNL

Date: January 3, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
41	Display of fault alarms/ indications of faults to the operators	failure of one or more displays	electronics failure, inhabitable control room	reduction in knowledge about plant operability	M	II	Redundancy of data display, automatic indication of failure	Failure of individual display panels is expected
42		failure of many display panels	shorts in wiring such as caused by fire or loss of power	Inability to control plant	H	IV	Cable separation, redundancy of lines such as through fiber optics.	Multiple loss of indicators should signal a manual scram.
43	Display of process information to operators/ inform operators of thermodynamic conditions	failure to display process information	electronics failure, inhabitable control room	reduction or lack of knowledge about plant operating conditions	M	III	Redundant and diverse presentation	Significant loss of process information should signal a manual scram.
44	Operator control to the process computers/ control reactor	failure of control	see above	inability to control reactor	H	III	Redundant controls	Loss of control should signal a manual scram
45	Enables process control and supervision via control processors 524/ purpose same as description	failure of control	see above	inability to control at the processor level	M	III	see above	see above

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Control Processors 524

Reference: PIUS PSID Preparer: R. Fullwood, BNL

Date: January 3, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
46	tandem microcom-puter system/ process control and signal processors	fails to work	electronic failure, fire, flood	loss of control and information that is processed by the failed FP	M	II	Redundancy - the use of tandem computers	Details on distribution and routing of signals is not indicated in the PSID. The processing should be diverse and redundant.
47	serial data links/ transmission of data from sensors, conditioners, to FP's and from FP's to control room	failure of transmission	fire, mechanical damage, corrosion	see above	M	III	Redundancy such as through fiber optics	see above
48	process interface units/ signal and data conditioning for use by FP	failure of operation	electronic failure, fire, flood	see above	M	III	Redundancy	
49	software/ computer instructions to make the FPs work	coding flaw	human error, design error	see above	H	IV	Software QA	A coding error might be a common mode effect affecting all FPs.

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Core Management Computer System 525

Reference: PIUS PSID Preparer: R. Fullwood, BNL

Date: January 3, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
50	Core management computer/ calculates the core composition as it produces power	computer fails	electronic, human error, environmental conditions	Loss of information on core burnup for the time the computer is failed	L	II	Parallel processing	Interruption in computation of core burnup has no safety implications as long as it is of short (~ few days) duration
51	Data storage/ store programs and data	failure of storage	human error, equipment error	Loss of information - presumably the data are backed up	L	II	Redundant backup of data	

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Safety Parameter Display System 527 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 3, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
52	Microcomputer/ process and display data of safety signif- icance	Computer failure	human error, electronic failure	very little	L	II	Training for operation in the absence of the SPDS	The SPDS is considered to be non-safety grade. Perhaps the greatest effect of its loss would be in the psychological effect of its absence.

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Reactor and Reactor Pool Control and Monitoring 530 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 3, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
52	Neutron flux monitoring systems (SRM, IRM, PRM, and TIP/ reactivity (power) monitoring	monitor reads lower than correct value	short, depletion of active material, power supply voltage low, component failures	power increase and or reactor trip, provide misleading information to operator	L	II	Periodic inspection maintenance, operator training	Operator confusion may be safety significant; common cause effects could be power supplies or detector depletion
53		monitor reads higher than correct value	short/ open circuit, power supply voltage high, component failures	power decrease and or reactor trip, provide misleading information to operator	L	II	Periodic inspection maintenance, operator training	Operator confusion may be safety significant; common cause effects could be power supply failures
54	Boron concentration monitoring system/ measure neutron absorption in primary coolant and pool water	monitor reads lower than correct value	electronic drift, or failure excessive voltage on detector	more reactivity than required, improper mixing of distilled and poisoned water.	M	II	Use of a process computer for self-consistency checking, laboratory measurements	Details on signal validation are not provided in the PSID but this is clearly a case requiring validation
55		monitor reads higher than correct value	electronic drift, or failure low voltage on detector	less reactivity than required, improper mixing of distilled and poisoned water.	M	II	see above	see above

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Reactor and Reactor Pool Control and Monitoring 530 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 7, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
56	pH control system 5333/ monitor cool- ant water pH to control corrosion etc.	monitor reads lower than correct value	electronics failure or drift, improper sampling, human error	improper water chemistry possi- bly leading to material degra- dation	L	II	Alternative meth- ods such as labo- ratory measure- ments	Improper water chemistry must be sustained for weeks to months to have an effect.
57		monitor reads lower than correct value	electronics failure or drift, improper sampling, human error	see above	L	II	see above	see above
58	hydrogen concentra- tion control system 5335/ monitor cool- ant hydrogen content control corrosion etc.	monitor reads lower than correct value	electronics failure or drift, improper sampling, human error	see above	L	II	see above	see above
59		monitor reads lower than correct value	electronics failure or drift, improper sampling, human error	see above	L	II	see above	see above

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Reactor and Reactor Pool Control and Monitoring 530 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 7, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
60	Water conductivity monitor 5337/ monitor water conductivity for water quality	monitor reads lower than correct value	electronics failure or drift, improper sampling, human error	improper water chemistry possibly leading to material degradation	L	II	Alternative methods such as laboratory measurements	Improper water chemistry must be sustained for weeks to months to have an effect.
61		monitor reads higher than correct value	see above	see above	L	II	see above	see above

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Main Coolant Pump Control System 534 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 7, 1992								
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
62	FP microcomputer shared with system 524/ provide the algorithmic computations for pump control	computer failure	electronics failure or failure of power or peripherals for the computer	improper main pump operation in the various system operating modes	M	II	tandem computation	Improper pump flow will result in activating the pool loop
63	Set value generator/ control pump speed based on comparing the average measurement of temperature in the lower density lock with the set value	set value drift	electronics failure, power supply failure, calibration error	shift of hot-cold interface to extreme of range such that a transient results in scram	L	II	redundant comparisons of set values and hot-cold interface measurements	The effect of set value drift is to increase the frequency of spurious scrams.
64	Set value generator/ control pump speed based on an algorithm that compares the outlet temperature with the set value	see above	see above	see above	L	II	see above	see note j
65	Input from mode selection input/ set pump control according to operating mode	failure to receive mode signal	electrical-electronic failure	improper pump profile for operating mode would cause a scram	L	III	redundant and validated mode specification	

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Main Coolant Pump Control System 534 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 8, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
66	Interface temperature monitors/ sense lower density lock temperature profile	failure of a sensor	open sensor, electronics failure, wiring failure	Inability to sense interface - probably result in scram	L	II	Multiple temperature profile sensing	It hardly seems that three temperature profile sensors is adequate. The loss of one sensor biases the sensing high or low or some place in the lock.
67		failure of validation circuitry	electronics failure	see above	L	II	Multiple channels	This has potential for causing a scram
68	Mixed outlet temperature sensors/ sense the outlet temperature to determine lower lock interface position	failure of a sensor	open sensor, electronics failure, wiring failure	Loss of one sensor would have little effect is the signal validation is sufficient. Multiple failures would cause scram	L	II	Multiple sensing already exists	
69		failure of validation circuitry	electronics failure	Inability to sense interface - probably result in scram	L	II	Multiple channels	This has scram potential

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Reactor Power Control System 535 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 8, 1992								
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
70	Shared FP microcomputer/ compute the power control algorithm	computer failure	electronics failure, peripherals failure, power failure	failure of automatic reactivity control - probably result in scram	M	II	tandem or redundant computation with uninterruptable power	Depending on the mode of failure, this could cause a reactivity excursion before scram. See note 1
71	Operating mode selector/ specify operating mode	improper mode selection	electronics failure, wiring, human error	see above	M	III	difficult to mitigate	Improper mode specification could lead to a variety of problems
72	Mixed reactor outlet temperature set value/ set desired temperature	sensor/electronics failure	see above	see above	M	III	redundant set values	
73	Current mixed reactor outlet temperature from reactor, pool and PCR channels K501-505/ input parameters for reactivity control	see above	see above	see above	M	III	redundant sensing and signal validation	
74	Current mixed reactor outlet temperature from reactor, pool and PCR channels K511-512/ input parameters for reactivity control	see above	see above	see above	M	III	redundant sensing and signal validation	

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Reactor Power Control System 535 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 8, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
75	Reactor neutron flux data from system 531/ provide neutron flux, power data for reactivity control	Loss of neutron flux signals	sensor failure, electronics failure, transmission failure	failure of automatic reactivity control - probably result in scram	L	II	redundancy and signal validation	
76	Upper density lock temperature data from pool inactivation system 322/ control for inactivation of pool	Loss of temperature information	see above	inactivation of pool flow	M	II	see above	
77	Reactivity control flow rate feedback / feedback control information	Loss of flow feedback information	see above	failure of automatic reactivity control - probably result in scram	L	II	see above	

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Reactor, Reactor Pool, and PCRV Monitoring System 536 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 8, 1992								
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
78	Reactor outlet temperature monitoring 5361/ measure temperature of reactor outlet for scram	Loss of outlet temperature measurement	sensor failure, electronics failure, transmission failure	Loss of information for scram	M	III	redundancy - the system has one spare sensor	
79	Reactor inlet temperature monitoring 5362/ measure temperature of reactor inlet for pump, reactor and trip signals	Loss of inlet temperature measurement	sensor failure, electronics failure, transmission failure	Loss of control information	L	III	redundancy - the system has two sensors	
80	Fuel assembly outlet water temperature monitor 5363/ measure radial power distribution and margin from boiling	Loss of temperature measurement	see above	Loss of outlet temperature measurement would lead to scram	L	IV	redundancy	There are 10 to 20 sensors. The concern is a common mode that would fail all of the sensor channels.
81	Reactor pool temperature monitor 5364/ monitor condition in upper pool, layering effects, failure of pool cooling	Loss of temperature measurement	see above	Loss of pool status information	L	IV	redundancy	There are 10 to 20 sensors. The concern is a common mode that would fail all of the sensor channels.

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis								
Pressurizer Pool Level Control System 537 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 8, 1992								
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
82	Shared FP microcomputer/ process input for the control of reactor pool water replacement system 325	Control failure	electronics, software, peripherals, human error	failure to control pressurizer level - too high will result in a hard system with pipe break potential, too low will introduce steam outside of pressure dome	M	II	tandem, redundant computers	See note m
83	Mode signal/ to indicate to the microcomputer the reactor mode	Mode indicator failure	electronics, transmission failure	see above	M	III	redundant signals and transmission	
84	Signals from the reactor pool replacement system/ process information	No or improper signals	see above	see above	M	III	see above	

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Reactor Pressure Control System 538 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 8, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
85	Shared FP microcomputer/ process input for the control of reactor pressure	Computer failure	electronics, software, peripherals, human error	failure to control reactor pressure - if too high may lead to pressure relief trip and scram, if too low reduce the boiling margin or boiling may result	M	II	tandem, redundant computers	See note n
86	Mode signal/ to indicate to the microcomputer the reactor mode	Mode indicator failure	electronics, transmission failure	see above	M	III	redundant signals and transmission	
87	Signals from the reactor pool water replacement system/ process information	No or improper signals	see above	see above	M	III	see above	
88	Signals from the pressurizer system/ process information	No or improper signals	see above	see above	M	III	see above	

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Pipe Rupture Monitoring System 546 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 8, 1992								
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
89	Shared FP microcomputer/ process input for the alarm of pipe rupture	Computer failure	electronics, software, peripherals, human error	failure to detect pipe break will delay mitigation actions	M	IV	tandem redundant computers, periodic testing, software QA.	The probability of this failure is the probability of failure of the pipe break instrumentation <u>given</u> that a pipe break has occurred. Since the latter is a low probability, the probability of the overall failure is very low.
90	4 pressure Monitors in rooms containing primary pumps, reactivity control system, pool water replacement systems and reactor water cleanup systems/ detect pipe break by pressure increase	No or improper signals	electronics, transmission failure	see above	M	IV	redundant sensors and signal conditioning and validation, periodic inspection and testing	The pressure monitors are in separate rooms hence are not redundant but the temperature monitors (below) act as redundant on the pressure monitors.
91	4 temperature monitors in rooms containing primary pumps, reactivity control system, pool water replacement systems and reactor water cleanup systems/ detect pipe break by temperature increase	No or improper signals	electronics, transmission failure	see above	M	IV	see above	see above

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Pipe Rupture Monitoring System 546 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 8, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
92	2 pressure and 2 temperature monitors in steam-feedwater line compartment outside of containment/ detect pipe break by pressure and or temperature increase	No or improper signals	electronics, transmission failure	failure to detect pipe break will delay mitigation actions	M	IV	redundant sensors and signal conditioning and validation, periodic inspection and testing	The pressure and temperature monitors act in redundancy and diversity
93	2 pressure and 2 temperature monitors high in the turbine building/ detect pipe break by pressure and or temperature increase	No or improper signals	electronics, transmission failure	failure to detect pipe break will delay mitigation actions	M	IV	redundant sensors and signal conditioning and validation, periodic inspection and testing	see above
94	8 flow monitors in the monitored floor drain system/ detect pipe break by water flow in drains	No or improper signals	electronics, transmission failure	failure to detect pipe break will delay mitigation actions	M	IV	redundant sensors and signal conditioning and validation, periodic inspection and testing	These floor drain sensors are too separated to be considered redundant.

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Radiation Monitoring Systems 550 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 8, 1992								
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
95	Extended delay offgas system input monitor 551/ measure radioactivity of gas input to the delay	Sensor failure	Failure of: detector, high voltage, electronics	Failure to detect excessive radiation entering delay	L	IV	Redundancy, periodic testing	Other sensors should indicate an upset condition indicate of releasing radioactive gas.
96	Half hour delay offgas system input radiation monitor 552/ measure radioactivity of turbine condenser gas removal system and input to the delay line from the half hour delay off gas system	Sensor failure	Failure of: detector, high voltage, electronics	see above	L	IV	Redundancy, periodic testing	see above
97	Stack radiation monitors 553/ monitor radiation exhausting the stack	Noble gas and or aerosol monitors 5532, 5533	see above	Failure to detect excessive radiation entering stack	L	IV	see above	see above
98		Emergency monitors 5536	see above	Failure to monitor very high accident dose levels	M	IV	see above	This system has two self-powered detectors to cover very high dose rates

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Steam Generator Tube Rupture Monitoring System 550 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 8, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
99	N-16 gamma detector/ detect steam generator tube failure by primary contamination	failure or low gamma detection efficiency	low high voltage, detector deterioration, electronics	failure to detect steam generator tube leak leading to radioactivity in secondary coolant and offsite release	M	III	redundancy, periodic testing, self calibration	Detector failure must coincide with steam generator tube failure. System should be examine for power supply independence
		high efficiency or noise in detector	high voltage, detector deterioration, electronics	unnecessary shutdown because of believed steam generator tube failure	L	III	signal validation, periodic testing, self calibration	2/3 voter logic used for signal validation
100	Control circuitry/ close isolation valve on detection of excessive N-16	control failure	electronics	failure to isolate steam generator on steam generator tube failure	M	III	periodic testing	

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Containment Monitoring System 583 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 8, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
101	Shared FP microcomputer/ process channels for containment monitoring	Computer failure	electronics, software, peripherals, human error	failure to indicate containment conditions normally and during accident conditions	M	IV	tandem redundant computers, periodic testing, software QA.	The probability of failure under accident conditions is the probability of this failure <u>given</u> that a challenge has been made to the containment
102	pressure monitoring channel/ monitor containment pressure under normal conditions	none or erroneous pressure indication	sensor, electronics, signal transmission	lack of containment status under normal conditions	L	II	redundancy, periodic testing	
103	4 pressure channel/ detect pipe break inside containment	see above	see above	failure to detect pipe break inside of containment	M	V	see above	This failure requires concurrent pipe break and all detector failures or else a common mode failure of pressure sensing
104	2 channels/ post accident containment pressure monitoring	see above	see above	inability to know post-accident containment pressure	M	IV	see above	Since there is no pressure relief on the containment, information regarding the high pressure cannot be acted upon

Table A.7-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Containment Monitoring System 583 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 8, 1992								
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
105	sump depth monitoring channel/ monitor for leaks into sump	none or erroneous level indication	sensor, electronics, signal transmission	lack of information on unidentified leaks	L	II	redundancy, periodic testing	
106	flow monitor/ detect drainage flow in ventilation cooler	none or erroneous flow indication	see above	see above	L	II	see above	
107	4 channels of temperature monitoring/ pipe break detection inside of containment	none or erroneous temperature indication	see above	failure to detect leaks	L	III	see above	
108	6 temperature sensing channels/ air temp., equipment monitoring and small leak detection	see above	see above	lack of information on equipment status inside of containment	L	III	see above	
109	1 temperature sensing channel/ air temperature above the steel vessel	see above	see above	lack of information about pressure dome status	L	III	see above	
110	2 channels gamma monitoring/ post accident monitor	none or erroneous reading	see above	lack of post accident information	L	III	see above	

APPENDIX A.8 - PLANT AND TURBINE CONTROL SYSTEMS

This system consists of the following subsystems:

- 591 Plant power overall control system
- 592 Turbine protection system
- 593 Turbine plant process control system
- 594 Turbine plant supervision systems
- 595 Turbine plant automatic operations systems
- 597 Turbine control system
- 598 Feedwater control system

A.8.1 Plant power overall control system - 591

PIUS plant control systems will be more computer-based than in most current nuclear plants for which there are no analogous systems. System 591 controls the plant power by:

- Selecting the mode of operation for major plant control systems,
- Generating set values for a number of variables in the

plant.

- To control the plant participation in grid frequency control,
- To control the plant during the switching to house load operation after a grid load rejection.

System 591 is incorporated as software in the main computer system (System 521) and is given keyboard instructions through the man-machine communication system (System 523). The system can be divided into the following major parts:

- The plant operation mode selector, discussed later,
- The plant power selector, which gives the input to the flow rate set value generator in the feedwater control system (System 598),
- The reactor outlet temperature set value generator, which controls the reactor power control system (System 535) in most plant operating modes. The present tasks and future development possibilities for this generator are discussed under system 535.
- The steam generator pressure set value generator, which has a normal operation output of a constant 4.0 MPa, but which may allow for pressure decrease during plant power increase, and pressure increase during power decrease. The pressure set value will normally be varied during changes in plant power to maintain a rather constant superheating of the steam from the steam generators, but temporarily wider changes in the set value may occur during step changes in plant power.

Power operating modes

The following operating modes are tentatively defined for the plant operating mode selector:

- A) Power operation mode,
- B) Power decrease mode,
- C) Power increase mode,
- D) Plant startup mode
- E) Hot standby mode
- F) Cold shutdown mode

In its various operating modes, System 591 orders plant control systems and other systems when and how to select their operating modes. For some of the major control systems (the main coolant pump speed control (System 534), the reactor power control (System 535), the turbine governor (System 597) and the feedwater control (System 598) systems), this mode selection is shown in Table A.8-1.

Table A.8-1 Relating Modes, Conditions and Systems

591			534	535	597	598
A	Power operation		A	A	A	A
B	Power decrease		A	B	B	B
C	Power increase		A	C	C	A
D	Plant startup	D	D	D	D	
E	Hot standby		E	E	E	E
F	Cold shutdown		(F)	F	(F)	(F)

A.8.2 Turbine protection system - 592

Modern ABB Atom BWR plants have analogous systems to that of PIUS, but not so computerized. System 592 has the task of protecting the turbine plant against accidents and other major structural damage by tripping the steam admission, the steam by-pass and the reactor trip and interlock systems (Systems 422, 413, 516) and others.

The hardware of system 592 is a shared FP micro-computer using software that is essentially identical to that of the reactor trip and interlock system (System 516).

A.8.3 Turbine plant process control systems - System 593

Is a common system for all turbine plant process systems. For the time being it is assumed to be the turbine manufacturer's standard design.

A.8.4 Turbine plant supervision systems - System 594

System 594 has the task of monitoring turbine vibrations to give alarms tripping the turbine protection system (System 592) if certain vibration amplitudes are exceeded. It of a shared FP micro-computer (System 524) or, if the turbine manufacturer so requires, a separate micro-computer. The system receives information from the turbine overspeed and vibration protection devices systems (Systems 4291 and 4293) and other systems and trips the turbine protection system (System 592), if the turbine is not operating properly.

A.8.5 Turbine plant automatic operations system - 595

System 595 coordinates turbine plant systems when the turbine plant switches from one operating mode to another. Its hardware consists of a shared FP micro-computer (System 524) or, if the turbine manufacturer so requires, a separate micro-computer. The system receives information from the plant power overall control system (System 591) and turbine plant systems and gives out signals to the turbine plant process control system (System 593).

Turbine control system - 597

All turbine manufacturers have their own turbine control systems. thus the turbine control is described in ABB Atom's terms. System 597 has the task of controlling the turbine control valves (422 VII-V14) and the turbine by-pass system (413 VII-V15) control valves to:

- Keep the pressure in the main steam lines system (System 411) constant at 4.0 MPa during normal plant operation at constant plant power, and at slightly higher or lower values during normal plant power decreases or increases,
- Permit temporary lowering, or increase, of the pressure in system 411, when rapid plant power increases, or decreases, are requested, and
- Keep the pressure in system 411 below the trip level of the steam lines relief valves during most plant transient and abnormal conditions.

The 597 hardware will essentially consist of a shared FP micro-computer processor (System 524). System 597 gets the following inputs:

- Operating mode selection orders from the plant power overall control system (System 591),
- Steam generator outlet pressure set value from system 591,
- Current steam pressure from the monitors K131- 132 in the steam lines system (System 411)

System 597 gives the following outputs:

- Orders to the steam admission system (System 422) control valves to open or close, and
- Orders to the steam by-pass system (System 413) control valves to open or close.

Operating modes

System 597 presently has the following operating modes, the selection among which will be made by the plant power overall control system (System 591):

Mode A: Power operation mode - System 597 keeps the steam system pressure at the set value (normally 4.0 MPa) received from the plant power overall control system (System 591) . The current value is received from the steam line system (System 411) pressure monitors K131-132.

Mode B: Power decrease mode - During rapid power decreases system 597 will probably temporarily operate at a pressure set value exceeding 4.0 MPa.

Mode C: Power increase mode - During rapid power increases system 597 will probably relax the set value of 4.0 MPa and permit lower values for the steam line pressure temporarily.

Operating Situations

The operating modes of system 597 may be divided into several modes, if features such as transition to house load operation and participation in grid frequency control are requested. During most plant operating situations, where the steam generators operate in the once-through mode, system 597 operates in mode A. It keeps the steam generator output pressure constant at 4.0 MPa or as adjusted by the steam pressure set value generator of the plant power overall control system (System 591) by controlling the turbine control valves in the steam admission system (System 422).

During plant power decreases, the steam pressure set value will probably be raised somewhat in order to make more rapid power decreases possible. System 597 will then operate in mode B.

During plant power increases, the restrictions against lowering of the steam line pressure may be eased somewhat in order to make rapid increase possible. System 597 will then operate in mode C.

A.8.6 Feedwater control system - 598

System 598 is analogous to some B and W PWR plants, e.g. the Mulheim-Kahrlich plant in the Federal Republic of Germany. System 598 controls the plant feedwater flow rate by means of the feedwater pump turbine (System 4632) steam admission control valve and the steam generator inlet piping (System 4636) control valves and:

- Controls the plant power during plant operation, and
- Controls the reactor outlet temperature during plant power decrease.

During startup and shutdown operations, the system will control the feedwater flow rate by means of the auxiliary feedwater pumps (System 4634) control system. The 598 hardware will essentially be limited to a shared FP micro-computer processor (System 524). System 598 receives the following inputs, all from the plant power overall control system (System 591):

- Operating mode selection orders,
- Reactor power order, and
- Limiting factor information.

System 598 gives the following outputs:

- Orders to the feedwater pump turbine (System 4632) control valves,
- Orders to the auxiliary feedwater pump control system (System 4634),
- Orders to the steam generator inlet piping (System 4636) control valves

System 598 operating modes

System 598 has at present the following operating modes, the selection among which is made by the plant power overall control system (System 591):

Mode A: Power operation mode - System 598 controls the feedwater flow rate through the main feedwater pump (Systems 4631 and 4632) and the steam generator inlet control valves (System 4636). It receives the set value from the plant power overall control system (System 591) and the current value from 463 K301 and K3 61-3 64 (in the steam generator inlet lines). The set value may vary during grid transients and loss of load situations. During plant power increase, system 598 receives a higher set value from the plant power overall control system (System 591) but also preprogrammed instructions of an increased feedwater flow rate, which take the time constant of the reactor primary loop into account.

Correspondingly, in plant power decrease situations a lower set value is received from the plant power overall control system (System 591).

Mode B: Power decrease mode - system 598 keeps the reactor outlet temperature at 290°C. The system receives the set value from the plant power overall control system (System 591) and the current value from the reactor outlet water temperature measurements K501-505 in the reactor, reactor pool and concrete vessel monitoring systems (System 536). The mode is terminated automatically by the plant power overall control system (System 591), when the set plant power has been attained and system 598 reverts to mode A.

A.8.7 Failure Modes Effects and Criticality Analysis

Table A.8-1 presents a failure modes effects and criticality analysis for system 590 according to the information available at the time of preparation.

FMECA Notes for Appendix A.8

Note a

System 598 gives the following outputs: to the feedwater pump turbine control valves (4632), to the auxiliary feedwater pump control system (4634), and to the steam generator inlet piping control valves (4636).

Table A.8-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Plant and Turbine Control Systems 590 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 9, 1992								
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
1	Plant power overall control system 591/ specifies the mode for other systems, generates set values, synchronizes to grid, controls plant during switching to house load after grid rejection	main computer failure	electronics, peripherals, human error, software, electric power	plant control failure	H	II	Redundancy, tandem computing	System 521 of which this is a part, is described as non-safety grade, two redundant sets of minicomputers
2		set value generation non-safe	electronics, human error, software	reactivity excursion followed by scram	H	III	Diverse periodic self-checking of set values	See above, the dual computers could include A-D converters or digital comparitors for diverse confirmation of set values
3	Turbine protection system 592/ protect the turbine from upsets	shared FP microcomputer failure	see above	turbine trip causing a cold water reactivity excursion	M	II	Redundancy, periodic testing, software QA	Implementing additional protection against spurious turbine trip may be counter to assuring trip when needed
4				turbine over-speed resulting in missile generation	M	V	diverse turbine speed limit	The location of the turbines seems to be such that a turbine missile would not fail the primary boundary
5	Turbine plant process control 593/ insufficient information							

Table A.8-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Plant and Turbine Control Systems 590 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 9, 1992								
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
5	Turbine plant supervision systems 594/ monitor turbine vibrations and expansions and trip on exceedance	shared FP microcomputer failure	electronics, peripherals, human error, software, electric power	turbine trip with cold water reactivity excursion	M	III	diverse and redundant monitoring	
6				failure to detect incipient turbine failure before blade throwing	M	III	see above	
7	Turbine plant automatic operations system 595/ coordinates turbine plant systems when mode changing	shared FP microcomputer failure	electronics, peripherals, human error, software, electric power	wrong turbine operation for operating mode	M	III	diverse and redundant monitoring	
8	Turbine control system 596/ control turbine control valves	command low or blocked flow	See above	under power transient, steam overpressure	M	III	self checking control	
9		command high flow	See above	over cooling transient	M	III	See above	
10	Feedwater control system 598/ control feedwater turbine	shared FP microcomputer failure	electronics, peripherals, human error, software, electric power	excessive feed-water flow reactivity increase	L	III	diverse and redundant monitoring	See note a, the excessive feed water flow transient would be fairly slow

Table A.8-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Plant and Turbine Control Systems 590 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 9, 1992								
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
11	Feedwater control system 598/ control feedwater turbine	shared FP microcomputer failure	electronics, peripherals, human error, software, electric power	inadequate feed-water flow reactivity decrease	L	III	diverse and redundant monitoring	See note a, the under feed water flow transient would be fairly slow

APPENDIX A.9 - ELECTRIC POWER

The PIUS electric power system consists of the following subsystems:

- 610 High voltage systems
- 620 Grid connections
 - 621 main grid switchyard
 - 622 Startup grid switchyard
- 630 Generator Bus systems
- 640 Regular auxiliary AC power supply
 - 642 6.0 kV auxiliary AC power
 - 643 660 VAC auxiliary
 - 648 Pressurizer heater power
 - 649 Main coolant pump power
- 650 Diesel plant
- 660 Diesel-backed AC power
- 670 Battery-backed power supply systems
 - 672 Regular 110 VDC auxiliary power
 - 673 Safety-grade 110 VDC power
 - 677 Battery-backed auxiliary AC power
- 680 Electrical power control
 - 682 Synchronizing
 - 683 Feeder switching
 - 684 Motor sequencing
 - 685 Relay protection
 - 686 Power measurements
 - 689 energy logging
- 690 Power cables

Some of these systems are identified in Figure A.9-1. At the top, left, is the tie to the 400 kV system, this connects with the 640 auxiliary power distribution system. The 660 V bus has disconnects at each end powered by diesel generators as well as diesel powering the rest of the bus (4 diesels). On the 110 V level are the battery DC and battery-powered non-interruptible AC power supplies.

A.9.1 High voltage systems - Systems 610

The PIUS electrical power is similar to that in most nuclear power stations with the exception that the transmission line voltage is assumed to be 400 kVAC corresponding to the main voltage in Europe.

611 Main Transformer - 611

Non-safety grade System 611 feeds the plant power to the power grid. During startup, it feeds power to the auxiliary systems, and during loss of load, disconnect the plant from the grid.

During plant startup, the generator circuit breaker (System 631) is open, and the grid feeds power to the 6.0 kV auxiliary power AC system (System 642) via the plant auxiliary transformer (642 T1). A redundant route is via the startup transformer (642 T2) from a startup grid. The following abnormal conditions open the plant main circuit breaker and reverts the plant to house load operation.

- Plant loss of load
- Nearby short circuits on the high voltage grid
- Short circuits farther out in the grid, which lower the grid voltage below certain limits for more than a certain period of time

A.9.2 Grid connections - Systems 620

The design is site specific.

A.9.3 Generator bus systems - Systems 630

ABB Atom provides full generator circuit breaker capacity rather than load switches as US practice for PIUS plants. The reasons are the complexity that accompany the load switch alternative and the freedom the circuit breaker alternative gives for the location of the main circuit breaker connection to the main grid (System 621).

Non-safety grade System 631 does the following:

- During normal plant operation, provides power from the turbine generator (System 431) to the plant auxiliary power systems (Systems 640, 660, 670) and the external grid (System 610).
- During plant startup, separates the turbine generator (System 431) from the main transformer (611 T1) and plant auxiliary transformer (642 T1) to make possible the transfer of grid power to the plant auxiliary power systems (Systems 640, 660, 670).
- During loss of the high voltage grid, feeds power to the plant auxiliary power systems (Systems 640, 660, 670) in house load operation.

In the two latter tasks, the system function is backed up by the plant startup transformer (642 T2).

System 631 consists of:

- Generator busbars of encapsulated isolated phase bus (IPBs) type
- Busbar cooling system
- Generator circuit breaker
- Circuit breaker cooling system
- Circuit breaker compressed air system
- Surge capacitors and grounding equipment

The system voltage is conventional to the site (about 20 kV).

During normal plant operation system 631 transmits power to the grid via the main transformer (System 611) and plant auxiliary power to the 6.0 kV auxiliary power AC system (System 642).

During plant startup, the generator circuit breaker may be open, with the main transformer (System 611) feeding power from the grid to the 6.0 kV auxiliary power AC system (System 642).

If the plant load is lost, the main circuit breaker opens, and the generator bus system (System 631) feeds power to the 6.0 kV auxiliary power AC system (System 642) in house load operation.

Total loss of the generator bus system (System 631) will trip the turbine and scram the reactor.

A.9.4 Regular auxiliary ac power supply systems - Systems 640

System 642 6.0 kVAC Power

Non-safety grade System 642 has the task of feeding power from the generator bus system (System 631) to the following loads:

- The 660 V AC buses in the regular low voltage auxiliary power supply system (System 643)
- Auxiliary plant loads which cannot be accommodated on the 660 V buses in system 643 because the motor size exceeds 350 kW.
- The general power distribution system (System 837) for 240V AC loads and occasional 400V loads.

Figure A.9-1. Electric Power Systems

System 642 shown in Figure A.9-1 consists of the following main parts:

- The plant auxiliary transformer (642 T1) with circuit breakers on the 6.0 kV side
- The plant startup transformer (642 T2) which is connected to another power line in the external grid than is the main transformer (611 T1).
- Two symmetric buses 642 A and 642 B with their output circuit breakers (with one cubicle each). Two cubicles per bus for incoming feeders and one measurements cubicle per bus.

Operating situations

During all plant operating situations where the generator bus system (System 631) gets power either from the main generator (System 431) or from the main transformer (System 611), system 642 transmits power from system 63 to the loads connected to the system.

The following situations leave system 642 without power:

- Loss of the generator bus system (System 631)
- Loss of the main transformer (System 611) and failure of the plant to switch to house load operation
- Loss of System 642 itself, e.g. by a bus short circuit and subsequent feeder breaker failures.

The loss of power will trip the turbine, scram the reactor and start up the diesel sets (System 651). The pumps supplied by system 642 will not be powered, and the diesel-backed AC power supply system (System 661) will be supplied with power from the diesel generators (System 65).

643 660 V Auxiliary power AC system

Non-safety grade System 643 has the task of feeding power to all 660 VAC loads in the plant except those that need diesel-backed or battery-backed power supply and occasional loads and some heaters, which are supplied with power by the general power distribution system (System 837). System 643 as shown in Figure A.9-1, consists of two identical buses, each with the following main components:

- Input transformers from the regular 6.0 kV auxiliary power AC system (System 642)
- Incoming feeder circuit breakers and measuring cubicles (two cubicles/busbar)
- Busbars
- Output air circuit breakers and contactors with fuses. (Feeders to sub-distributions are provided with load switches and fuses).

Each bus in System 643 is fed from the corresponding bus in the regular 6.0 kV power supply system (System 642). The number of cubicles for each bus is determined by the load. Some of the loads belong to the diesel-backed AC power supply system (System 661) and do not have cubicles in system 643.

If power is lost in the regular 6.0 kV auxiliary (System 642), System 643 also loses power, the turbine trips, the reactor scrams and the outgoing feeders to process loads trip on undervoltage. The power supply to the diesel-backed AC power (System 661) is lost, the diesels are started to provide power to the 661 and capital investment loads.

648 Pressurizer heater power supply system

Non-safety grade System 648 supplies the pressurizer heaters (System 3283) with power in accordance with the need of the pressurizer (System 328) for pressurization of the reactor system during startup and for maintaining the reactor pressure during normal operation. System 648 consists of two subsystems:

- Two 6.0 kV/400 V three phase transformers (648 TAI and TBI) of 1.5 MW.
- Low voltage switchgear and distribution equipment for the pressurizer heater (System 3283). The heater elements are connected in eight groups with separate feeders from the distribution equipment. The transformers are powered from the regular medium voltage power supply system (System 642), through a circuit breaker. The low voltage switchgear outgoing feeders are controlled by the reactor pressure control system (System 538). The heating power of each heater group is 350 kW which is sufficient to compensate for all heat losses in the pressurizer

(System 328) during normal steady state plant operation. Power is provided to one heater group of the pressurizer heater (System 3283) with on-off control keeping the pressure at 9.0 MPa in the pressurizer steam volume. When necessary, more heater groups are switched on.

Notable cases are:

- Plant startup, when heating the pressurizer, and pressurizing the reactor system.
- Scram, when the total mass of water in the reactor vessel shrinks, the steam volume in the pressurizer increases, and the pressure in the reactor vessel thus drops.
- At increasing reactor power, which makes the volume of the water in the primary system decrease.
- At blow-down of the pressurizer with the pressurizer blowdown system (System 3285).

During plant startup all the heater groups are connected, drawing a total heating power of 2.8 MW. This is followed by a "pressurization" period - of about 5 hours.

649 Main coolant pump power supply system

System 649 is similar to System 649 of ABB Atom's BWR plants in that they have thyristor converters and control wet motor pumps. The BWR system has reactor power control tasks, while the PIUS power control is indirect by controlling the hot/cold water interface position in the lower density lock (System 3221). The requirements on the PIUS system are concentrated on the prevention of interface sloshing in system 3221 rather than on fast pump rundown.

System 649:

- Supplies the main coolant pump motors in the recirculation system (System 313) with power
- Transfers reactor power to the steam generators
- Controls of the hot-cold interface level in the reactor pool loop lower density lock (System 3221).
- Performs the reactor scram function in association with the trip and interlock (System 516) by tripping one reactor main cooling pump.

The system is generally not a safety-grade system, but one of the four converters is assigned to Safety Class 3 because of the reactor scram function.

System 649 consists of the following main components:

- Transformers of 6.0 kV which are connected to the buses A and B of the regular 6.0 kV auxiliary power supply system (System 642). The transformers are distributed on the two buses to even the loads.
- Thyristor converters, which are fed from the transformers and in turn feed the main coolant pump motors (System 313) with 0-6000 VAC power at a constant frequency/voltage ratio with 4 transformers supplying 4.0 MVA and 4 converters supplying 3.5 MVA.

During normal full power operation, all main coolant pumps operate at almost full speed and receive full power from system 649. The main coolant pump speed control system (System 534) keeps the interface in the lower density lock (System 3221) at a constant level. During plant startup, the pumps are started individually, one at a time and run up to their minimum operation speed. Then the main coolant pumps are operated at progressively higher speed, as the reactor primary loop temperature increases. (This is described more fully for Systems 313 and 322.)

A.9.5 Diesel plant - System 650

Non-safety grade System 651 in PIUS plants should not be compared with the diesel plants (1.2 MW each) of present day LWR plants which have important tasks with respect to the safety of the public. The only role of the PIUS diesels is to protect the plant capital investment in the turbine plant, concrete vessel and other major plant components, when all regular power systems are lost in the plant by supplying power to the buses in the diesel-backed AC power supply system (System 661). They are provided with systems for automatic startup and fuel for ten days of operation at rated power.

Tests may be performed as:

- Test of a diesel startup reliability by allowing it to start against a non-loaded system 661 bus
- Test of a diesel unit power generating capacity by allowing it to take over the load on one of the system 661 buses, in a plant operating situation where this is permissible, or by running it in parallel with the regular power supply to the 660 V auxiliary power AC system (System 643).
- Test of the automatic power take-over function by disconnecting one of the system 661 buses from system 643 during plant shutdown and possibly at hot stand-by.

System 651 is only operated for normal system testing. If auxiliary power is lost on a system 661 bus, the corresponding diesel is started automatically to feed power to system 661. The sequence of load takeover is described for system 661.

Malfunction of the diesel unit during testing may make plant shutdown advisable, although it is only required for plant capital investment protection reasons.

A.9.6 Diesel-backed AC power supply systems - Systems 660

System 661 in PIUS plants have no safety function and should not be compared with the diesel-backed power supply systems of present day nuclear plants.

Non-safety grade System 661 has the following tasks:

- Supply power for plant capital investment protection loads, normally from the regular 660 V auxiliary power AC system (System 643) and from the non-safety-grade diesel plant (System 651) when the regular supply of power from System 643 is interrupted
- Supply battery-backed AC bus (System 677) with power, when its regular supply fails
- Supply the safety-grade 110 V DC AC power supply system (System 673) with power.

System 661 shown in Figure A.9-1 consists of two buses, which can be connected either to the corresponding buses, or to the diesel generating units (System 651). The number of cubicles for each bus is 24.

A.9.7 Battery-backed power supply systems - 670

Safety-grade 110 VDC auxiliary power supply system - Systems 673

System 673 has the task of supplying power to some safety-grade loads needing uninterruptable power. It is not shown in Figure A.9-1 but consists of:

- A rectifier fed from the diesel-backed AC power supply (System 661),
- 673 bus,
- Battery capable of supplying the load for one hour.

Normally the batteries are kept charged. If power is lost, the non-safety grade diesel (System 651) will provide power to the bus. Failures such as rectifier failure will not impact for a few hours because of the charge in the battery. Complete loss of 673 will cause loss of the associated measuring channels, but the plant can be monitored and controlled by the normal instrumentation powered by system 672/673, and by channels powered by the 673 bus until repairs are made. The system is located in separated safety-grade equipment rooms in the bottom of the reactor building.

440 V Battery-backed auxiliary power AC system - System 677

Non-safety grade System 677 feeds power to loads requiring uninterruptable power. It consists of two identical subsystems containing:

- Thyristor rectifier powered by the corresponding 660 VAC diesel-backed supply
- An intermediate 110 VDC bus
- Lead-acid battery capacity for 30 minutes
- Thyristor converter from 110 VDC to 440 VAC, 3-phase
- Thyristor switch
- 440 battery-backed AC system busbar
- Standby connection to the corresponding 660 V diesel-backed system by a transformer.

The thyristor switch provides switching from the inverter to the standby feeder from the diesel-backed system without power interruption.

In the event of loss of power, the 440 V busbars will be battery fed but within a minute, the diesels will come up to power and assume the load. If one diesel fails to start, its loads will be supplied by battery for 30 minutes. Rectifier failure does not directly affect operation because power will be provided by the battery. Loss of battery is mitigated by the thyristor providing power from the 440 V busbar to the diesel-backed feeder. Failure of an inverter causes a similar problem.

A special problem related to static inverters is the fuse size limitation but the thyristor switch provides a backup. This switch allows switching the 440 V bus to the standby feeder at any time, e.g., repair and maintenance. The loss of the 677 bus trips the turbine and scrams the reactor; the reactor can be maintained at cold shutdown by the other 677 bus. Total loss of the 677 bus activates the self-protection features of the reactor for shutdown and heat removal.

A.9.8 Power cable systems - Systems 690

The cable systems comply with normal safety requirements for cables that are part of battery-backed safety-grade systems including separate A and B parts which are in the cable room.

In cable rooms the physical separation is by distance only for:

- Division A and B cables,
- Power system and control system cables.

The 6 kV and 660V power cables are run on separated cable racks, with the larger cross-section cables installed with appropriate spacing to ensure cooling. Small area cables are lumped together without any requirements on spacing in one or several layers.

The control and power supply cables to minor equipment, e.g. the distributions from the battery-backed systems, are run in closed or semiclosed conduits of galvanized magnetic material or aluminum to provide some electro-magnetic shielding and mechanical protection. The cable conductors are generally copper but for larger cross-sections cables with aluminum conductors may be used. The in-plant earthing system consists of copper conductors which are run along the cable racks, between major components. Motors and pumps are normally connected to earth by means of the copper shielding in the power cable. The in-plant earthing system is connected to the external earthing system (System 872) at one location - a fix-point in the vicinity of the main transformer (611 TI). Some major cable rooms in the reactor, turbine and chemical systems buildings (Systems 121, 122 and 124) may be made part of the control building (System 126) from the entrance, ventilation and fire protection points of view.

A.9.9 Failure Modes Effects and Criticality Analysis

Table A.9-1 presents a failure modes effects and criticality analysis for the electric power system based on the available information at the time of preparation.

FMECA Notes for Appendix A.9

Note a

The generator bus, during normal operation, distributes generator power to the plant auxiliary systems and the external grid. During startup, it disconnects the generator from the main transformer and plant auxiliary transformer to transfer grid power to the plant auxiliary power systems. During loss of off-site power, it feeds power from the auxiliary plant power systems to house loads.

Note b

The 6 kVAC power system supplies power to the 660 VAC buses in the low voltage auxiliary power system, to auxiliary plant loads requiring higher voltage than 660 VAC (power > 350 kW), general distribution to 220 and 440 VAC loads.

Note c

The non-safety grade 660 VAC system 643 supplies all 660 loads in the plant except those requiring diesel or battery backing and a few loads and heaters which are supplied by the general power distribution system.

Note d

In power operation sufficient pressurizer heating can be achieved with one of eight heaters. Other heaters are needed at: startup, after scram when the mass of water in the reactor vessel shrinks, thus expanding the steam volume and requiring more steam to keep the pressure constant, with increased reactor power which decreases the volume of water in the primary, and at blowdown of the pressurizer.

Note e

Diesel emergency power preforms no safety function (according to the vendor) at PIUS. It is provided for capital investment protection. It supplies 660 VAC auxiliary power (643), powers the battery-backed AC bus 677 and safety grade 110 VDC system with power.

Table A.9-1 PIUS 600 Failure Modes and Effects and Criticality Analysis								
Electric Power 600: High Voltage System 610 & 620			Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 9, 1992					
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
1	High voltage systems, main transformer 611/ increase generator voltage to transmission levels	failed transformer	over heating, moisture ingress, transient insulation failure, leaked oil, lightning	loss of load, loss of off-site power, cold water reactivity transient	M	III	Multiple transformers, fast switching	
2	Grid connections 620/ connect to electrical grid	Depends on siting insufficient information						

Table A.9-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Electric Power 600: Generator bus system 630 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 9, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
3	Busbars/ distribute power - see note a	Failed insulation	Failure of busbar cooling, moisture ingress, foreign matter, mechanical damage	Multiple common mode failures probably leading to scram	M	IV	Inspection, maintenance, QA, electrical breakdown monitors	Failure of auxiliary power results in many failures however for systems designed to fail safe, the results of the failure lead to scram and passive heat removal
4	Circuit breaker/ connect and disconnect power	Fails to open	Failure of circuit breaker or CB control, contacts welded, circuit breaker cooling, circuit breaker compressed air system to blow out the arc	Failure to disconnect auxiliary bus from main transformer - unable to supply emergency power	H	III	See above. Also use multiple series disconnects	Manual disconnect would allow a delayed restoration of emergency power
5		Fails to close	Failure of circuit breaker or CB control, circuit breaker cooling	Failure to connect auxiliary power to generator limits power availability and would force a shutdown	L	III	Inspection, maintenance, QA,	Manual closure could be used but might inhibit some safety action

Table A.9-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Electric Power 600: Generator bus system 630 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 9, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
6	Surge capacitors/ suppress electrical transients	short	over-heating, transient- caused insula- tion break- down, vibra- tion	Short out the auxiliary power - prevent the use of emergency power	H	III	Fuse the capaci- tors	
7		open	vibration, conductor failure	Failure to sup- press line tran- sients that may affect computing equipment lead- ing to common mode failures	M	III	Parallel capaci- tors, periodic inspection, moni- tor line noise	
8	Grounding equip- ment/ transient suppression	open circuit	corrosion, vibration	Electrical noise affecting com- puting equipment leading to common mode failures	M	II	Monitor line noise	

Table A.9-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Electric Power 600: Regular Auxiliary AC Power Systems 640 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 9, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
9	6 kVAC plant auxiliary transformer (642 T1)/ see note b	Shorted transformer	Heating, moisture ingress, vibration	Loss of power for plant systems on 6 kVAC power - result in turbine trip, scram and diesel start	M	III	Inspection, electrical breakdown monitoring, disconnects to remove failed transformer from auxiliary power	
10		Open transformer	Vibration, conductor failure	see above	L	III	Inspection, electrical breakdown monitoring	
11	6 kVAC plant startup transformer (642 T2)/ see note b	Shorted transformer	Heating, moisture ingress, vibration	Loss of power for plant systems on 6 kVAC power - result in turbine trip, scram and diesel start	L	IV	Inspection, electrical breakdown monitoring, disconnects to remove failed transformer from auxiliary power	In normal operation, this transformer is standby as well as this grid connection
12		Open transformer	Vibration, conductor failure	see above	L	IV	Inspection, electrical breakdown monitoring	see above
13	Busses 642 A and B/ see note b	Open breaker	Control circuitry, vibration, particulate	see above	M	III	See above, manual transfer	

Table A.9-1 PIUS 600 Failure Modes and Effects and Criticality Analysis								
Electric Power 600: 660 VAC Auxiliary AC Power Systems 643 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 9, 1992								
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
14	660 VAC plant auxiliary transformer / see note c	Shorted transformer	Heating, moisture ingress, vibration	Loss of power for plant systems on 660 VAC power - result in turbine trip, scram and outgoing feeders trip on under-voltage	M	III	Inspection, electrical breakdown monitoring, disconnects to remove failed transformer from auxiliary power	See note d
15		Open transformer	Vibration, conductor failure	see above	L	III	Inspection, electrical breakdown monitoring	
16	Busbar and circuit breakers/ distribute power	Open breaker	Control circuitry, vibration, particulate	see above	M	III	See above, manual transfer	

Table A.9-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Electric Power 600: Pressurizer Heater Power Supply System 648 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 9, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
17	6 kVAC/400 VAC 1.5 MW / supply power to the pres- surizer heaters during startup	Shorted trans- former	Heating, moisture ingress, vibra- tion	Loss of power for the pressur- izer heaters while in power operation would result in a loss of operating pressure and necessitate a shutdown. If the loss occurred at startup, the plant would shutdown from repair.	L	III	Inspection, elec- trical breakdown monitoring	
18	Low voltage switch- gear/ control heaters and to run on only one	Open switchgear	Control fail- ure, grit, improper maintenance	see above	L	III	Inspection, elec- trical breakdown monitoring	

Table A.9-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Electric Power 600: Main Coolant Pump Power 649 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 9, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
19	Transformers 6 kV/6 kV/ transfer power to thyristors	Shorted transformer	Heating, moisture ingress, vibration	Loss of a power transformer for the primary pumps would cause an automatic scram. The disconnects should free the auxiliary power from the short to allow emergency power	L	III	Inspection, electrical breakdown monitoring	The transformers are connected to busses A and B of the auxiliary power to distribute the load. The thyristors generate electrical noise detrimental to the computer operation. The transformers will provide some isolation to suppress this noise.
20	Electrical noise suppressors/ prevent electrical noise generated by pump control thyristors from affecting computers	Failure to suppress noise	Open capacitor(s), shorted inductance, open spike clippers	Electrical noise may result in erratic computer control and possibly unsafe operation	H	III	Inspection, electrical noise monitoring	The PSID is mute with regards to problems in thyristor operation and the generation and suppression of electrical noise in the system. Thyristor electrical noise varies with the load - it is greater at low load.
21	Thyristors and support circuitry/ control the power and speed to the primary pumps	One or more open thyristors	Thyristor failure, thyristor control failure	Loose one or more phases to one or more pumps to cause a scram	L	II	Waveform monitoring	

Table A.9-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Electric Power 600: Main Coolant Pump Power 649 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 10, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
22	Thyristors and support circuitry/ control the power and speed to the primary pumps	One or more shorted thyristors	Voltage punch-through, thyristor control failure	Pump overspeed resulting in scram	L	II	Waveform monitoring	
23	Speed control electronics/ convert computer control to thyristor control	Overspeed	Electronics	See above	L	III	Redundant monitoring	
24		Underspeed	Electronics	Pump underspeed resulting in scram	L	III	Redundant monitoring	

Table A.9-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Electric Power 600: Diesel Plant 650 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 10, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
25	Diesel engines/ power emergency generators	Failure to start	Mechanical, starting equip- ment, fuel, maintenance, electrical control	No emergency power (except battery power) under a loss of off site power	L	II	Inspection, test- ing, redundancy	See note d. Diesel power, although non-safety grade supplies power to the safety- grade battery-powered AC supply system. This apparent contradiction may be explained by the diesel power extending the time the batteries can supply the power. Presumably the batteries will supply the load during critical times.
26		Failure to run	See above	See above	L	II	See above	See above

Table A.9-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Battery-Backed Power 670: Safety-Grade 110 VDC System 673 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 10, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
27	Rectifier/ charge batteries from AC mains	Failure to rectify AC power	Diode(s) short, transient, excessive heat, failure of cooling	Loss of charging power for batteries - eventually loose vital equipment supplied by DC power	M	III	Store spare parts that are readily replaced	One hour may elapse before discharge of the batteries becomes critical
28	Battery/ store charge	Battery failure	Open circuit, fire, hydrogen explosion from ventilation failure, sulfation, poor maintenance	Little effect because rectifier can supply the loads	L	III	Inspection, maintenance, battery condition monitoring	See above. Batteries are usually connected in series and parallel. If failed units are fused, the configuration is highly redundant
29		Battery failure coincident with loss of charging power to battery (e.g. rectifier failure, AC power to rectifier)	Open circuit, hydrogen explosion from ventilation failure, sulfation, poor maintenance	Loss of vital DC power. Plant will scram and core will be passively cooled but unable to monitor plant status and perform controls	H	IV	Training. Prompt emergency actions to repair and improvise	
30	bus/ distribute DC power	Open or shorted bus	Corrosion, thermal cycling, moisture ingress, overheating	See above	H	V	See above	

Table A.9-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Battery-Backed Power 670: Uninterruptible AC Power 677 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 10, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
31	Thyristor rectifiers powered by diesels/ charge batteries	Rectifier failure	Semiconductor failure, electronic control, excessive heat, excessive moisture	Loss of charging for batteries	M	III	Alarm in control room on loss of charging current	Batteries have a 30 minute duration without charging
32	Batteries/ store electrical charge	Battery failure	Maintenance, heat, fire, explosion	No effect if charging current is still available	L	IV	Alarm in control room on battery failure (obvious from ripple on DC to the inverter)	Batteries are usually connected in series and parallel. If failed units are fused, the configuration is highly redundant
33	Inverter/ transform DC int 440 VAC, 3 phase	Inverter failure	Semiconductor failure, electronics control, heat, moisture	Little effect because the uninterruptable power can be provided from the normal AC power.	L	III	Alarm on loss of inverter	
34	Inverter/ transform DC int 440 VAC. 3 phase	Inverter failure coincident with failure of off-site power and diesels	Semiconductor failure, electronics control, heat, moisture	Loss of uninterruptible AC power - automatic reactor scram and passive cooling but loss of control	M	V	Training and emergency action to restore uninterruptible AC	

Table A.9-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Battery-Backed Power 670: Uninterruptible AC Power 677 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 10, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
35	Thyristor switch/ provide switching from the inverter to the standby feeder from the diesel generator	Failure to transfer	Semiconductor failure, electronics control, heat, moisture	Loss of uninterruptible AC power - automatic reactor scram and passive cooling but loss of control	M	III	Training and emergency action to restore uninterruptible AC	In this case the requirement to transfer is initiated by the inverter failure hence correlated. If the transfer switch fails, uninterruptible AC fails.
36	Busbar/distribute uninterruptible AC	Open or shorted bus	Corrosion, thermal cycling, moisture ingress, overheating	See above	H	V	See above	
37	Transformer connection to 660 VAC diesel power/ backup AC	Failed transformer	Heat, moisture ingress, vibration	Loss of this transformer is similar to item 35 in preventing the AC backup to the uninterruptible power	M	III	Inspection, electrical breakdown monitoring, replacement parts	
38	Fuses/ limit inverter current especially at inverter starting	Blown fuse	Heat, moisture ingress, vibration	Loss of one fuse loses one inverter but it is backed by item 35 - the reactor scrams.	M	III	Detection of inverter status and fast repair	

Table A.9-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

Electric Power 600: Power Cable System 690 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: January 10, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
38	Division A cables/ supply power to division A loads	Open or shorted cables	Corrosion, thermal cy- cling, moisture ingress, overheating, fire	Loss of all divi- sion A power resulting in scram and pas- sive heat removal	M	V	See above	
39	Division B cables/ supply power to division B loads	See above	See above	Loss of all divi- sion B power resulting in scram and pas- sive heat removal	M	V	See above	
40	Power system cables/ supply power to miscellaneous nonsafety loads	See above	See above	Loss of all non- safety loads would affect the operators and result in scram and passive cooling	M	V	See above	
41	Control ca- bles/control opera- tions	See above	See above	Loss of control would result in scram, passive cooling but no control	H	V	See above	
42	Items 38 through 40 in various combinations	See above	Fire is the common mode that could do this	Scram, passive cooling but no control or information	H	V	see above	

APPENDIX A.10 - COOLING

A.10.1 Diesel-backed service water system

Non-safety grade System 712 has the task of cooling the following systems in the plant:

- Systems, which can use sea or river water for cooling
- Systems, which need diesel-backed cooling for protection of plant capital investment
- Systems, from which no leakage of radioactive matter needs to be assumed.

System 712 is shown in Figure A.10-1. The protection against uncontrolled release of radioactive matter to the plant discharge by leakage to system 712 is prevented by either of the following arrangements:

- System 712 is separated from radioactive or potentially radioactive systems by intermediate cooling systems, i.e. the diesel-backed closed cooling system for the reactor pool (System 721), the diesel-backed closed cooling system for components and pools (System 722) and the diesel plant (System 651). These intermediate systems are sampled regularly.
- System 712 has a higher pressure in the heat exchanger than the potentially radioactive intermediate cooling system.

Figure A.10-1 Diesel-Backed Service Water System

The Service Water system consists of four pumps (712 P1 to P4) and pipes from the pumps to the heat exchangers in the diesel-backed closed cooling systems for the reactor pool, for components and pools, and for the diesel plant (Systems 721, 722 and 651). The pump motors P1 and P2 are powered from the A bus of the diesel-backed AC power supply system (System 661) and P3 and P4 from the B bus. One pump on each bus is part of the automatic start-up sequence after loss of external power. The pumps are located in the water intake (System 1121) after the cooling water intake rinsing system (System 711). They supply sea or river water to the heat exchangers. The main pipes are fabricated from carbon steel, lined with rubber, or from centrifugally cast concrete. Distribution pipes to other loads are made from stainless steel. The outlet water from system 712 goes to the warm water channel in the condenser cooling system (System 451). System 712 assists the circulation water systems makeup water system (System 456) in its water makeup function, when the plant is provided with a cooling tower. The system consists of 4 pumps at 240 kg/s (3800 gpm), consuming 180 kW from supply system 661.

System 712 is equipped with the following instrumentation:

- Local flow meters (712 K301-304) for the pump flow rates
- Flow meters (712 K311-312) with central indication and low flow rate alarm
- Local thermometer (712 K501) for the system inlet temperature
- Local thermometer (712 K507) for the system outlet temperature
- Pump motor temperature monitors (712 K511-514)

During normal plant operation, one of the pumps P1 and P2 and one of P3 and P4 operate continuously, irrespective of the water temperature. About 12 MW of heat is transported from the diesel-backed cooling system for the reactor pool (System 721) and about 3 MW from the diesel-backed closed cooling system for component cooling (System 722). At an assumed maximum river water temperature of 30°C, the outlet temperature will be about 40°C. Activation of the reactor pool loop will increase the outlet temperature to 60 to 70°C.

Pump failure causes the standby unit on the bus to start automatically. Loss of offsite power fails the pumps, activates the pool loop, and initiates diesel startup followed by two pumps startup, according to diesel-backed AC power supply (System 661) startup sequence. Loss of all of system 712 in a station black-out situation may cause damage to the turbine. The reactor core integrity does not depend on the function of system 712.

A.10.2 Non-diesel-backed service water system - 713

In PIUS plants, non-safety grade system 713 serves both the reactor plant and the turbine plant. It has the task of cooling systems in the plant:

- Which use river water for cooling
- Which do not need diesel-backed cooling
- From there can be no leakage of radioactive matter

System 713 is shown in Figure A.10-2. It consists of two pumps (713 P1 and P2) and the pipes from these pumps to the heat exchangers, served by the system. The pumps are 100% units and the motors are powered from the A and B buses of the regular low voltage auxiliary power supply system (System 643). The pumps are located in the water intake (System 1121) after the cooling water intake rinsing system (System 711). The main 713 pipes are fabricated from carbon steel, lined with rubber, or from centrifugally cast concrete. Distribution pipes to other loads are made from stainless steel. The 713 outlet water goes to the warm water channel in the condenser cooling system (System 451) and system 713 thus assists the circulation water systems makeup water system (System 456) in its makeup water function. The two pumps provide a flow rate of 300 kg/s (4750 gpm) consuming 240 kW from system 643.

System 713 instrumentation consists of:

- Local flow meters (713 K301-320) for the pump flow rates
- Flow meters (713 K311-312) with central indication and low flow rate alarm

Figure A.10-2 Non-Diesel Backed Service Water

- Local thermometer (713 K501) for the system
- Local thermometer (713 K507) for the system outlet temperature
- Pump motor temperature monitors (713 K511-512)

A.10.3 Cooling tower for cooling the reactor pool - 714

Safety-grade System 714 has no analogy in any other LWR plants. It has the task of cooling the water in the passive closed cooling system for the reactor pool (System 724) in order to insure the reactor pool cooling function during plant transient conditions. The system consists of four loops which are located in separate compartments and physically separated from each other.

Each loop consists of:

- A cooling tower (EA1, EA2, EB1 and EB2),
- Two water coolers (EA11/EA12, EA21/EA22, EB11/EB12 and EB21/EB22).
- Water from the passive closed cooling system for the reactor pool (System 724) flowing through the pipes of the dry cooling tower and
- Air flowing by natural circulation on the outside of the pipes cools the water. The water cooler pipes are equipped with fins to increase the transfer area. The four cooling towers are located on top and at each corner of the containment building. Each cooling tower is capable of removing 4.3 MW at an inlet temperature of 32°C (90°F - dry). The carbon steel water cooler pipes (aluminum fins) have a design pressure of 10.5 MPa (1522 psi).

The cooling capacity depends on the temperature difference between the pool and inlet air. Normally, the pool temperature is 50°C (122°F) and the capacity is low. Activation of the loop activates systems 714 and 724 to limit the pool temperature to 95°C (203°F) even if one loop in system 724 has failed or one cooler in system 714 has failed and the reactor pool cooling system (System 323) has failed.

A pipe rupture or leak in system 714 will be detected by loss of water in the passive closed cooling system for the reactor pool (System 724). Rupture or leak will not affect the other water coolers because of their separation.

A.10.4 Closed Cooling Systems - Systems 720

Systems 720 consist of the following subsystems:

- 721 Diesel-backed closed cooling system for the reactor pool
- 722 Diesel-backed closed cooling system for components and pools
- 723 Non-Diesel-backed closed cooling system
- 724 Passive closed cooling system for the reactor pool
- 726 Chilled water system
- 727 Local air coolers

Diesel-backed closed cooling system for the reactor pool - 721

Non-safety grade System 721:

- Cools the reactor pool (System 321) via the reactor pool cooling system (System 323) and
- During plant operation keeps the reactor pool temperature (System 321) between 40°C to 60°C (104 - 140°F)

- During plant transient conditions, when the reactor is shut down and the reactor pool loop is activated, limits the water temperatures in the top of the reactor pool (System 321).

System 721 consists of the following subsystems:

- 7211 Reactor pool cooling system
- 7215 721 stainless steel AISI 304 pump system

Figure A.10-3 Diesel-Backed Closed Cooling System for the Reactor Pool

Reactor pool cooling system (7211)

Provides the heat exchangers in the reactor pool cooling system (System 323) with cooling water. System 7211, shown in Figure A.10-3, consists of the following main parts:

- Pipes connecting the pump system (7215) with the heat exchangers (323 EAl and EBl) in the reactor pool cooling system (System 323).
- Containment isolation valves 721 VAl through VA4, 721 VBl through VB4.

System 7215, has the task of supplying system 7211 with water, when regular or diesel-backed power is available in the plant. It consists of:

- Two heat exchanger (721 EAl, EBl)
- Two pumps (721 Pl to P2)
- A buffer tank (721 TAl)
- A shunt system for temperature control.

The heat exchangers, the pumps and the shunt system are located in the auxiliary cooling systems building (System 123). There are 2 pumps of 250 kg/s (4000 gpm) consuming 230 kW powered from the A and B buses of the diesel-backed AC power supply system (System 661), respectively. The pumps participate in the diesel startup sequences.

The temperature control system shunts water past the heat exchanger, and keeps the outlet temperature at 25°C, if the inlet water temperature to system 721 drops below 18°C.

During normal plant operation, system 721 transmits 5-10 MW from the reactor pool cooling system (System 323) to the diesel-backed closed cooling system for the reactor pool (System 712). The system outlet temperature varies between 25°C and 37°C and the return between 35°C and 50°C when the inlet temperature varies between 0°C and 30°C.

Activation of the reactor pool loop will increase the temperatures in system 721. The system forward temperature will rise to about 60 to 90°C, and the return temperature to above 140°C, depending on the inlet water temperature (0 to 30°C). Loss of plant external power stops all pumps in the system. Loss of an operating pump will cause automatic start-up of a standby pump.

Black-out in the plant auxiliary power systems causes loss of the system cooling function. The cooling of the reactor pool is then performed by natural circulation in the passive closed cooling system for the reactor pool (System 724).

Leakages or ruptures in system 721 outside the reactor containment will cause loss of the main reactor pool cooling function and necessitate shutting down of the plant for repair.

Leakages or rupture inside the reactor containment will lead to loss of one (out of two) of the heat exchangers, when that part is isolated. The plant operation may continue without significant restrictions after isolation of the leakage or rupture until a suitable time for plant shutdown for repair.

Leaks and ruptures in the tubes of the heat exchangers 323 EAl, 323 EBl in the reactor pool cooling system (System 323) cause loss of reactor pool water to system 721 in which case alarms are initiated and the containment isolation valves closed. If the leak or rupture has occurred in one cooler only, that cooler can be isolated and the plant can continue operation without restrictions. If both coolers are leaking, the reactor plant is shut down in a regular way.

Diesel-backed closed cooling system for components and pools - 722

The system pools are the unconventional aspect of system 722. The pools give extra inherent capital protection features beside those provided by the pump system itself. During normal operation, the system:

- Cools the upper part of the wall of the concrete vessel cavity liner (System 157),

- Anchor flange of the pressure vessel upper parts (System 2115),
- Main coolant pump motors in the recirculation system (System 313) and the
- Pump motors in the reactor pool cooling system (System 323).
- Cools the spent fuel pool via the fuel pool cooling and cleanup system (System 324) and the condensation pool (System 316) via the containment pool cooling system (System 355).

During plant transients, including extended loss of auxiliary power, reactor shut down and reactor pool loop activated, it limits the temperatures in the upper part of the concrete vessel wall and in the anchor flange of the pressure vessel upper parts. System 722 has no safety-grade function.

System 722 consists of:

- Wall cooler system
- Pump cooler system
- Condensation pool cooling system
- Spent fuel pool cooling system
- Pump system

System 722 is completely fabricated of stainless steel AISI 304.

Wall cooler system (7221)

System 7221, shown in Figure A.10-4 supplies the anchor flange cooler (System 2115) and the concrete vessel liner cooler (System 157) with cooling water, consists of the following main parts:

- Two steel vessel pools (722 TA1, 722 TB1), each with a total water volume of 150 m³ (4000 gal.). They are located in dedicated closed compartments on the top of the containment structure and filled with highly borated water (2000 ppm) at 30-40°C during normal plant operation.
- A jet pump in each pool fed by the incoming water from the pump system (System 7225).
- Pipes with isolation valves (722 VA3 through 722 VB4) connecting the pools with the anchor flange coolers and the concrete vessel liner coolers (System 157).
- Pipes with containment isolation valves (722 VA1 through 722 VB2) connecting the pools with the pump system (System 7225)
- A vent pipe system with containment isolation valves (722 VA5 through 722 VB6) from the pool to the atmosphere and to the condensation system (System 316).

The jet pumps receive water from the pump system (System 7225) and inject it into the inlet pipes to the anchor flange and the concrete vessel liner cooling system (System 157). The water flow rate to the coolers will be about twice the flow rate from the pump system because of the jet pump effect.

Pump cooler system (7222)

System 7222, shown in Figure A.10-4, supplies the motor coolers of the pumps in the recirculation system (System 313) and the reactor pool cooling system (System 323) with cooling water. It consists of:

- A pipe system connecting the water supply with the motor coolers of the main coolant pumps (313 EA31 through 313 EB32) and the coolers of the reactor pool cooling pumps (323 EA2 and
- Four containment isolation valves (722 VA7 through 722 VB8)

Condensation pool cooling system (7223)

Shown in Figure A.10-4, has the task of supplying cooling water to the containment pool cooling system. It consists of pipes with valves connecting the pump system (System 7225) with the heat exchangers 355 EA11 and EB11.

Figure A.10-4 Diesel-Backed Closed Cooling System for Components

Spent fuel pool cooling system (7224)

System 7224, shown in Figure A.10-4 supplies cooling water to the spent fuel pool cooling and cleanup system (System 324). It consists of pipes with valves connecting the pump (System 7225) with the heat exchangers 324 EAI and EBI.

Pump system (7225)

System 7225 (Figure A.10-1) supplies the wall cooler, pump cooler and pool cooling systems (Systems 7221 through 7224) with water, when regular or diesel-backed power is available. The system consists of:

- Two heat exchangers (722 EAI and EBI)
- Two pumps (722 PAI and PBI)
- A pipe system
- A shunt system for temperature control

The heat exchangers, pumps, and shunt system are located in the auxiliary cooling systems building. There are 2 100% capacity pumps delivering 200 kg/s (3200 gpm) and drawing 125 kW. The pump motors are powered from the diesel-backed AC power supply system (System 661). The pumps participate in the diesel startup sequences. The temperature control system shunts water past the heat exchanger, and keeps the 722 outlet temperature at 25°C, if the inlet water temperature to the diesel-backed service water system (System 712) drops below 18°C. It keeps the level constant in the pools 722 TAI and 722 TBI.

Plant operating situations

During normal plant operation, system 722 transmits about 3 MW from the pump motor coolers in the recirculation system (System 313) and the reactor pool cooling system (System 323) to the diesel-backed service water system (System 712) in which case pools 722 TAI and TBI are hardly involved. Residual heat from spent fuel is transmitted from the spent fuel pool cooling and cleanup system (System 324) and whenever needed heat is transmitted from the condensation pool (System 316) via the containment pool cooling.

Activation of the reactor pool loop will increase the temperatures in system 722. Loss of plant external power supply causes all pumps in system 722 to stop. When the diesels have started, both pumps are started according to their start up sequence. Loss of an operating pump will cause automatic start-up of the standby unit.

A black-out in the plant auxiliary power systems (including loss of diesels) results in loss of the forced cooling functions, i.e. loss of the reactor pool cooling system (System 323), the diesel-backed closed cooling system for the reactor pool (System 721), the diesel-backed closed cooling system for components and pools (System 722) and the diesel-backed service water system (System 712). The upper part of the wall of the concrete vessel cavity liner and the anchor flange of the vessel are then cooled via the pools (722 TAI, TBI) in natural circulation. The pool water volumes are large enough to provide cooling for a total station black-out for 48 hours with boiling if the blackout is protracted.

Envelope ruptures in system 722 necessitate shut down of the plant for repair. The plant will be brought down to cold conditions in the normal way using the steam generators and secondary systems or by means of the natural circulation in the passive closed reactor pool cooling system (System 724).

Non-diesel-backed closed cooling system - System 723

Non-safety grade System 723 (Figure A.10-5) cools systems in the plant which meet both of the following characteristics:

- Need of an extra protection barrier against leakage of radioactive matter out of the plant with cooling water
- No need of diesel-backed power for the pump motors.

Figure A.10-5 Non-Diesel Backed Closed Cooling System

It consists of the following main components:

- One heat exchanger (723 E1) to the non-diesel-backed service water system (System 713),
- Two pumps which are 100% for normal loads and 50 to 75% for maximum load
- A pipe system
- A shunt system for control of the system temperature with a control valve (723 V1)
- An expansion vessel (723 T1) with auxiliary equipment
- Pneumatically operated valves in pipes to systems which are pressurized by radioactive water.

System 723 is fabricated from carbon steel except for the heat exchanger. The heat exchanger, pumps and the shunt system are located in the auxiliary cooling systems building (System 123).

The dominating load for system 723 is the cooling of the primary system water withdrawn by the reactivity control system (System 312) and cooling in the reactor water cleanup system (System 331). Another major load is the cooling demands from the containment atmosphere cooling system (System 741), the local air coolers (System 727) and the air inlet coolers of the ventilation systems (Systems 740). The 312 E3 heat exchanger handles up to 10 MW during normal plant operation. The maximum loads occur during power decreases - going up to about 20 MW. The system provides a flow rate of 240 kg/s (3800 gpm) consuming 270 kW from system 643.

It has instrumentation as follows:

- Local thermometers (723 K501-502) for the system inlet and outlet temperatures.
- Temperature measuring channel (723 K503) for the control of the system outlet temperature.
- Flow monitor (723 K301) for indication, alarm and standby pump startup automatic function.
- Level monitors (723 K401 and K402) in the expansion vessel.

During normal plant operation, system 723 transmits a power of some 10 to 15 MW to the non-diesel-backed service water system (System 713). At the maximum inlet water temperature of 30°C to system 713 and maximum load on system 723, system 723 will have outlet and inlet temperatures of about 35°C and 50°C, respectively, at normal system loads. If the 713 inlet temperature drops below 20°C, the shunt valve of system 723 will keep the 723 outlet temperature at 25°C. Failure of a 723 pump will automatically result in startup of the standby unit during plant operation. Loss of external power will stop the function of system 723

Passive closed cooling system for the reactor pool - 724

Safety-grade System 724 (Figure A.10-6), having no analog in other LWR plants, cools the reactor pool (System 321) during plant transient conditions, including a total loss of feedwater and an extended loss of diesel-backed auxiliary power, when the reactor pool loop is activated. The system which limits the temperature in the upper part of the reactor pool to 95°C consists of eight parallel loops. Each loop consists of the following main components:

- One heat exchanger (EA11, EA12, EA21, EA22, EB11, EB12, EB21 and EB22).
- Two inner containment isolation valves.
- Two outer containment isolation valves.
- One pressurizing vessel (TA11, TA12, TA21, TA22, TB11, TB12, TB21 and TB22).

System 724 passively transfers, by natural circulation, the residual heat of the reactor core from the reactor pool (System 321) to the cooling towers. The design is for an essentially atmospheric pressure system that limits the water temperature in the upper part of the reactor pool to < 95°C when the inlet air (dry) temperature to system 714 is 32°C with each loop providing a cooling capacity of about 2.1 MW. If one loop is assumed to be out of operation, the remaining seven loops have a total heat transfer capacity of 15 MW. The top of the heat exchangers (reactor pool water inlet) is located approximately 20 meters below the water coolers in the cooling towers for cooling the reactor pool (System 714). The design is for 8, 500 tube heat exchangers with 2.14 MW heat transfer capacity receiving 95°C water at 0.15 MPa (design maximum is

Figure A.10-6 Passive closed Cooling System for the Reactor Pool

< 10.5MPa). Each tube is 16/19 i.d./o.d. (mm). The system parts located outside the concrete pressure vessel are fabricated from carbon steel - other parts are fabricated from stainless steel.

Pipe ruptures or leaks in system 724 are detected by alarms for high/low pressure and or high low level in the pressurizing vessel. A rupture or leak in one loop will not affect the operation of the remaining loops in system 724 since they are separated from each other. Pipe ruptures or leaks in the water coolers in the cooling towers for cooling the reactor pool (System 714) are detected by alarms for low pressure and low level in the corresponding pressurizing vessel. The containment isolation valves in the actual loop will then be closed. The rupture or leak will not affect the operation of the remaining loops in system 724 since they are separated from each other.

Chilled water system 726

Non-safety grade System 726 has the task of supplying coolers in the air cooler system (System 727) and several ventilation systems (System 740) with chilled water. It consists of the following subsystems:

- 7261 The refrigerator system
- 7263 The closed cooling water system

The large components of the system are located in the auxiliary cooling systems building (System 123). System 7261 consists of two 50% refrigerating units having a cooling capacity of 200 kW each at a temperature drop of 5°C in the coolers cooled by the non-diesel-backed closed cooling system (System 713). Each refrigerator uses 100 kW powered from the A and B buses of the regular low voltage auxiliary power supply system 643.

The closed cooling water system consists of:

- 100% redundant pumps (726 P31 and p32) providing a flow of 25 kg/s (400 gpm)
- 30 kW pump motors power from the regular low voltage auxiliary power supply system 643
- Carbon steel pipe thermally insulated.

System 726 is used in normal operation in rooms occupied by personnel. The control building has a small air conditioning system powered by diesel-backed power.

A.10.5 Water Treatment and Distribution Systems - 730

Systems 730 consists of the following subsystems:

- 731 Raw water treatment system
- 732 Water demineralizing system
- 733 Demineralized water distribution system
- 734 High pressure distilled water system
 - 7341 Pump system
 - 7343 High pressure purge water system

Raw water treatment system - 731

System 731 cleans available raw water, so that a water quality suitable for water demineralizing system and the tap water system (Systems 732 and 761) is achieved. The system comprises a chemical flocculation and two parallel continuous sand filters, designed to remove flocks and other suspended solids. In order to avoid problems of incrustation and bacterial attack, the water is pre-chlorinated.

The system is supplied with water by raw water pumps which deliver the water flow to be cleaned through the sand filters. Flocculent solution is added to the water at the filter inlet. The filter sand is continuously cleaned during normal operation, and a small proportion of the raw water flow is discharged as waste water. The filtered water is then, after pre-chlorination, delivered to collecting pool T1 of 300 m³. The system is sized to supply fresh water to meet the total plant demands during normal operation as well as during overhaul work. The system can deliver filtered water at a rate of 2 x 20 m³/h. The raw water is assumed to have such a quality that flocculation and sand filtering are sufficient for cleaning.

Water demineralizing system

Non-safety grade System 732 (Figure A.10-7) has the task of producing demineralized water for the following uses:

- Makeup water for the reactor water cleanup system, the condenser makeup water system (Systems 331, 446) and other systems
- Makeup water for the condensation pool (System 316) and the elevated water pools (System 722)
- System cleaning and flushing
- Controlled access area room cleaning

The system consists of two parallel-water treatment lines, each with a production capacity of 5 m³/h of demineralized water. The water pre-treatment comprises prefiltering through a fine filter and filtering through a reverse osmosis (RO) membrane. For the demineralization the system consists of two mixed-bed ion exchangers. Common equipment for both lines is an ion exchange feeding tank, a chemical storage tank, a dosing pump for pH adjustment of the water to the RO lines and regeneration chemical tanks for the ion exchangers. The filtered raw water from the raw water treatment system (System 731) or municipal water from the tap water system (System 761) is pumped to the RO lines. Sulfuric acid is added to the water flow by means of a dosing pump to get a suitable pH in order to attain a long lifetime of the RO membranes. The flow is divided into two pre-treatment lines. In each line the water is led through a fine filter (C11/C21) where undissolved particles are removed. A high pressure pump (P11/P21) gives the necessary pressure for the reverse osmosis membrane (C12/C22), after which the water from both lines is collected in a common tank (T2) of 5 m³. This tank is a feeder tank for the two parallel ion exchanger lines of the final demineralization stage. Each line contains a mixed-bed ion exchanger (C13/C23) with a feeder pump (P12/P22). The regeneration system with tanks T3 and T4 is common for both ion exchangers. From the exchangers a common pipe takes the water to the storage tank in the demineralized water distribution system (System 733) via the resin trap C4.

The system operation is automatic and governed by water demand. There are level monitors in the storage tank of the demineralized water distribution system (System 733). During normal system operation, one line is on duty producing the demineralized water needed; the other line is on standby, as backup and spare capacity for special situations. When regeneration of an ion exchanger is started, the parallel line is put into operation. Regeneration is initiated manually by starting an automatic sequence. The mixing of regeneration chemicals is manual. Washing of the reverse osmosis membrane is automatic at predetermined time intervals. Each line can produce at least 600 m³ of demineralized water between regeneration of the mixed-bed ion exchangers when the incoming water meets the water quality requirements. The water quality is monitored after each treatment step, and if limits are exceeded disconnections and alarms are initiated.

Demineralized water distribution system 733

Non-safety grade System 733 (Figure A.10-8):

- Stores and distributes demineralized water from the water demineralizing system (System 732) to the points where it is used
- Stores demineralized water for service campaigns so as to diminish the load on system.

The system consists of the following main components:

- A storage vessel with a volume of 400 m³
- Two pumps of 4 kg/s each (733 P1 and P2)
- A gas cushion pressurizer (733 T2)
- A feeder pipe system

Figure A. 10-7 Demineralized Water System

Figure A.10-8 Demineralized Water Distribution System

The system, fabricated from AISI 304 stainless steel, supplies water to the following systems:

In the reactor building (System 121):

- Passive closed cooling system for the reactor pool (System 724)
- Faucets, including the sampling system (System 336) (System 733)

The concrete vessel cavity is filled and replenished via the reactor pool water replacement system (System 325). The spent fuel pool (System 243) and the fuel service pool (System 242) are filled via the spent fuel pool cooling and cleanup system (System 324). The condensation pool (System 315) is filled from the reactor water cleanup system (System 331). During refueling shutdown the reactor service pool (System 244) is filled with water via the spent fuel pool cooling and cleanup system (System 324), with water from the condensation pool and from the storage tanks in the reactor water cleanup system.

In the turbine building (122) the systems are:

- Condensate makeup water system (System 446)
- Condensate cleanup system (System 467)

In the auxiliary cooling systems building (System 123) the system is the non-diesel-backed closed cooling system (System 723). The diesel-backed closed cooling systems (Systems 721 and 722) are supplied with highly borated water from the chemical additives system (System 3331).

In the chemical systems building (System 124) the systems are:

- Faucets, including the sampling system (System 336) (System 733)
- Reactor water cleanup system (System 331)
- Chemical additives system (System 333)
- Liquid waste system (System 342)
- Waste solidification system (System 343)
- Decontamination system (System 344)

In the active workshops building (System 129) the systems are:

- Faucets, including the sampling system (System 335) (System 733)
- Decontamination system (System 344)

In the water treatment building (System 135) the system is the water demineralizing system (System 732)

High pressure distilled water system 734

Non-safety grade System 734 (Figure A.10-9) has the task of supplying wet motor high pressure pumps (313 PAI, PA2, PBI, PB2 and 323 PAI, PBI) with purge water. System 734 consists of the following subsystems

7341 Pump system

7343 High pressure purge water system

The system consists of the following main components:

- Two redundant reciprocating pumps (P11 and P12) for 1.7 kg/s (27 gpm) at 9.4 MPa powered from the regular low voltage auxiliary power supply system (System 643).
- Two pressure pulse compensators (T11, T12)
- Two filters (C11, C12)
- One gas cushion pressurizer (T 13) of 0.2 m3
- Level and pressure instrumentation and central alarms in the pressurizer T13 and a control systems by which the pressurizer level controls the pumps.

P11 and P12 take water from the distilled water tank T22 in the reactor water cleanup system (System 331) or from the demineralized water distribution system (System 733) and deliver it to System 7343.

Figure A.10-9 High Pressure Distilled Water System

High pressure purge water system (7343)

System 7343 consists of the following main components

- A chemical degassifier (C31) for 100 g/s
- A distribution pipe system to the pumps in the recirculation system and the reactor pool cooling systems

(System 313 and 323)

- A central flow alarm and a local flow monitor (K331) in the main feeder pipe
- One local flow monitor in each individual pump feeder
- Isolation valves and containment vessel penetrations for each individual pump feeder.

System 7343 delivers about 20 g/s of degassed water to each wet motor, high pressure pump in the plant. This water keeps oxygen-rich and high boron content water from entering the pump motors. The water to the pumps of the recirculation system (System 313) flows on into the primary loop and the water to the pumps of the reactor pool cooling system (System 323) into the reactor pool. 10 ppm boron is acceptable to the 313 and 323 pump motors. At this concentration boric acid acts rather as a corrosion inhibitor, and the water in the pump motors has a very much higher conductivity than the purge water. 7341 is operated at a constant capacity during normal plant operation. The pump system is operated at about 50% capacity.

A.10.6 Containment Ventilation System - 740

System 740 consists of:

- 741 Containment ventilation system
- 742 Reactor ventilation system
- 743 Chemical systems building ventilation system
- 744 Turbine building ventilation system
- 745 Other controlled area ventilation systems
- 746 Control building ventilation systems
- 747 Electrical and diesel buildings ventilation systems
- 748 Ventilation systems for other non-controlled areas

Containment ventilation system -741

Non-safety grade System 741 (Figure A.10-10):

- Keeps the temperature of the atmosphere in the containment drywell at about 50°C during normal operation and hot shutdown
- Keeps a low humidity in the containment drywell atmosphere
- Cools some components directly in the containment during normal operation.

System 741 comprises the following subsystems:

- 7411 Air supply system
- 7413 Air exhaust system
- 7415 Air cooling system

Air supply system (7411) consists of:

- Air supply inlet dampers (741 VA12, VB12)
- Two parallel fans (QA11, QB11) providing a pressure of 1.5 kPa, 35,000 m³/h each using 20 kW motors powered from the A and B buses of the regular low voltage auxiliary power supply system (System 643).
- Check dampers for each fan (741 VA11, VB12)
- One air filter (741 C11)
- Cooling coils fed by system 723 (741 E11)
- Differential pressure meters across the filter (741 K211) and across the fans (741 K212)
- Two containment isolation valves (741 VA13, VB13)

The supply system is connected to the air cooling system (System 7415) inside the containment.

Air exhaust system (7413) consists of:

- Two parallel fans (QA31, QB31) for 2 kPa, 35,000 m³/h each, and with 30 kW motors, powered from the 643 A and B buses
- Check dampers for each fan
- One differential pressure monitor across the fan set for alarm purposes (741 K231)
- Air exhaust outlet dampers
- Two containment isolation valves (741 VA33, VB33)

Air cooling system (7415) consists of:

- Two air filters (741 FA51/FB51).
- Two sets of cooling units (EA51, EB51), each comprising a number of identical coolers.
- Two sets of fans (QA51/QA52, QB51/QB52), each comprising two fans with 50% capacity each.
- Temperature monitors (KA551/KB551, KA552/KB552) before and after the coolers.
- Differential pressure monitors across the fans (741 K251 and K252).
- Condensation flow rate monitors (System 583) from the coolers to the condensation pool (System 316).
- Drain pipes from the cooler sets to the condensation pool (System 316).
- One combined iodine (charcoal) and HEPA filter unit with fans (QA53/QB53).
- Differential pressure monitors across the filter (741 K253) and across the fans (741 K254)

The coolers and fans are located in the containment drywell area. The cooling units are equipped with drain pipes, discharging to the containment monitoring systems (System 583) for monitoring of leakage from systems carrying hot, pressurized primary system water, condensing in the coils. The cooling coils are cooled by the non-diesel-backed closed cooling system (System 723). The material used in the system is painted carbon steel and galvanized steel. Aluminum may be used for certain components. Drain parts are fabricated from stainless steel. The fans QA51 through QB52 are axial flow fans providing 1.5 kPa, 100,000 m³/h each, by 60 kW motors, powered from the 643 A and B buses. The filter unit fans (QA53/QB53) provide 20,000 m³/h each, with 10 kW motors.

During normal plant operation, the air supply system (System 7411) and the air exhaust system (System 7413) are shut down and the inlet and outlet dampers are closed. The air cooling system (System 7415) operates, with two circulation fans, one per set. The motors of the fans that are not running are heated by 0.1 kW electric heaters in order to prevent condensation. Some air is drawn to the cooling units from the top of the containment and some is drawn directly from the space surrounding the steel vessel dome. The cooled air is distributed through annular ducts to the spaces where major heat dissipating components and equipment are located, to the space surrounding the steel vessel dome and the floor above the concrete vessel, and along the walls of the shafts.

The filter circuit (with the fans QA53/QB53) is used to clean the atmosphere from radioactive material, when personnel is to enter the containment for inspection and minor service. When major personnel activity inside the containment is planned, the air supply and air exhaust systems (Systems 7411 and 7413) are put into operation.

The normal system operation mode of system 7411 and 7413 is that one fan in each subsystem is operating, yielding an air exchange rate of 2-2.5 times per hour. Failure of one fan results in automatic startup of the redundant fan.

In the event of a failure in system 7415 or the non-diesel-backed closed cooling system (System 723) that causes the air temperature in the containment to rise, an alarm is initiated by the temperature monitors.

Reactor building ventilation system 742

Non-safety grade System 742:

- Supplies the reactor service room with clean air
- Heats the reactor building during cold season refueling and during reactor service outages
- Cools the reactor service room during hot season, especially during refueling and reactor service outages.
- Facilitates monitoring of the radioactivity of the area exhaust air
- Facilitates filtered emergency ventilation of the reactor building after postulated refueling accidents.

System 742 serves the plant controlled reactor service room in the upper part of the reactor building and other compartments in the reactor building. It consists:

7421 Air supply system

7423 Air exhaust system Air supply system (7421)

System 7421 consists of:

- Air supply inlet dampers (742 VA12 and VB12)
- Two parallel fans (742 QA11 and QB11) each providing 1 kPa at 30.000 m³/h using 10 kW motors powered from the A and B buses of the regular low voltage auxiliary power supply system (System 643) - Check dampers for each fan (742 VA11 and VB11)
- Two motor operated dampers in the supply duct
- One air filter (742 C11)
- Heating coils fed by the plant heating system (System 763)
- Cooling coils fed by the non-diesel-backed service water system or the chilled water system (System 713 and 726)
- Differential pressure monitors across the filter for supervision (742 K211) and across both fans for alarm (742 K212)
- Temperature monitors for local indication (742 K511), alarm (742 K512) and building temperature control (742 K513)

Air exhaust system (7423) consists of:

- Two parallel fans (742 QA31 and QB31) giving 1 kPa, 30,000 m³/h each, using 10 kW motors powered from the 643 A and B
- Check dampers for each fan (742 VA31 and VB31)
- One differential pressure monitor across the fan set for alarm purposes (742 K231)
- Air exhaust outlet damper (742 VA32 and VB32)

The exhaust air is discharged to the plant stack. In addition, there is an outlet damper (742 VA41) for connection to the emergency ventilation system (System 749).

During normal plant operation, the reactor service room is not occupied. There are no moving parts and no heat sources located there. System 742 replaces the room air about once per-hour. The temperature ranges between 5°C and 40°C. When work is to be performed on tools and components in the reactor service pool, the temperature on the floor is 15°C to 30°C and is done during refueling and service outages. If required, the chilled water system and the local air cooler system (System 726 and 727) remove the heat and moisture from the open reactor pool. The temperature in the reactor pool will not exceed 50°C when the reactor vessel cover has been removed. Water evaporating from the pool surface removes the maximum cooling power stated in system 742 to be 200 kW. In a refueling accident with radioactive release to the reactor service room, radiation monitors (System 555) will stop the fans, close the inlet and outlet air dampers, and start up system 749 (the emergency ventilation system).

Chemical systems building ventilation system 743

Non-safety grade System 743 (Figure A.10-11) does the following with respect to the area defined below:

- supply it with clean air
- heat it during the winter
- cool it during the summer
- facilitate filtration of the exhaust air from rooms with potential for contamination
- facilitate monitoring of the radioactivity of the exhaust air.

The 5000 m³ plant controlled areas served by system 743 are the chemical systems building (System 124) and the active workshop building (System 129) built together as one building unit, in addition, System 743 serves the chemical system building area which consists of:

- The chemical systems parts
- The waste handling systems parts

System 743 consists of the following subsystems:

- 7431 Air supply system
- 7433 Air exhaust system
- 7435 Point exhaust system

Air supply system (7431)

This system, located in the chemical system building, consists of the following main components:

- Air supply inlet damper (743 V12)
- Two parallel, 100% fans (743 QA11 and QB11) giving 1.5 kPa, 50,000 m³/h each from 20 kW motors, powered from the A and B buses of the regular low voltage auxiliary power supply system (System 643)
- Check dampers for each fan (743 VA11 and VB11)
- Heating coils fed by the plant heating system (System 763)
- Cooling coils fed by the non-diesel-backed service water system and the chilled water system (System 713 and 726)
- Differential pressure monitors across the filter for supervision (743 K211) and across both fans for alarm (743 K212)
- Temperature monitors for local indication (743 K511) alarm (743 K512) and building temperature control (743 K513).

Air exhaust system (7433)

The system consists of the following main parts:

- Two parallel 100% fans (743 QA31 and QB31) providing 1.5 kPa, 50,000 m³/h each from 20 kW motors powered from the 643 A and B buses
- Check dampers for each fan (743 VA31 and VB31)
- A differential pressure monitor across the fan set, for alarm purposes (743 K231)
- An exhaust air radiation monitor (743 K731) for alarm purposes, and switchover to the offgas filter system (System 749)
- Air exhaust outlet damper (743 V32)

The exhaust air is passed to a stack. If excessive radiation is detected, damper (743 V33) routes the air through the offgas filter system (System 749).

Point exhaust system (7435)

Consists of the following:

- Point exhausts in the rooms containing the offgas systems (Systems 341 and 349), filters and evaporators in the water cleanup system (System 331), the storage tanks for filter aids and resins from cleanup systems (System 342) and the solidification equipment in the solid waste system (System 343).

Figure A.10-11 Chemical Systems Building Ventilation System

- An exhaust fan (743 Q51) for about 1.5 kPa, 3,000 m³/h with 4 kW motor, powered from the 643 A bus.
- A filter set, comprising an air filter (743 C51) and a HEPA filter (743 C52)
- A differential pressure monitor across the filter set for supervision (743 K251)

System 743 has one operating mode. For the most part, the permissible temperature range is 5°C to 40°C in the building. In rooms where people work, the range will be 15°C to 30°C provided by local heating and cooling systems. At certain locations, with potential for contamination, directed ventilation may be operated via the point exhaust system with filtering of the exhaust air. If excessive radioactivity is detected in the exhaust air, by the radiation monitor, the air exhaust outlet damper is closed, and the exhaust air will then be routed to the offgas filter system (System 349).

A.10.7 Failure Modes Effects and Criticality Analysis

Table A.10-1 presents a failure modes effects and criticality analysis for the cooling systems based on the available information at the time of preparation.

FMECA Notes for Appendix A.10

Note a

The PSID indicates that a backup pump may be available following the loss of an operating pump. However, backup pumps are not shown on Figure A.10-3.

Note b

The PSID does not identify the components located in containment. The operation of these components could be affected by excessive containment temperature and humidity.

Note c

Figure A.10-10 does not indicate that filter differential pressure measurements are displayed remotely.

Table A.10-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Diesel-backed Reference: PIUS Preliminary Safety Information Document Preparer: B. Shier, BNL Date: January 10, 1992
Service Water System: System 712

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
1	Pumps (P1,P2,P3,P4)	Electrical failure	Motor short,open circuit	Loss of flow	M	IV	Pump flow rates are monitored with remote indication and a low flow alarm	Two pumps are needed for normal operation. Two pumps are backup
2	Piping system	Rupture/leak	Missile, vibration,weld failure	Loss of all flow,depending on location	M	V	See above	Loss of all service water flow may cause turbine damage but not core damage.
3	Flow instrumentation	Loss of flow indication	Electrical short, vibration, human error	No flow indication	L	IV		

Table A.10-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System:Non-diesel-backed **Reference:** PIUS Preliminary Safety Information Document **Preparer:** B. Shier, BNL **Date:** January 10, 1992
Service Water System: System 713

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
1	Pumps (P1,P2)	Electrical failure	Motor short,open circuit	Loss of flow	L	IV	Pump flow rates are monitored with only locally	
2	Piping system	Rupture/leak	Missile, vibration,weld failure	Loss of all flow,depending on location	L	V	See above

5 of 6

Table A.10-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Cooling tower for cooling reactor pool System 714 **Reference:** PIUS Preliminary Safety Information Document **Preparer:** B. Shier, BNL **Date:** January 10, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
6	Piping for cooling water	Rupture or leak	External missile; seismic	Loss of 1/4 pool cooling capacity	M	V	Leakage will be detected by loss of water in System 724	
7	Piping system	Plugging due to foreign material buildup	Corrosion products	Reduced pool cooling capacity	L	IV	Reduced cooling by System 721 will cause a slow pool heatup	

Table A.10-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Diesel backed closed Reference: PIUS Preliminary Safety Information Document Preparer: B. Shier, BNL Date: January 10, 1992
cooling for reactor pool: System 721

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
8	Heat exchangers (EA1,EA2)/ heat exchange with System 712	Leakage	Corrosion / cyclic fatigue/ unrelieved strains	Leakage from System 712	L	III	Reduced flow in System 712 is alarmed	
9	Pipe rupture/leak	Loss of cooling function	Missiles / vibration/ corrosion/ weld failure	Loss of cooling function	L	V	Buffer tank appears to have pressure and level instrumentation	Leakages inside containment will result in loss of one (out of two) heat exchangers and can be isolated. Leakage outside containment will require plant shutdown for repair.
10	Isolation valves/isolate one cooling system	Failed closed or inadvertent closure	Valve failure or human error	Loss of one cooling system	L	IV	System outlet temperature will increase	Passive closed cooling system (System 724) provides additional means of cooling the reactor pool

Table A.10-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Diesel backed closed
cooling for reactor pool: System 721

Reference: PIUS Preliminary Safety Information Document
Preparer: B. Shier, BNL

Date: January 10, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
11	Pumps/circulation of cooling water	mechanical failure, i.e., failed impeller, failed shaft, cavitation	Overheating, fatigue failure, aging	See above	L	IV	See above	See note a
12	Temperature instrumentation/ control of heat exchanger outlet temperature	Loss of temperature control	Electrical failure	Erratic system fluid temperature	L	IV		
13	Temperature sensors	Loss of integrity	Corrosion / cyclic fatigue/ vibrations	Loss of temperature control	L	V	Fluid temperature at heat exchanger outlet will be erratic	

Table A.10-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Diesel-backed closed **Reference:** PIUS Preliminary Safety Information Document **Preparer:** B. Shier, BNL **Date:** January 10, 1992
cooling for components and pools: System 722

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
14	Heat exchangers (EA1,EB2)	Leakage	Corrosion / cyclic fatigue/ unrelieved strains / tube tubesheet weld cracking	Loss of cooling capability	L	III	Reduced cooling will cause system fluid temperature to rise	
15	Piping system	Rupture/leak	Missiles / vibration/ corrosion/ weld failure	Loss of cooling function	L	V	The PSID does not describe any remote operator actions	
16	Isolation valves/isolate one cooling system	Failed closed or inadvertent closure	Valve failure or human error	Loss of one cooling system	L	IV	See above	

Table A.10-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Diesel-backed closed cooling for components and pools: System 722 Reference: PIUS Preliminary Safety Information Document Preparer: B. Shier, BNL Date: January 10, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
17	Pumps (PA1,PB2)	Loss of flow	Overheating, electrical failure, shaft failure, cavitation	Loss of cooling capability	L	IV		Each pump can supply 100% of the required flow
18	Temperature control system	Loss of function	Electrical failure, sensor failure, aging effects	Erratic system temperature response could effect system cooling capability	L	IV		

Table A.10-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Passive Closed Cooling

Reference: PIUS Preliminary Safety Information Document

System for Reactor Pool: System 724

Preparer: B. Shier, BNL

Date: January 10, 1992

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
19	Heat Exchangers (EA11,EA12,EA21,EA22,EB11,EB12,EB21,EB22)	Reduced heat removal	Flow obstruction	Reduced heat removal	L	V		PSID does not indicate instrumentation for temperature or flow monitoring
20	Heat Exchangers	Low flow	Foreign material accumulation, corrosion	Reduced heat removal	L	V		System consists of eight parallel loops
21	Piping system	Rupture/leak	Weld failure, missiles, vibration, seismic	Loss of one loop cooling	M	V		
22	Isolation valves	Failed closed	Electrical failure, obstruction	Loss of one loop cooling	M	V		

Table A.10-1 PIUS 600 Failure Modes and Effects and Criticality Analysis								
System: Containment Reference: PIUS Preliminary Safety Information Document Preparer: B. Shier, BNL Date: January 10, 1992 Ventilation System: System 741								
No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
23	Air Filter (CA51,CB51)	Clogged	Foreign material accumulation, moisture	Reduced containment cooling	L	IV	Fig. A10-10 indicates a differential pressure measurement across the filters. Containment temperature is monitored and excessive temperature change activates an alarm	See Notes b. and c.
24	Cooling units (EA51,EB51)	Coolant leakage	Vibration, corrosion,aging effects	Loss of unit cooling capacity	L	IV	See above for containment temperature monitoring	Exact number of cooling units is not specified
25	Cooling units (EA51,EB51)	Air flow blockage	Foreign material accumulation	Reduced cooling capacity	L	V	See above	

Table A.10-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Containment Reference: PIUS Preliminary Safety Information Document Preparer: B. Shier, BNL Date: January 10, 1992
 Ventilation System: System 741

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
26	Fans (QA51/QA52) (QB51/QB52)	Loss of electrical power	Short circuit, open circuit	Loss of unit cooling	L	IV	See above	
27	Fans (QA51/QA52) (QB51/QB52)	Loss of flow	Fan blade failure	Reduced unit cooling capacity	L	V	See above	

Table A.10-1 PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Reactor Building Reference: PIUS Preliminary Safety Information Document Preparer: B. Shier, BNL Date: January 10, 1992
Ventilation System: System 742

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
28	Air supply/exhaust dampers	Blocked or failed closed	Foreign material accumulation/mechanical failure	Reduced heating and cooling capacity	L	IV	Building temperature is monitored and alarmed	
29	Air supply and exhaust fans	Loss of power	Electrical open circuit or short	Loss of room heating and cooling	L	IV	See above	
30	Building temperature control system	Loss of temperature control	Electrical failure, sensor failure	Erratic building temperature	L	V	See above	

APPENDIX B

HAZOP SESSION LOGSHEETS

Following are logsheets for HAZOP sessions of the following portions of the system:

the pressurizer, including the dome, riser funnel, siphon breaker standpipe, and pool, down to below the level at which steam is injected;

the pressurizer heating system (boiler and associated piping, pumps, etc.);

the upper and lower downcomers;

the core channels (two were defined in the thermal hydraulic nodalization).

Some time was spent investigating the core entrance region and the lower density lock. These sheets are not provided, because the discussions were based on an inaccurate picture of those portions of the system.

Each portion of the record is introduced by a sheet giving a description of the segment being analyzed and a characterization of the design intent. The participants in that particular session are also listed. The record of each session gives the guide words, the causes associated with each, and the consequences associated with each cause. Special assumptions which must be borne in mind by the reader are also noted. In many instances, questions came up whose resolution could not be accomplished within the time frame of the discussion; these were noted, and individuals were assigned to determine the answers. In most cases, the information available did not lend itself to resolution of the issue.

In a CPI HAZOP, the "Recommendations" column would contain such guidance as "add a valve." Such recommendations are beyond our scope; we have recommended analyses, not hardware modifications.

These sheets were generated on-line by the HAZOP recorder, as the discussion proceeded. The software (obtained from Technica) provided a screen image of the logsheet; this was projected live by a viewgraph projector, so that the team could read the record of their deliberations as it was being generated.

Many column entries are "PC" for Previously Considered, or "NA" for Not Applicable.

In addition to the HAZOP reviews of particular subsystems, a HAZOP of the startup procedure (as described in the PSID) was conducted. Each step is characterized by its design intent, and deviations from this intent are generated and discussed. A typical deviation might be "more action", meaning that the operator went too far in carrying out some mandated action. On these sheets, the "cause" column for that deviation lists instances or examples of the indicated deviation (turn the knob too far), rather than "cause," which, for errors, might be "argument with boss." The consequence column has its usual meaning.

Project: PIUS SYSTEM STUDY	Revision: 3	Section: 001	Page: 1						
	Date: 11/27/91	Time: 11:20:07							
	Session:	Team: (see below)							
Section Description: PRESSURIZER (UPPER HEAD, UPPER PLENUM, TOP PORTION OF UPPER RISER) .. NOT INCLUDED ARE STEAM INJECTION SYSTEM OR POOL COOLING	Leader: RY	Recorder: CC							
	Organization: BNL								
	Location: BLDG 130								
Design Intention: GENERATES AND MAINTAINS NORMAL OPERATING SYSTEM PRESSURE BY MAINTAINING TEMPERATURE IN TOP FEW FEET OF WATER, PRESSURE RELIEF, CONTROL OF NON-CONDENSIBLES, LIMITS PRESSURE VARIATIONS DURING TRANSIENTS, MAINTAINS PZR POOL LEVEL, MAINTAINS RISER FUNNEL LEVEL, MAINTAINS SIPHON BREAKER LEVEL	P&ID's: Revision: PIUS 600 DWG, FIG 211.A, 89-12-15								
Team Members: <table border="0"> <tr> <td>R. YOUNGBLOOD</td> <td>R. MORANTE</td> <td>B. CHAN</td> </tr> <tr> <td>S. CHENG</td> <td>B. LUCKAS</td> <td>P. SOO</td> </tr> </table>				R. YOUNGBLOOD	R. MORANTE	B. CHAN	S. CHENG	B. LUCKAS	P. SOO
R. YOUNGBLOOD	R. MORANTE	B. CHAN							
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Section Desc: PRESSURIZER (UPPER HEAD, UPPER PLENUM, TOP PORTION OF UPPER RISER) NOT INCLUDED ARE STEAM INJECTION SYSTEM OR POOL COOLING

GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
LOWER REACTIV.	N/A	..				
HIGHER REACTIV.	N/A					
MORE BORON	N/A					
LESS BORON IN RISER FUNNEL	NORMAL STEAM CONDENSATION	UNBORATED SLUG IN THE RISER FUNNEL	SOME HYDRAULIC PERSON SHOULD LOOK AT CONSEQUENCES OF GETTING THIS SLUG INTO THE PRIMARY LOOP			
LESS BORON IN SIPHON BREAKER	SEE "LESS BORON IN RISER FUNNEL"		REF. SYSTEM 328 DISCUSSES BORON CONCENTRATION OF BOILER FEEDWATER AND THE MEANING OF THIS HAS TO BE INVESTIGATED			
NO STEAM FLOW	CONTROL SYSTEM TOLD IT NOT TO COME ON, MALFUNCTION IN NON-HAZOPED PART	SLOW PRESSURE DROP LEADING TO A TRIP DOWNWARD TEMPERATURE DRIFT				
	BREAK IN STEAM INJECTION SYSTEM	SEE CONSEQUENCE FOR "MORE RISER FUNNEL LEVEL"				
REVERSE FLOW	PIPE BREAK	LOSE PRESSURE VERY FAST DEPRESSURIZATION INVENTORY LOSS				
MORE STEAM FLOW	CONTROL SYTEM MALFUNCTION	INCREASE PRESSURE OSCILLATING PRESSURE STRUCTURAL EXCITATIONS OPEN RELIEF VALVE, BLOWING DOWN TO CONTAINMENT INCREASE IN POWER GENERATION?				ASSUMING THE RELIEF CAPACITY EXCEEDS STEAM INJECTION CAPACITY

Section Desc: PRESSURIZER (UPPER HEAD, UPPER PLENUM, TOP PORTION OF UPPER RISER) NOT INCLUDED ARE STEAM INJECTION SYSTEM OR POOL COOLING

GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
MORE STEAM FLOW (con't)	INADVERTENT OPENING OF SAFETY RELIEF VALVE	BLOWING DOWN TO CONTAINMENT REDUCE PRESSURE POTENTIAL SCRAM				
	ANY LEAK IN PRIMARY PRESSURE BOUNDARY	PC				
LESS STEAM FLOW	MALFUNCTION IN STEAM INJECTION SYSTEM	GRADUAL PRESSURE DROP PC (SEE NO STEAM FLOW)				
OSCILLAT. STEAM FLOW	OSCILLATING PRESSURE	PC				
	CHUGGING RELIEF VALVE	STRUCTURAL EXCITATIONS				
	PRESSURE CONTROL SYSTEM PROBLEM					

Project: PIUS SYSTEM STUDY	Revision: 2	Section: 002	Page: 1
	Date: 12/23/91	Time: 09:30:00	
	Session:	Team: (see below)	
Section Description: PRESSURIZER (UPPER HEAD, UPPER PLENUM, TOP PORTION OF UPPER RISER) .. NOT INCLUDED ARE STEAM INJECTION SYSTEM OR POOL COOLING	Leader: RY	Recorder: CCCJP	
	Organization: BNL		
	Location: BLDG 130		
Design Intention: GENERATES AND MAINTAINS NORMAL OPERATING SYSTEM PRESSURE BY MAINTAINING TEMPERATURE IN TOP FEW FEET OF WATER, PRESSURE RELIEF, CONTROL OF NON-CONDENSIBLES, LIMITS PRESSURE VARIATIONS DURING TRANSIENTS, MAINTAINS PZR POOL LEVEL, MAINTAINS RISER FUNNEL LEVEL, MAINTAINS SIPHON BREAKER LEVEL	P&ID's: Revision: PIUS 600 DWG, FIG 211.A, 89-12-15		
Team Members: <div style="display: flex; justify-content: space-between;"> <div>R. YOUNGBLOOD S. CHENG (AM ONLY)</div> <div>R. MORANTE B. LUCKAS</div> <div>B. CHAN P. SOO</div> <div>G VANTUYLE (AM ONLY)</div> </div>			

Section Desc: PRESSURIZER (UPPER HEAD, UPPER PLENUM, TOP PORTION OF UPPER RISER) NOT INCLUDED ARE STEAM INJECTION SYSTEM OR POOL COOLING

GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
MORE PZR POOL LEVEL	DROP IN PRESSURE AND MORE STEAM INJECTED	LOW PRESSURE SCRAM --				WHAT SYSTEM IS RESPONSIBLE FOR LEVEL CONTROL?
	REACTIVITY CONTROL SYSTEM ERROR	OVERFLOW IN RISER, OUTFLOW THROUGH DENSITY LOCKS PRESSURE INCREASE?		WJL	LOOK INTO POOL LEVEL CONTROL, SYSTEM #325	WHAT LEVEL IS CONTROLLED, RISER OR POOL?
	LOSS OF POOL COOLING	POOL TEMPERATURE SCRAM				IS CVCS LEVEL SENSING INDEPENDENT OF SCRAM LEVEL SENSING?
	MALFUNCTION OF POOL LEVEL CONTROL SYSTEM	INFLUX THROUGH THE DENSITY LOCKS				SCRAM IS ON POOL LEVEL.
	LOSS OF SECONDARY HEAT SINK	PRIMARY HEATS UP				
LESS PZR POOL LEVEL	BREAK IN PRESSURE BOUNDARY	PC				
	STEAM INJECTION PIPE BREAK	PC (SEE REVERSE FLOW)				
	REACTIVITY CONTROL SYSTEM ERROR	UNCOVER HOT LEG WHICH WILL LEAD TO A SHUTDOWN (UNCOVER POOL COOLING SYSTEM?)		GS	CHECK WHETHER YOU UNCOVER SUCTION TO POOL COOLING SYSTEM BEFORE HOT LEG BECOMES UNCOVERED	
	POOL LEVEL CONTROL SYSTEM ERROR	OUTFLOW THROUGH DENSITY LOCKS LEVEL DROPPED BELOW STEAM INJECTION POINT HAS UNDETERMINED CONSEQUENCES FOR PRESSURE CONTROL		WL	EXPLORE THESE UNDETERMINED CONSEQUENCES	
MORE RISER FUNNEL LEVEL	DROP IN PRESSURE AND MORE STEAM INJECTION	MORE POOL LEVEL				
	REACTIVITY CONTROL SYSTEM ERROR	FUNNEL OVERFLOW, OUTFLOW THROUGH DENSITY LOCKS...				
	LOSS OF SECONDARY HEAT SINK	FUNNEL OVERFLOW, INFLUX ...				
	STEAM INJECTION PIPE BREAK	BLOW DOWN THE PRIMARY THROUGH THE RISER FUNNEL, STRUCTURAL MISSILE/JET IMPINGEMENT ON VESSEL HEAD		ABB	EVALUATE LOADS FROM THIS EVENT AND ANALYZE VESSEL HEAD FOR STRUCTURAL DAMAGE	
	INADVERTENTLY OPENED RELIEF VALVE	PC (SEE MORE STEAM FLOW)				

Project: PLUS SYSTEM STUDY					
Section Desc: PRESSURIZER (UPPER HEAD, UPPER PLENUM, TOP PORTION OF UPPER RISER) NOT INCLUDED ARE STEAM INJECTION SYSTEM OR POOL COOLING					
GUIDEWORDS		CAUSE	CONSEQUENCE	RECOMMEND.	INDV
LESS RISER FUNNEL LEVEL	BREAK IN PRIMARY PIPING (NON-PRESSURIZER BREAK)	PC (SEE LESS POOL LEVEL)	PC (SEE LESS POOL LEVEL)	GS	LOOK INTO THIS
	REACTIVITY CONTROL SYSTEM ERROR	PC (SEE LESS POOL LEVEL)			
MORE SIPHON BREAKER LEVEL	DECREASE IN PRIMARY FLOW	POSSIBLE COLD WATER OF THE SIPHON BREAKER AND IMINGE ON THE PRESSURIZER DOME	RM	BC	LOOK INTO HEIGHTS AND PRESSURES CAN THE BROKEN SIPHON BREAKER TUBE JAM INSIDE THE RESTRICTED FLOW AREA
	BREAK IN THE WALL OF THE COLD LEG, THE COLD LEG, COMPROMISE OF SIPHON BREAKER FUNCTION	SIPHON BREAKER			
LESS SIPHON BREAKER LEVEL	INCREASE IN PRIMARY FLOW RATE	DENSITY LOCK, PC	AB8	CVT	NEEDS FURTHER REVIEW
	INCREASE IN PRIMARY SYSTEM PRESSURE	PC			
MORE STEAM DOME PRESSURE	TOO MUCH STEAM INJECTION	TRIP	PC	RM	THIS BREAK MAY GO UNDETECTED AND DISABLE THE SIPHON BREAKER. BREAK ANYWHERE ABOVE THE COLD LEG HAS THIS PROBLEM.
	LOSS OF SECONDARY HEAT SINK	BOILING IN THE PRIMARY AND INTRUSION OF UPPER DENSITY LOCKS			
LESS STEAM DOME PRESSURE	POWER INCREASE	HEAVY LIQUID SWELL, POSSIBLE STEAM, DENSITY LOOKS	PC	AB8	RE-EVALUATE ABOVE CONSEQUENCE
	FAILURE OF STEAM INJECTION SYSTEM	LOSS OF SUBCOOLING			
LESS STEAM DOME PRESSURE	BREAK OF STEAM INJECTION SYSTEM		PC	CVT	
	COLD LEG				

Section Desc: PRESSURIZER (UPPER HEAD, UPPER PLENUM, TOP PORTION OF UPPER RISER) NOT INCLUDED ARE STEAM INJECTION SYSTEM OR POOL COOLING

GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
LESS STEAM DOME PRESSURE (con't)	OPENING SAFETY RELIEF VALVE	PC				
	PRIMARY BREAK	PC				
	PREMATURE ELONGATION OF STEAM DOME BOLTS	RAPID DEPRESSURIZATION, PC?				
	POOL COOLING SYSTEM BREAK	PC				
	STEAM INJECTION SYSTEM BREAK	PC				
	DOMES FAILURE	DEPRESSURIZATION, LOSS OF SUBCOOLING				
OSCILLATING STEAM DOME PRESSURE	STEAM INJECTION CONTROLLER FAILURE	PC				
	CHUGGING RELIEF VALVE	PC				
	OSCILLATING POWER LEVEL	IF SUBCOOLING MARGIN IS LOW, OSCILLATIONS CAN BE LARGE; IF SUBCOOLING MARGIN IS GREAT, OSCILLATIONS ARE LIMITED				
	OSCILLATING WATER LEVEL (VOIDING AND COLLAPSING)	DISTURB DENSITY LOCKS, POTENTIAL FOR POWER OSCILLATION				
MORE STEAM DOME TEMPERATURE	MORE STEAM INJECTION	PC				
MORE PZR POOL TEMPERATURE	LOSS OF POOL COOLING FUNCTION	HEATUP OF UPPER POOL REGION IF POOL COOLING CIRCULATION IS LOST; SLOW HEATUP OF UPPER THIRD OF POOL IF CIRCULATION CONTINUES BUT HEAT REMOVAL IS LOST				
	FAILURE OF REACTOR THERMAL INSULATION	MORE HEAT LOAD ON POOL COOLING SYSTEM; ACTIVATE THE SECOND LOOP				
	SHUTDOWN	PRIMARY WATER OUTFLOW INTO POOL				
	MORE STEAM INJECTION	PC				
	LOSS OF SECONDARY HEAT SINK	PC				
OSCILLAT. STEAM DOME TEMP.	CYCLING OF STEAM INJECTION SYSTEM	CYCLING OF BOLTED FLANGE CONNECTION				

Project: PIUS SYSTEM STUDY			Section: 002		Page: 5	
Section Desc: PRESSURIZER (UPPER HEAD, UPPER PLENUM, TOP PORTION OF UPPER RISER) NOT INCLUDED ARE STEAM INJECTION SYSTEM OR POOL COOLING						
GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
OSCILLAT. STEAM DOME TEMP. (con't)	OSCILLATING RELIEF VALVE	PC				
OSCILLAT. POOL TEMPERATURE	OSCILLATING POOL COOLING SYSTEM: OSCILLATING SYSTEM FLOW	THERMAL CYCLING OF STEEL PRESSURE VESSEL NEAR REACTOR POOL WITHDRAWAL SYSTEM 325.3		WJL	DETERMINE WHAT CONTROLS AMOUNT OF COOLING	
	OSCILLATING POOL COOLING SYSTEM: CONSTANT FLOW, OSCILLATING HEAT REMOVAL	POOL RETURN WATER THERMAL OSCILLATIONS AFFECTING CAVITY LINER				
	STEAM INJECTION SYSTEM	PC				
	OSCILLATING PRESSURE RELIEF VALVE	PC				
	OSCILLATING RISER FUNNEL LEVEL	(1) PUMP SPEED CONTROL RESPONSE, (2) MAKEUP LEVEL CONTROL RESPONSE, (3) CHALLENGE UPPER DENSITY LOCKS				
OSCILLATING RISER FUNNEL TEMPERATURE	CORE POWER OSCILLATION	LARGE OSCILLATION WILL CHALLENGE DENSITY LOCKS				
	OSCILLATION OF SECONDARY SIDE HEAT REMOVAL	LARGE OSCILLATION WILL CHALLENGE DENSITY LOCKS				
	OSCILLATING PRIMARY PUMP SPEED	LARGE OSCILLATION WILL CHALLENGE DENSITY LOCKS				

Project: PIUS SYSTEM STUDY	Revision: 1	Section: 003	Page: 1						
	Date: 12/30/91	Time: 09:30:00							
	Session:	Team: (see below)							
Section Description: PRESSURIZER (UPPER HEAD, UPPER PLENUM, TOP PORTION OF UPPER RISER) .. NOT INCLUDED ARE STEAM INJECTION SYSTEM OR POOL COOLING	Leader: RY	Recorder: CC							
	Organization: BNL								
	Location: BLDG 130								
Design Intention: GENERATES AND MAINTAINS NORMAL OPERATING SYSTEM PRESSURE BY MAINTAINING TEMPERATURE IN TOP FEW FEET OF WATER, PRESSURE RELIEF, CONTROL OF NON-CONDENSIBLES, LIMITS PRESSURE VARIATIONS DURING TRANSIENTS, MAINTAINS PZR POOL LEVEL, MAINTAINS RISER FUNNEL LEVEL, MAINTAINS SIPHON BREAKER LEVEL	P&ID's: Revision: PIUS 600 DWG, FIG 211.A, 89-12-15								
Team Members: <table border="0"> <tr> <td>R. YOUNGBLOOD</td> <td>B. LUCKAS</td> <td>B. CHAN</td> </tr> <tr> <td>R. MORANTE</td> <td>P. SOO</td> <td>B. SHIER</td> </tr> </table>				R. YOUNGBLOOD	B. LUCKAS	B. CHAN	R. MORANTE	P. SOO	B. SHIER
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Section Desc: PRESSURIZER (UPPER HEAD, UPPER PLENUM, TOP PORTION OF UPPER RISER) NOT INCLUDED ARE STEAM INJECTION SYSTEM OR POOL COOLING

GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
LESS SIPHON BREAKER/STAND-PIPE TEMP.	LESS RISER FUNNEL TEMP	PC				
MORE VISCOSITY	N/A					
LESS VISCOSITY	N/A					
CONTAMINATION OF STEAM SPACE	FAILURE TO REMOVE NON-CONDENSIBLES	STIFFER (FASTER) PRESSURE RESPONSE MORE LIKELY TO POP A RELIEF VALVE OXYGEN CONTAMINATION CAN LEAD TO AN EXPLOSION				
	GAP RELEASE	CONTAMINATE CONTAINMENT IF RELIEF VALVE OPENS INCREASE WORKER EXPOSURE DUE TO BUILDUP OF CONTAMINATION IN SYSTEM; NEED TO DECONTAMINATE MORE OFTEN				
CONTAMINATION OF RISER FUNNEL	GAP RELEASE	PC				
CONTAMIN. OF SIPHON BREAKER	N/A					
CONTAMIN. OF PZR POOL	CONTAMINATION FROM STEAM INJECTION	CONSEQUENCES DEPENDS ON THE NATURE OF THE CONTAMINANTS AND THIS WILL BE HAZOPED IN THE STEAM INJECTION SYSTEM; POSSIBLE CONSEQ. INCLUDE CORROSION AT DOME FLANGE				
RELIEF			SYS 314 - WOULD SOMEONE DETERMINE THE DESIGN BASIS OF THE PZR RELIEF VALVES			
INSTRUM/CONTROL				WL,WS	GENERATE A HIGH LEVEL SCHEMATIC OF PROCESS SIGNALS WHICH HAVE CONTROL FUNCTIONS	
SAMPLING				RY	GENERATE A HIGH LEVEL SCHEMATIC OF SAMPLED QUANTITIES WHICH HAVE CONTROL FUNCTIONS, SEE SYS 336	

Project: PIUS SYSTEM STUDY			Section: 003		Page: 4	
Section Desc: PRESSURIZER (UPPER HEAD, UPPER PLENUM, TOP PORTION OF UPPER RISER) NOT INCLUDED ARE STEAM INJECTION SYSTEM OR POOL COOLING						
GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
CORROS./EROSION	STEAM IMPINGEMENT ON RISER FUNNEL	BIG HOLE WILL REDUCE PRESSURE IN THE LOOP AND CAUSE INGRESS THROUGH DENSITY LOCKS SMALL HOLE MAY GO UNDETECTED BECAUSE IT IS COMPENSATED FOR BY THE POOL LEVEL CONTROL SYSTEM & THE REACTIVITY CONTROL SYSTEM				
	STEAM-INDUCED EROSION AT THE STEAM INJECTION INLET	POPPING THE LINE OFF, PC--REFER TO STEAM INJECTION LINE BREAK				
	CORROSION ON EXTERIOR OF PZR OR BOLTS FROM BORIC ACID LEAKAGE THROUGH FLANGE	DOME RUPTURE OR LEAKAGE				

Project: PIUS SYSTEM STUDY	Revision: 1	Section: 004	Page: 1						
	Date: 12/30/91	Time: 15:24:23							
	Session:	Team: (see below)							
Section Description: PRESSURIZER (UPPER HEAD, UPPER PLENUM, TOP PORTION OF UPPER RISER) ** NOT INCLUDED ARE STEAM INJECTION SYSTEM OR POOL COOLING	Leader: RY	Recorder: CC							
	Organization: BNL								
	Location: BLDG 130								
Design Intention: GENERATES AND MAINTAINS NORMAL OPERATING SYSTEM PRESSURE BY MAINTAINING TEMPERATURE IN TOP FEW FEET OF WATER, PRESSURE RELIEF, CONTROL OF NON-CONDENSIBLES, LIMITS PRESSURE VARIATIONS DURING TRANSIENTS, MAINTAINS PZR POOL LEVEL, MAINTAINS RISER FUNNEL LEVEL, MAINTAINS SIPHON BREAKER LEVEL	P&ID's: PIUS 600 DWG, FIG 211.A, 89-12-15 Revision:								
Team Members: <table> <tr> <td>R. YOUNGBLOOD</td> <td>B. LUCKAS</td> <td>B. CHAN</td> </tr> <tr> <td>R. MORANTE</td> <td>P. SOO</td> <td>B. SHIER</td> </tr> </table>				R. YOUNGBLOOD	B. LUCKAS	B. CHAN	R. MORANTE	P. SOO	B. SHIER
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Project: PIUS SYSTEM STUDY				Section: 004		Page: 2
Section Desc: PRESSURIZER (UPPER HEAD, UPPER PLENUM, TOP PORTION OF UPPER RISER) NOT INCLUDED ARE STEAM INJECTION SYSTEM OR POOL COOLING						
GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
RADIATION	CONTAMINATION (GAP RELEASE)	PC ..				
STRUCT. INTEGR.	OVERPRESSURE: FAILURE OF PRESSURE RELIEF SYSTEM IN CONJUNCTION WITH A SEVERE TRANSIENT	PREFERABLY HEAD BOLT ELONGATION TO RELIEVE PRESSURE; POTENTIALLY DOME RUPTURE				
	FLANGE INTEGRITY IS THREATENED BY THERMAL CYCLING DUE TO LEVEL OSCILLATION	POSSIBLE DOME FAILURE				
	(VARIETY OF BREAKS AND DEPRESSURIZATION EVENTS HAVE BEEN CALLED OUT UNDER OTHER GUIDEWORDS)	PC				
	INTERNAL DELTA P'S GENERATED DURING HOT LEG BREAK	POSSIBLE EFFECTS ON MANY NODES		GVT	MAJOR RESEARCH PROJECT TO ESTABLISH LOADS	
GEOMETRY	PC					
CIRCULATION	SPILOVER FROM RISER FUNNEL	PC				
	LEAKAGE THROUGH RISER FUNNEL	PC				
THERMAL STRATIFICATION	PC					
BORON STRATIFICATION	NORMAL OPERATION MAY LEAD TO BORON CONCENTRATION IN UPPER PZR POOL	WE ARE NOT SURE WE CARE				
FLOW TILT	N/A					
POWER TILT	N/A					
SERVICE FAILURE	LOSS OF SUPPORTS TO RELIEF VALVES	MORE LIKELY TO LIFT SAFETIES				
		POSSIBLE OVERPRESSURE				
ABNORM. OP.	N/A AS FAR AS WE KNOW					
MAINTENANCE	N/A					
IGNITION	HYDROGEN/OXYGEN MIXTURE IN THE STEAM DOME SPACE	OVERPRESSURES		JB	LOOK UP AUTO-IGNITION TEMP OF HYDROGEN	
SPARE EQUIP.	N/A					
EXTERNAL EVENTS	SEISMIC	STRUCTURAL FAILURE				
	EXTERNAL MISSILE	STRUCTURAL FAILURE				
SAFETY	NONE IDENTIFIED					

Project: PRESSURIZER HEATING SYSTEM (3283)	Revision: 2	Section: 001	Page: 1										
	Date: 02/14/92	Time: 10:00:00											
	Session:	Team: (see below)											
Section Description: BOILER, BOILER FEED PUMPS, ASSOCIATED PIPING AND VALVES (REVISIT LOWER REACT. THROUGH OSCILLAT. FLOW AND CONSIDER ENTIRE SYSTEM, NOT JUST BOILER)	Leader: RY	Recorder: CC											
	Organization: BNL												
	Location: BLDG 130												
Design Intention: MAINTAIN TEMP & MIXTURE OF WATER & STEAM, HEATING TOP 3 METERS, MAINTAIN PRIMARY SYSTEM PRESSURE AT 90 BARS, RECIRCULATE PZR WATER, MAINTAIN BOILER PRES. OF 100 BARS, MAINTAIN BOILER LEVEL, LIQUID RETURN LINE TO MAINTAIN PUMP MINI-FLOW & RECIRCULATE POOL WATER TO PREVENT BORON BUILDUP IN BOILER	P&ID's:		Revision:										
	ABB AA 245 889												
Team Members: <table border="0"> <tr> <td>R. YOUNGBLOOD</td> <td>R. MORANTE</td> <td>G. VAN TUYLE</td> <td>B. LUCKAS (REV1)</td> <td>V. WILKINSON (ORNL)</td> </tr> <tr> <td>P. SOO</td> <td>A. SZUKIEWICZ (REV1)</td> <td>S. CHENG</td> <td>B. SHIER</td> <td>M. OLSZEWSKI (ORNL)</td> </tr> </table>				R. YOUNGBLOOD	R. MORANTE	G. VAN TUYLE	B. LUCKAS (REV1)	V. WILKINSON (ORNL)	P. SOO	A. SZUKIEWICZ (REV1)	S. CHENG	B. SHIER	M. OLSZEWSKI (ORNL)
R. YOUNGBLOOD	R. MORANTE	G. VAN TUYLE	B. LUCKAS (REV1)	V. WILKINSON (ORNL)									
P. SOO	A. SZUKIEWICZ (REV1)	S. CHENG	B. SHIER	M. OLSZEWSKI (ORNL)									

Project: PRESSURIZER HEATING SYSTEM (3283)					Section: 001	Page: 2
Section Desc: BOILER, BOILER FEED PUMPS, ASSOCIATED PIPING AND VALVES (REVISIT LOWER REACT. THROUGH OSCILLAT. FLOW AND CONSIDER ENTIRE SYSTEM, NOT JUST BOILER)						
GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
BOILER:		..				ALL FOUR LINES PROBABLY HAVE THE CONSEQUENCES PREVIOUSLY ASSOCIATED WITH STEAM INJECTION LINE BREAK
LOWER REACT	N/A					
HIGHER REACT	N/A					
MORE BORON	BOTH PUMPS FAIL	SHUT OFF HEATERS ON LOW BOILER LEVEL				
		LOSE SYSTEM PRESSURE, SHUTDOWN				
	PLUG RECIRCULATION LINE	HEATERS TURN OFF				
		BOILER GOES SOLID				
		LOSE SYSTEM PRESSURE, SHUTDOWN				
	BOILER STEAM SPACE BREAK	LOSE PRIMARY PRESSURE CONTROL, SHUTDOWN				QUESTION WHETHER V13 WILL CLOSE UNDER THESE CONDITIONS WITH HOT WATER FLASHING TO STEAM
		POSSIBLE BLOWDOWN OF PRIMARY IF V13 DOES NOT CLOSE				IS THERE A NEED FOR A REDUNDANT ISOLATION MECHANISM IN THE FEEDLINE FOR BOILER STEAM SPACE BREAKS?
		POSSIBLE BLOWDOWN OF PRIMARY IF V17 DOES NOT CLOSE				
	BREAK IN STEAM LINE	PC				
	EXCESS HEATERS	HIGHER PRESSURE IN BOILER				
		PRESSURE WILL INCREASE IN THE PRIMARY				
		POTENTIAL TO STICK OPEN PRIMARY RELIEF VALVE				QUESTION WHAT CONTROLS V14, DOES V14 CLOSE ON EXCESS BOILER PRESSURE?
LESS BORON	NO CAUSES KNOWN TO HAVE APPRECIABLE CONSEQUENCES					
NO FLOW	N/A					
REVERSE FLOW	N/A					
MORE FLOW	N/A					

Project: PRESSURIZER HEATING SYSTEM (3283)			Section: 001		Page: 3	
Section Desc: BOILER, BOILER FEED PUMPS, ASSOCIATED PIPING AND VALVES (REVISIT LOWER REACT. THROUGH OSCILLAT. FLOW AND CONSIDER ENTIRE SYSTEM, NOT JUST BOILER)						
GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
LESS FLOW	N/A	..				
OSCILL. FLOW	N/A					
MORE LEVEL	SECOND PUMP STARTING WHILE FIRST PUMP STILL RUNNING	MAY ACTUALLY CONTROL PROPERLY, NOT SURE				
	LESS RECIRCULATION	PC				
	LESS STEAM FLOW	PRESSURE IN PRIMARY TRENDING DOWNWARDS, SHUTDOWN				
	V9 OPEN TOO FAR	BOILER TRENDS TOWARD A WATER SOLID CONDITION, CHALLENGE RELIEF VALVE V11 POSSIBLY WITH LIQUID				
LESS LEVEL		HEATER SHUTOFF ON HIGH BOILER PRESSURE				
	BREAK IN THE BOILER	PRIMARY INVENTORY LOSS THROUGH PUMP FEED LINE				
		DEPRESSURIZATION EVENT				
	LESS INLET FLOW	HEATER SHUTOFF				
		PRIMARY PRESSURE LOSS, PLANT SHUTDOWN				
	BREAK IN RECIRCULATION LINE DOWNSTREAM OF CHECK VALVE	V20 SHOULD CLOSE TO PREVENT PRIMARY INVENTORY LOSS				
		BOILER BLOWS DOWN AND PRIMARY BLOWS DOWN THROUGH THE BOILER				
		SEE DISCUSSION UNDER BOILER BREAK				
	BREAK IN RECIRCULATION LINE NEAR VESSEL	PC				
	BREAK IN FEED LINE NEAR VESSEL	PC				
	BREAK IN FEED LINE DOWNSTREAM OF ISOLATION VALVE	V1 SHOULD CLOSE, OTHERWISE YOU WILL BLOW DOWN THE PRIMARY				
	BREAK IN STEAM SPACE OR OPEN RELIEF VALVE	V1 SHOULD ISOLATE, OTHERWISE THERE WILL BE PRIMARY INVENTORY LOSS				
	POOL LEVEL DROPPING BELOW TAKEOFF POINT	REACTOR SHUTDOWN				
		PUMPS WILL BE STEAM-BOUND OR WORSE				

Project: PRESSURIZER HEATING SYSTEM (3283)					Section: 001	Page: 4
Section Desc: BOILER, BOILER FEED PUMPS, ASSOCIATED PIPING AND VALVES (REVISIT LOWER REACT. THROUGH OSCILLAT. FLOW AND CONSIDER ENTIRE SYSTEM, NOT JUST BOILER)						
GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
LESS LEVEL (con't)	POOL LEVEL DROPPING BELOW TAKEOFF POINT	POSSIBILITY OF SIPHON BREAKER LEVEL DROPPING TO VENTURI, FOR EXAMPLE?		SC	CHECK DIMENSIONS AND CONTROLLER SETPOINTS TO SUPPORT DISCUSSION OF POOL LEVEL DROP	
	TOO MUCH HEATING	INCREASED FEED TO TRY TO MAINTAIN BOILER LEVEL MORE STEAM TO PZR				

Project: PRESSURIZER HEATING SYSTEM (3283)	Revision: 2	Section: 002	Page: 1
	Date: 02/14/92	Time: 13:30:00	
	Session:	Team:	(see below)
Section Description: BOILER, BOILER FEED PUMPS, ASSOCIATED PIPING AND VALVES ..	Leader: RY	Recorder: CC	
	Organization: BNL		
	Location: BLDG 130		
Design Intention: MAINTAIN TEMP & MIXTURE OF WATER & STEAM, HEATING TOP 3 METERS, MAINTAIN PRIMARY SYSTEM PRESSURE AT 90 BARS, RECIRCULATE PZR WATER, MAINTAIN BOILER PRESSURE OF 100 BARS, MAINTAIN BOILER LEVEL, LIQUID RETURN LINE TO MAINTAIN PUMP MINI-FLOW & RECIRCULATE POOL WATER TO PREVENT BORON BUILDUP IN BOILER	P&ID's:	Revision:	
	ABB AA 245 889		
Team Members:			
R. YOUNGBLOOD P. SOO	R. MORANTE S. CHENG	G. VAN TUYLE B. SHIER	V. WILKINSON (ORNL) M. OLSZEWSKI (ORNL)

Project: PRESSURIZER HEATING SYSTEM (3283)				Section: 002		Page: 2
Section Desc: BOILER, BOILER FEED PUMPS, ASSOCIATED PIPING AND VALVES						
GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
MORE PRES.	RUNAWAY HEATERS	PC ..				P. 538-2, THE REACTOR PRESSURE CONTROL SYSTEM IS NOT YET DESIGNED, EXAMPLE, "THERE MAY BE A PZR SPRAY SYSTEM IN THE FUTURE"
	COMPLETE CLOSURE OF STEAM LINE INCLUDING V14	DOWNWARD TREND OF PRIMARY PRESSURE (PC) POSSIBLE CHALLENGE TO V11 (SEE COMMENT)				
	STARTUP OF SECOND PUMP	PC				
	RECIRCULATION LINE BLOCKAGE	PC				
LESS PRES.	BREAK IN BOILER, ASSOCIATED LINES	PC				
	LOSS OF PUMP FLOW	PC				
	LOSS OF HEATERS	PC				
	LESS LEVEL	PC				
	LOSE FEED FLOW					
	INCREASED STEAM DEMAND	PC?				
	TREMENDOUS INCREASE IN STEAM FLOW THROUGH STEAM LINE EITHER BY INCREASED DEMAND OR LINE PERMITTING TOO MUCH FLOW					
OSCILLAT. PRESSURE.	OSCILLATIONS IN HEATER CONTROLLER	GRADUAL AND WE DON'T CARE				
	CHUGGING RELIEF VALVE	SMALL LOSS OF INVENTORY				
	OSCILLATING V14	SMALL OSCILLATIONS OF PRIMARY				
	OSCILLATIONS IN THE PZR WHICH GENERATES OSCILLATORY BEHAVIOR OF V14 (STEAM CONTROL VALVE)	PC				
	OSCILLATING POWER OF REACTOR	PC				
MORE TEMP.	EXCESS POWER TO HEATERS	INCREASE IN BOILER PRESSURE; CHALLENGE PRESSURE RELIEF				
	REDUCED INVENTORY	PC				
LESS TEMP.	HEATER FAILURE	PC				

Section Desc: BOILER, BOILER FEED PUMPS, ASSOCIATED PIPING AND VALVES

GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
LESS TEMP. (con't)	LESS POWER TO HEATERS	PC ..				
	MORE INVENTORY DUE TO CONTROLLER MALFUNCTION	PC				
	LOWER INLET TEMP DUE TO UNSPECIFIED CIRCULATION IN THE POOL	WE NEED NUMBERS TO DETERMINE CONSEQUENCES OF THIS				
	LESS PRESSURE	PC				
OSCILLAT TEMP	SEE OSCILLATING PRESSURE					
MORE VISC.	N/A					
LESS VISC.	N/A					
CONTAMINAT.	NON-CONDENSIBLES IN THE BOILER	IF THIS CAN OCCUR THEN CHALLENGE OF RELIEF VALVE WOULD RELEASE COMBUSTIBLE GAS				
	CONTAMINATED INLET WATER	CHALLENGE OF RELIEF VALVE WOULD RELEASE CONTAMINATED STEAM				
	SOMETHING THROUGH THE PUMP OIL	HEAVIER THAN WATER CONTAMINANT WOULD GUM UP PRESSURE TAP AT BOTTOM OF VESSEL; POSSIBLE LEVEL CONTROL ERROR				
RELIEF	NO SIGNIFICANT ISSUES IDENTIFIED					
INSTRUM/CONT						CONSIDER MEASURING FLOW IN FEEDLINE, IN STEAM LINE, AND IN RECIRCULATION LINE FOR DIAGNOSTIC PURPOSES
						CONSIDER MONITORING ACTIVATION OF RELIEF VALVE
SAMPLING						
CORROS/EROS	LEAKAGE OF BORATED WATER CAN LEAD TO RAPID WASTAGE OF CARBON STEEL, E.G., HEATER MOUNTING FLANGES	POSSIBLE INVENTORY LOSS				CONCERN WITH LEAKAGE AROUND HEATER FLANGES DUE TO CORROSION; CORROSION PRODUCTS AFFECTING SENSORS
	EROSION OF VENTURI	LEAD TO TOO MUCH RECIRCULATION				
	BUILDUP OF OXIDE ON HEATER SURFACE ADVERSELY AFFECTING SYSTEM CONTROL	SLUGGISH CONTROL RESPONSE				

Project: PRESSURIZER HEATING SYSTEM (3283)				Section: 002		Page: 4
Section Desc: BOILER, BOILER FEED PUMPS, ASSOCIATED PIPING AND VALVES						
GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
RADIATION	CONTAMINATED POOL WATER	PC				NEED DESIGN/FABRICATION DWGS TO SHOW PHYSICAL LAYOUT
STRUCT. INTEG.	VALVE SLAMMING LEADING TO PRESSURE TRANSIENT	POSSIBLE VIOLATION OF STRUCT. INTEG.	FURTHER ANALYSIS REQUIRED			
	VIBRATION IN AREA OF PUMPS	POSSIBLE BREAK IN PRIMARY				
GEOMETRY	LOOSE CHECK VALVE DISK	BLOCK ANY OF THREE LINES				
	HEATERS CONTINUING TO RUN IN DRY BOILER (NO FLOW THRU FEEDLINE SEGMENT)	LOSS OF BOILER INTEGRITY				
CIRCULATION	N/A					
STRATIFICATION	N/A					
FLOW TILT	N/A					
POWER TILT	N/A					
SERV. FAIL.	LOSS OF HEATER POWER	PC				
	LOSS OF CONTROL POWER	UNPREDICTABLE CONTROL SYSTEM RESPONSE: PRIMARY PRESSURE, HEATER POWER (BOILER PRESSURE), BOILER LEVEL				
		MAY AFFECT ISOLATION VALVE LOGIC AS WELL				
	LOSS OF MOTIVE POWER TO THE ISOLATION VALVES	INABILITY TO ISOLATE				
	LOSS OF MOTIVE POWER OF PUMPS	LOSS OF FEEDLINE FLOW				
ABNORM. OP.	N/A					
MAINTENANCE	N/A					
IGNITION	IGNITION OF NON-COMBUSTIBLES COMING OUT OF THE RELIEF VALVE	PC				
	POWER CABLE FIRES, PUMP FIRES	CAN'T REALLY ASSESS CONSEQUENCES				
SPARE EQUIP.	NO SIGNIFICANT SPARE EQUIPMENT ISSUES					
EXTERNAL EVENTS	SEISMIC QUALIFICATION AND ASSOCIATED PIPING IS AN ISSUE					

Project: PRESSURIZER HEATING SYSTEM (3283)					Section: 002	Page: 5
Section Desc: BOILER, BOILER FEED PUMPS, ASSOCIATED PIPING AND VALVES						
GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
SAFETY	NO MAJOR ISSUES IDENTIFIED	.				

Project: PIUS SYSTEM STUDY	Revision:	Section: 001	Page: 1
	Date: 12/04/91	Time: 09:52:49	
	Session:	Team: (see below)	
Section Description: UPPER AND LOWER DOWNCOMERS (UDC AND LDC) ..	Leader: RM	Recorder: CC	
	Organization: BNL		
	Location: BLDG 130		
Design Intention: SUPPLY PRIMARY COOLING WATER TO THE CORE, TAKES WATER FROM THE COLD LEG AND DELIVERS THE WATER TO THE CORE ENTRANCE, PROVIDES SUFFICIENT DRIVING HEAD FOR THE MAIN COOLING LOOP, HOUSES SIPHON BREAKER WHICH WILL PREVENT REVERSE FLOW IN CASE OF COLD LEG BREAK	P&ID's:	Revision:	
	S. CHENG'S PIPA	12/4/91	
Team Members:			
R. MORANTE	P. SOO	G. VAN TUYLE (11-12)	
B. LUCKAS	S. CHENG	J. HIGGINS (11-12)	

Section Desc: UPPER AND LOWER DOWNCOMERS (UDC AND LDC)

GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
LOWER REACTIV.	N/A	..				
HIGHER REACTIV.	N/A					
MORE BORON	MALFUNCTION IN REACTIVITY CONTROL SYSTEM	REDUCE POWER		SC,WL	FURTHER INVESTIGATION OF RAMIFICATIONS OF INADVERTENT BORON INJECTION IN PRIMARY LOOP	
	NORMAL REDUCTION OF POWER THROUGH THE REACTIVITY CONTROL SYSTEM	REDUCE POWER				
	ACTIVATION OF THE POOL LOOP	REDUCE POWER SHUTDOWN				
LESS BORON	MALFUNCTION IN REACTIVITY CONTROL SYSTEM	INCREASE POWER		SC,WL	FURTHER INVESTIGATION OF RAMIFICATIONS OF INADVERTENT BORON DILUTION IN PRIMARY LOOP	
	NORMAL INCREASE OF POWER THROUGH THE REACTIVITY CONTROL SYSTEM	INCREASE POWER				
NO FLOW	ACTIVATION OF SIPHON BREAKER DUE TO COLD LEG BREAK	PC				
	COLD SHUTDOWN	OK				
REVERSE FLOW	COLD LEG BREAK PRIOR TO ACTIVATION OF SIPHON BREAKER	LOSS OF PRIMARY COOLANT UNTIL ACTIVATION OF SIPHON BREAKER				
MORE FLOW	PLANNED INCREASE IN PUMP SPEED	CHANGE IN POWER				
	MALFUNCTION IN PUMP CONTROLLER	PUSH WATER THROUGH LOWER DENSITY LOCKS INTO THE POOL, LEADING TO A SLOW SHUTDOWN				
	MALFUNCTION OF REACTIVITY CONTROL SYSTEM	?			TEAM WILL DISCUSS THIS UNDER REACTIVITY SYSTEM MODE	
LESS FLOW	BREAK IN DOWNCOMER WALL	ACTIVATE DENSITY LOCK				EXPECTED MINIMAL CONSEQUENCES, BUT MAY REQUIRE FURTHER STUDY
	BREAK IN THE COLD LEG	PC (REFER TO REVERSE FLOW)				

Section Desc: UPPER AND LOWER DOWNCOMERS (UDC AND LDC)

GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
LESS FLOW (con't)	MALFUNCTION OF PUMP CONTROLLER	PULL WATER FROM THE POOL INTO THE CORE THROUGH THE LOWER DENSITY LOCKS, LEADING TO SHUTDOWN				
	LOSS OF PUMP	PUSH WATER FROM THE POOL INTO THE CORE THROUGH THE LOWER DENSITY LOCKS, LEADING TO SHUTDOWN				
	MALFUNCTION OF REACTIVITY CONTROL SYSTEM				TEAM WILL DISCUSS THIS UNDER REACTIVITY SYSTEM NODE	
	PLANNED DECREASE IN PUMP SPEED	CHANGE IN POWER				
	PRIMARY LOOP BREAK ANYWHERE OUTSIDE OF THE REACTOR, EXCEPT COLD LEG BREAK	ACTIVATE DENSITY LOCK, LEADING TO SHUTDOWN RAPID DEPRESSURIZATION, LEADING TO VOIDING POTENTIAL FOR STRUCTURAL DAMAGE		GVT	FURTHER EVALUATION NEEDED	THIS APPEARS TO BE A DESIGN-BASIS ACCIDENT CONDITION
	FLOW BLOCKAGE IN PRIMARY LOOPS	ACTIVATE DENSITY LOCK, LEADING TO SHUTDOWN				
	STEAM GENERATOR TUBE RUPTURE	CONTAMINATION OF SECONDARY SIDE DEPRESSURIZATION PRIMARY INVENTORY LOSS				
OSCILLAT. FLOW	BREAK IN COLD LEG	POTENTIAL TO DEFEAT THE SIPHON BREAKER IF THREE INTACT LOOPS CONTINUED IN OPERATION (THIS IS A POTENTIAL CONSEQUENCE FOR ANY "BREAK IN COLD LEG" CAUSE		GVT	NEED TO UNDERSTAND HOW THE SIPHON BREAKER PREVENTS FLOW FROM THREE INTACT LOOPS	CAN THE COLD LEG BREAK BE FED BY CONTINUED OPERATION OF THE THREE INTACT LOOPS EVEN IF THE SIPHON BREAKER HAS BEEN ACTIVATED AND IS FUNCTIONING?
	MALFUNCTION OF PUMP CONTROLLER	POTENTIAL FOR LOWER DENSITY LOCK DAMAGE AND/OR LOWER CORE DAMAGE DUE TO FLOW INDUCED VIBRATION				
	MALFUNCTION OF REACTIVITY CONTROL SYSTEM				WILL CONSIDER UNDER THE REACTIVITY CONTROL SYSTEM NODE	
MORE LEVEL	N/A					
LESS LEVEL	N/A					

Project: PIUS SYSTEM STUDY			Section: 001		Page: 4	
Section Desc: UPPER AND LOWER DOWNCOMERS (UDC AND LDC)						
GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
MORE PRESSURE	INCREASED PRESSURE IN THE PRESSURIZER	SEE PRESSURIZER NODE ..				
	HIGHER PUMP SPEED	PC (REFER TO MORE FLOW)				
LESS PRESSURE	DECREASED PRESSURE IN THE PRESSURIZER	SEE PRESSURIZER NODE				
	LOWER PUMP SPEED	PC (REFER TO LESS FLOW)				
	BREAK ANYWHERE IN THE PRIMARY LOOP	PC (REFER TO LESS FLOW)				
	BREAK IN DOWNCOMER WALL	PC (REFER TO LESS FLOW)				
OSCILLAT. PRESS	OSCILLATING PRESSURE IN THE PRESSURIZER	SEE PRESSURIZER NODE				
	BREAK IN THE COLD LEG	PC (REFER TO OSCILLAT. FLOW)				
	MALFUNCTION OF PUMP CONTROLLER	PC (REFER TO OSCILLAT. FLOW)				
MORE TEMP.	DECREASE IN SECONDARY SIDE LOAD	POWER REDUCTION				
	MALFUNCTION OF REACTIVITY CONTROL SYSTEM	??				CONSIDERED AS A CAUSE, BUT DISMISSED BASED ON USE OF REGENERATIVE HEAT EXCHANGERS AND LOW FLOW
LESS TEMP.	INCREASED TEMPERATURE IN THE RISER	INCREASE IN CORE INLET TEMPERATURE			RM NEEDS FURTHER EXPLANATION	
	INCREASE IN SECONDARY SIDE LOAD	PC				
	PLANNED INCREASE IN PUMP FLOW	PC				
	MALFUNCTION OF PUMP CONTROLLER	PC				
	MALFUNCTION OF REACTIVITY CONTROL SYSTEM	TBC (TO BE CONSIDERED)				
	LOSS OF REACTOR INSULATION	REMOVE HEAT FROM THE DOWNCOMER				
		REACTOR POWER INCREASE				IT IS POSSIBLE THAT THE REACTOR WILL FIND A NEW STABLE OPERATING POINT
	POOL HEATUP					
	INSULATION COLLECTS ON SUPPORT SKIRT; FLOW IMPEDANCE IN SKIRT					

Section Desc: UPPER AND LOWER DOWNCOMERS (UDC AND LDC)

GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
LESS TEMP. (con't)	LOSS OF REACTOR INSULATION	..				
	DECREASED TEMPERATURE IN THE RISER	PC				
OSCILLAT. TEMP.	MALFUNCTION OF REACTIVITY CONTROL SYSTEM	TBC				
	MALFUNCTION OF PUMP CONTROLLER	PC				
	TURBINE BYPASS MALFUNCTION	REACTOR POWER OSCILLATION PROBABLE SHUTDOWN				
	OTHER SECONDARY SIDE UPSETS	PC				

Project: PIUS SYSTEM STUDY	Revision:	Section: 002	Page: 1
	Date: 12/06/91	Time: 10:03:45	
	Session:	Team: (see below)	
Section Description: UPPER AND LOWER DOWNCOMERS (UDC AND LDC)	Leader:	Recorder: CC	
	Organization: BNL		
	Location: BLDG 130		
Design Intention: SUPPLY PRIMARY COOLING WATER TO THE CORE, TAKES WATER FROM THE COLD LEG AND DELIVERS THE WATER TO THE CORE ENTRANCE, PROVIDES SUFFICIENT DRIVING HEAD FOR THE MAIN COOLING LOOP, HOUSES SIPHON BREAKER WHICH WILL PREVENT REVERSE FLOW IN CASE OF COLD LEG BREAK	P&ID's:	Revision:	
	S. CHENG'S PIPA	12/6/91	
Team Members: R. MORANTE P. SOO (TILL 11 AM) R. YOUNGBLOOD B. LUCKAS S. CHENG			

Project: PIUS SYSTEM STUDY				Section: 002		Page: 2
Section Desc: UPPER AND LOWER DOWNCOMERS (UDC AND LDC)						
GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
MORE VISC.	N/A	..				
LESS VISC.	N/A					
CONTAMINATION	SEE CONTAMINATION GUIDE WORD IN CORE CHANNEL NODE	PC				
	SCALE COLLECTION IN THE ELBOW OF THE LOWER PLENUM DOWNCOMER INTERFACE	POSSIBLE BLOCKAGE OF PIPING TO LOWER DENSITY LOCK				PREVIOUSLY NOTED ACTION TO INVESTIGATE THAT
RELIEF	N/A					
INSTRUM/CONTROL				TEAM	REVISIT THIS GUIDEWOR	ALL KINDS OF INCCRE INSTRUMENTATION GOES THROUGH THE DOWNCOMER
SAMPLING	N/A ?					
CORROS/EROS	FLOW HITTING SIDE OF SIPHON BREAKER	PC (REFER TO PRESSURIZER NODE)				
	EROSION AT TIP OF SIPHON BREAKER IN THE THROAT OF THE VENTURI	PC (REFER TO PRESSURIZER NODE)				
	FAILURE OF PRIMARY CHEMISTRY CONTROL	PC				
RADIATION	FUEL FAILURE	PC (REFER TO CORE CHANNEL NODE)				
STRUCT. INTEG.	UNEXPECTED OSCILLATIONS IN FLOW, TEMPERATURE, AND/OR PRESSURE	POTENTIAL FOR FATIGUE FAILURES AND GENERATION OF LOOSE PARTS				
	LARGE PRIMARY BREAK	CRUSH DOWNCOMER				
	LEAKAGE IN JOINTS IN SECTIONS OF REACTOR ENVELOPE STRUCTURE	LEAKAGE FROM DOWNCOMER TO RISER				
		LEAKAGE OUT OF DOWNCOMER INTO THE POOL				
		EFFECTS ON INSULATION?		RM	FIND OUT HOW THE INSULATION IS PUT ON	IF INSULATION IS SECTIONED, THEN THERE WOULD BE A THERMAL GRADIENT AT THE JOINT
GEOMETRY	N/A					
MORE CIRCULATION	BIGGER HOLE BETWEEN RISER AND DOWNCOMER	SEE ACTION ITEM		SC	ASK ABB WHAT THE DEAL IS	REF P. 213-19
LESS CIRCULATION	BLOCKAGE OF HOLE BETWEEN RISER AND DOWNCOMER	SEE ACTION ITEM FOR MORE CIRCULATION				
STRATIFICATION	N/A					

Project: PIUS SYSTEM STUDY					Section: 002	Page: 3
Section Desc: UPPER AND LOWER DOWNCOMERS (UDC AND LDC)						
GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
FLOW TILT	CAN'T THINK OF ANY CAUSES	..				
POWER TILT	N/A					
SERVICE FAIL.	N/A					
ABNORM. OP.	TBC					
MAINTENANCE	TBC					
IGNITION	N/A					
SPARE EQUIP.	N/A					
EXTERNAL EVENTS	EARTHQUAKE	PC				
SAFETY	N/A					

Project: PIUS SYSTEM STUDY	Revision:	Section: 001	Page: 1								
	Date: 11/25/91	Time: 15:04:06									
	Session:	Team: (see below)									
Section Description: CORE CHANNEL 1 (CR1-1A) ..	Leader: RY	Recorder: CC									
	Organization: BNL										
	Location: BLDG 130										
Design Intention: OPERATING AT CRITICALITY, POWER LEVEL SHOULD BE NOMINAL FULL POWER, PRESSURE 9 MEGA PASCAL, 13,200 KG PER SEC, INLET TEMP 260, OUTLET TEMP 290, BORON CONCENTRATION 375 PPM, FUEL TEMP 469, 1000 MWT, NO NITROGEN, NO OXYGEN, NOMINAL VISCOSITY, POWER TILT-NONE, FLOW TILT-NONE, AMBIENT CONT,	P&ID's:	Revision:									
	S CHENG'S PIPA	11/25/91									
Team Members: <table> <tr> <td>R. YOUNGBLOOD</td> <td>G. VAN TUYLE</td> <td>B. LUCKAS</td> <td>P. SOO</td> </tr> <tr> <td>R. MORANTE</td> <td>S. CHENG</td> <td>B. CHAN</td> <td>R. FULLWOOD</td> </tr> </table>				R. YOUNGBLOOD	G. VAN TUYLE	B. LUCKAS	P. SOO	R. MORANTE	S. CHENG	B. CHAN	R. FULLWOOD
R. YOUNGBLOOD	G. VAN TUYLE	B. LUCKAS	P. SOO								
R. MORANTE	S. CHENG	B. CHAN	R. FULLWOOD								

Project: PIUS SYSTEM STUDY					Section: 001	Page: 2
Section Desc: CORE CHANNEL 1 (CR1-1A)						
GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
LOWER REACTIV.	BORON INTRUSION VIA THE REACTIVITY CONTROL SYSTEM	POWER OSCILLATING DOWNWARD				YOU CAN'T EXPECT ANY EXCITEMENT OUT OF LOWER K-EFFECTIVE
	BORON INTRUSION VIA THE DENSITY LOCKS	POWER OSCILLATING DOWNWARD				
	MODERATOR TEMP INCREASE	REDUCES POWER				
	DEPRESSURIZATION CAUSE VOIDING	OSCILLATION LEADING TO SHUTDOWN				
	REDUCED RATE FLOW TO REACTOR	CHALLENGE LOWER DENSITY LOCK				
	FAILURE OF REACTIVITY CONTROL SYSTEM (#312)	GRADUAL POWER REDUCTION				
	LOSS OF FEEDWATER FLOW TO SG	REDUCES POWER				
	RC PUMP SLOWS DOWN	CHALLENGE THE LOWER DENSITY LOCK				
	LOSS OF OFFSITE POWER	SCRAMS				
HIGHER REACTIV.	GEOMETRY CHANGE	LOWER POWER LEVEL				
	BORON DILUTION	INCREASED POWER, OSCILLATION				
	TEMPERATURE REDUCTION					
	GEOMETRY CHANGE					
	INCREASE IN FEEDWATER FLOW					
MORE BORON	DECREASE IN INLET FEEDWATER TEMPERATURE					
	INCREASE IN STEAM FLOW					
	REACTIVITY CONTROL SYSTEM	DECREASING POWER				
	POOL LOOP ACTIVATION	OSCILLATION AND DECREASING POWER				
LESS BORON	TEMPERATURE REDUCTION	MORE POWER				
	REACTIVITY CONTROL SYSTEM	MORE POWER				
	BURNUP	GRADUAL UPWARD DRIFT OF POWER				
	TEMPERATURE INCREASE	LESS POWER				

Project: PIUS SYSTEM STUDY			Section: 001		Page: 3	
Section Desc: CORE CHANNEL 1 (CR1-1A)						
GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
NO FLOW	MAJOR DISRUPTION OF GEOMETRY	MELTDOWN ..				
REVERSE FLOW						
MORE FLOW	GEOMETRY	DON'T KNOW		WS	FURTHER STUDY	
	VOIDING	DON'T KNOW		WS	FURTHER STUDY	
	INCREASE IN PUMP SPEED	HIGHER VELOCITY DISRUPT DENSITY LOCK				
LESS FLOW	GEOMETRY	CORE HEATUP ACTIVATE DENSITY LOCKS				
	DECREASE IN PUMP SPEED	SCRAM				
	LOOSE PARTS	CORE DAMAGE				
	PRIMARY BREAK	ACTIVATE DENSITY LOCKS				
OSCILLAT. FLOW	PUMP CONTROL SERVO SYSTEM	THERMAL CYCLING OF FLOW STRUCTURE (DENSITY/GAS LOCK) FLOW INDUCED VIBRATION		RM	REVIEW KEIJO'S REPORT	THIS IS POTENTIALLY ONE OF THOSE FURTHER ANALYSIS THINGS WE ARE CONSIDERING INCLUDING IN THE FINAL RECOMMENDATION
	LOOP ACTIVATION					
	VOIDING					
MORE LEVEL						
LESS LEVEL						
MORE PRESSURE	CHANGE IN PRESSURIZER SETPOINT DRIFT					
	BLOCKAGE DOWNSTREAM OF THIS NOOE	PREVIOUSLY CONSIDERED (BURP OUT THE LO''R DENSITY)				
	REDUCTION OF SECONDARY HEAT SINK	PREVIOUSLY CONSIDERED				
	BOILING IN THE CORE	POWER REDUCTION DUE TO VOIDING DENSITY LOCKS ACTIVATED				
LESS PRESSURE	CHANGE IN PRESSURIZER SETPOINT DRIFT	POWER REDUCTION BOILING				

Section Desc: CORE CHANNEL 1 (CR1-1A)

GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
LESS PRESSURE (con't)	BREAK	PREVIOUSLY DISCUSSED ..				
	INCREASE IN SECONDARY HEAT SINK	POWER INCREASE				
	UPSTREAM OF NODE BLOCKAGE	PREVIOUSLY DISCUSSED				
	POWER REDUCTION	NO NEW CONSEQUENCES				
	REDUCTION IN PUMP FLOW	PREVIOUSLY DISCUSSED				
OSCILLATING PRES.	RELIEF VALVE CYCLING	BOILING IN THE CORE SIGNIFICANT OSCILLATION IN POWER AND FLOW ACTIVATE DENSITY LOCKS		SC	RUN THIS WITH PIPA	
MORE TEMP.	LESS FLOW	PREVIOUSLY CONSIDERED				
	LOSS OF SECONDARY HEAT SINK	PREVIOUSLY CONSIDERED (PC)				
	MORE CORE POWER	PC				
LESS TEMP.	MORE FLOW	PC				
	INCREASE IN SECONDARY HEAT SINK	PC				
	LESS CORE POWER	PC				
	ACTIVATE LOWER DENSITY LOCK	PC				
	LOWER PRESSURE	PC				
	COOL WATER INJECTION REACTIVITY SYSTEM	PC				
OSCILLAT. TEMP.	FLOW OSCILLATION	THERMAL CYCLIC STRESSES/FATIGUE				
	POWER OSCILLATION	PC				
	PRESSURE OSCILLATION	PC				
MORE VISCOSITY						
LESS VISCOSITY						

Project: PIUS SYSTEM STUDY	Revision:	Section: 001 Page: 1								
	Date: 11/26/91	Time: 12:30:29								
	Session:	Team: (see below)								
Section Description: CORE CHANNEL 1 (CR1-1B) ..	Leader: RY	Recorder: CC								
	Organization: BNL									
	Location: BLDG 130									
Design Intention: OPERATING AT CRITICALITY, POWER LEVEL SHOULD BE NOMINAL FULL POWER, PRESSURE 9 MEGA PASCAL, 13,200 KG PER SEC, INLET TEMP 260, OUTLET TEMP 290, BORON CONCENTRATION 375 PPM, FUEL TEMP 469, 1000 MWt, NO NITROGEN, NO OXYGEN, NOMINAL VISCOSITY, POWER TILT - NONE, FLOW TILT - NONE, AMBIENT CONT.	P&ID's: Revision:									
Team Members: <table border="0" style="width: 100%;"> <tr> <td>R. YOUNGBLOOD</td> <td>S. CHENG</td> <td>B. CHAN</td> <td>R. FULLWOOD</td> </tr> <tr> <td>R. MORANTE</td> <td>B. LUCKAS</td> <td>P. SOO</td> <td>B. SHIER</td> </tr> </table>			R. YOUNGBLOOD	S. CHENG	B. CHAN	R. FULLWOOD	R. MORANTE	B. LUCKAS	P. SOO	B. SHIER
R. YOUNGBLOOD	S. CHENG	B. CHAN	R. FULLWOOD							
R. MORANTE	B. LUCKAS	P. SOO	B. SHIER							

Project: PIUS SYSTEM STUDY			Section: 001		Page: 2	
Section Desc: CORE CHANNEL 1 (CR1-1B)						
GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
CONTAMINATION	REFUELING OPERATIONS	STRESS CORROSION CRACKING CORROSION REACTIVITY CONTROL DISTURBANCE CONDUCTIVITY				
	FAILURE OF COOLANT CLEANUP SYSTEMS	INCREASED WORKER EXPOSURE				
	SECONDARY SIDE INLEAKAGE					
	MINOR FUEL FAILURES					
RELIEF	NOT CONSIDERED FOR THIS MODE					
INSTRUM/CONTROL	(NOT DOABLE)					
SAMPLING	(SEE CONTAMINATION)					
CORROS/EROS	(SEE CONTAMINATION)					
RADIATION	CLAD FAILURE	CONTAMINATING THE PRIMARY AND SECONDARY (PC)				
	FAILURE OF CHEMISTRY CONTROL	PC				
STRUCT. INTEGR.	FLOW INDUCED VIBRATION	PC				
	THERMAL CYCLING					
GEOMETRY	CORE MELTDOWN	PC				
	RADICAL POWER EXCURSION	PC				
	STRUCTURAL COLLAPSE	PC				
CIRCULATION	N/A					
STRATIFICATION	N/A					
FLOW TILT	POWER TILT	NON-UNIFORM TEMP. CHANNEL OVERHEATING DAMPED OSCILLATION				
	ASYMMETRICAL BLOCKAGE	NON-UNIFORM TEMP CHANNEL OVERHEATING POSSIBILITY OF CHALLENGING THE DENSITY LOCK				
	DISRUPTION IN THE ORIFICING					

Project: PIUS SYSTEM STUDY					Section: 001	Page: 3
Section Desc: CORE CHANNEL 1 (CR1-1B)						
GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
POWER TILT	FLOW TILT	PC				
	XENON OSCILLATION	PC				
	NON-UNIFORM MIXING OF INJECTED BORON	CHANNEL OVERHEATING				
	ASYMMETRIC LOADING OF THE GADOLINIUM	CHANNEL OVERHEATING				
SERVICE FAIL.	N/A					
ABNORM. OP.	N/A					
MAINTENANCE	N/A					
IGNITION	N/A					
SPARE EQUIP.	N/A					
EXTERNAL EVENTS	EARTHQUAKE	PC (BLOCKAGE)				
SAFETY	N/A					

Project: STARTUP OF PIUS - 1	Revision:	Section: 001	Page: 1
	Date: 03/30/92	Time: 14:24:28	
	Session:	Team: (see below)	
Section Description: PRESSURIZE AND PUT REACTOR PRESSURE UNDER SYSTEM CONTROL (file: startup)	Leader: RY	Recorder: CSC	
	Organization: BNL		
	Location: BLDG 130		
Design Intention: (1) TURN ON STEAM INJECTION, AND UNDER MANUAL OPERATION, PRESSURIZE TO FINITE AMOUNT ABOVE LOWER LIMIT OF CONTROL SYSTEM; (2) PLACE PRESSURE CONTROL IN AUTOMATIC; ASSUMING POOL COOLING IS WORKING PROPERLY AND EVERYTHING ELSE IS NORMAL	P&ID's: Revision:		
Team Members: R YOUNGBLOOD B SHIER P SOO P KROEGER R MORANTE			

Project: STARTUP OF PIUS - 1				Section: 001		Page: 2
Section Desc: PRESSURIZE AND PUT REACTOR PRESSURE UNDER SYSTEM CONTROL (file: startup)						
GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
MORE ACTION	PRESSURIZE TOO HIGH, MAYBE TO RELIEF VALVE PRESSURE SETPOINT	LOW TEMP OVERPRESSURIZATION?				
LESS ACTION	PRESSURIZE BELOW DESIRED, BUT ENOUGH TO ENABLE SYSTEM CONTROL TO TAKE OVER	NOTHING SIGNIFICANT				
	FAILURE TO ACHIEVE MINIMUM CONTROLLABLE PRESSURE	NOTHING SIGNIFICANT				
WRONG ACTION	NONE IDENTIFIED			PK	LOOK FOR WRONG MODE SETTINGS OF CONTROLLER	
NO ACTION	PRESSURIZES BUT DOES NOT PUT SYSTEM UNDER AUTOMATIC CONTROL	STARTUP WITH INSUFFICIENT PRESSURE WILL LEAD TO BOILING IN THE CORE LATER?				
INDICATIONS	WRONG PRESSURE INDICATION (HIGH OR LOW) MAY LEAD TO INSTANCES OF INCORRECT ACTION	PC				
STEP TOO SOON	GO UNDER AUTO CONTROL TOO SOON	PC				
STEP TOO LATE	GO UNDER AUTO CONTROL TOO LATE	OVERPRESSURIZE, MAY GO THROUGH SUBSEQUENT STEPS AT EXCESSIVE PRESSURE				
	GO UNDER AUTO CONTROL TOO LATE	MAY BE POSSIBLE TO TAKE SUBSEQUENT STEPS AT LOWER PRESSURE, LEADING TO BOILING IN THE CORE LATER?				
MORE TIME	NA OR PC					
LESS TIME	NA					
WRONG TIME	START MAIN COOLANT PUMPS BEFORE DESIRED PRESSURE REACHED	NPSH NOT MET?				
NO TIME	PC					
TOO FAST	NA					
TOO SLOW	PC					

Project: STARTUP OF PIUS - 2	Revision:	Section: 001	Page: 1
	Date: 03/30/92	Time: 15:39:00	
	Session:	Team: (see below)	
Section Description: ADD HYDRAZINE (file: hydraz)	Leader: RY	Recorder: CSC	
	Organization: BNL		
	Location: BLDG 130		
Design Intention: TO ACHIEVE ADEQUATE CONCENTRATION OF HYDRAZINE IN ORDER TO SCAVENGE OXYGEN	P&ID's: Revision:		
Team Members: R YOUNGBLOOD B SHIER P KROEGER R MORANTE			

Section Desc: ADD HYDRAZINE (file: hydraz)

GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
MORE ACTION	ADDITION OF EXCESS HYDRAZINE	???		PSOO	MATERIALS PROBLEM?	
LESS ACTION	ADDITION OF TOO LITTLE HYDRAZINE	TOO LITTLE OXYGEN SCAVENGING WITH ASSOCIATED LONG TERM PROBLEMS				
WRONG ACTION						
NO ACTION	PC					
INDICATIONS	NS					
STEP TOO SOON	ADD HYDRAZINE TOO SOON	NO SIGNIFICANT CONSEQUENCE				
STEP TOO LATE	ADD HYDRAZINE TOO LATE	PC				
MORE TIME	NA					
LESS TIME	NA					
WRONG TIME	PC					
NO TIME	PC					
TOO FAST	NA					
TOO SLOW	NA					

Project: STARTUP OF PIUS - 3	Revision:	Section: 001	Page: 1
	Date: 03/31/92	Time: 09:31:23	
	Session:	Team: (see below)	
Section Description: START MAIN COOLANT PUMPS AT 20% PUMP SPEED .. (file: startmcp)	Leader: RY	Recorder: CSC	
	Organization: BNL		
	Location: BLDG 130		
Design Intention: GET ALL FOUR PUMPS RUNNING BEFORE THE NEXT STEP IS TAKEN, MANUAL 20% OF FULL SPEED	P&ID's: Revision:		
Team Members: R YOUNGBLOOD P KROEGER B CHAN R MORANTE B SHIER			

Project: STARTUP OF PIUS - 3				Section: 001		Page: 2
Section Desc: START MAIN COOLANT PUMPS AT 20% PUMP SPEED (file: startmcp)						
GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
MORE ACTION	N/A	..				
LESS ACTION	TURN ON LESS THAN 4 PUMPS	REDUCED CORE FLOW REVERSE FLOW IN IDLE LOOP(S) IF PUMP IS TURNED ON AFTER THE GAS LOCK IS ESTABLISHED, GAS LOCK WILL BE LOST				
WRONG ACTION	BRING UP PUMPS TO WRONG MODE	TOO MUCH FLOW, DIFFICULTY ESTABLISHING GAS LOCK PUMP MAY RUN FLAT OUT IF IT TRIES TO CONTROL THE INTERFACE LEVEL (THERE IS NO INTERFACE) (MAY LEAD TO PUMP TRIP, P. 534-4)				
NO ACTION	STARTING NO COOLANT PUMPS	DELAYED STARTUP, TREATED BELOW				
INDICATIONS	NEED RELIABLE FLOW INDICATION	DEBORATING WITHOUT PROPER FLOW WOULD BE A CONCERN				
STEP TOO SOON	PC					
STEP TOO LATE	POTENTIAL TO START DEBORATING BEFORE PUMPS ARE STARTED				ESTABLISH WHAT BORON CONCENTRATION TAKES THE CORE CRITICAL AT 50C, 100C; DITTO FOR PROMPT CRITICAL; HOW DOES LETDOWN WORK DURING STARTUP, WHAT IS NATURAL CIRCULATION RATE THROUGH THE STEAM GENERATORS IN COLD SHUTDOWN,	
	POTENTIAL TO ESTABLISH GAS LOCK BEFORE PUMPS ARE STARTED	WHEN PUMP STARTS, NITROGEN IS LOST AND THEY HAVE TO RE-ESTABLISH THE GAS LOCK THERE IS A SYNERGISM BETWEEN THE TWO CAUSES OF "STEP TOO LATE" AND BOTH CAUSES CAN BE CAUSED BY THE SAME ERROR				
MORE TIME	NA					
LESS TIME	NA					

Project: STARTUP OF PIUS - 3				Section: 001		Page: 3
Section Desc: START MAIN COOLANT PUMPS AT 20% PUMP SPEED (file: startmcp)						
GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
WRONG TIME	SEE "STEP TOO LATE"	..				
NO TIME	MAIN COOLANT PUMP IS NEVER TURNED ON	IT MAY BE A THEORETICAL POSSIBILITY TO START REACTOR ON NATURAL CIRCULATION; THERMAL PROBLEMS COULD BE ANTICIPATED AS POWER IS INCREASED				
TOO FAST	TURNING ON ALL PUMPS AT ONCE	POSSIBLE ELECTRICAL OVERLOAD				
TOO SLOW	NO PROBLEM					

Project: STARTUP OF PIUS - 4	Revision:	Section: 001	Page: 1
	Date: 03/31/92	Time: 15:10:37	
	Session:	Team: (see below)	
Section Description: ESTABLISH GAS LOCK IN LOWER DENSITY LOCK (file: gaslock)	Leader: RY	Recorder: CSC	
	Organization: BNL		
	Location: BLDG 130		
Design Intention: FEED NITROGEN THROUGH LOWER DENSITY LOCK UNTIL LOCK IS ESTABLISHED AND THEN REDUCE NITROGEN FLOW TO LEAKAGE RATE	P&ID's:	Revision:	
Team Members: <div style="display: flex; justify-content: space-around; margin-top: 10px;"> <div>R YOUNGBLOOD B SHIER</div> <div>P KROEGER B CHAN</div> <div>R MORANTE</div> </div>			

Project: STARTUP OF PIUS - 4				Section: 001		Page: 2
Section Desc: ESTABLISH GAS LOCK IN LOWER DENSITY LOCK (file: gaslock)						
GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
MORE ACTION	EXCESSIVE NITROGEN FEED	A LOT OF NITROGEN IN THE PRESSURIZER STEAM SPACE				THESE COMMENTS WERE PREDICATED ON THE DESCRIPTION IN SECTION 34.4; SECTION 322.5 CAN BE READ TO SUGGEST THAT THEY ARE NEVER CRITICAL WITH THE GAS LOCK PRESENT
	OVERBLEED CAP AFTER ESTABLISHING IT	SAME EFFECT AS TOO LITTLE NITROGEN				
LESS ACTION	TOO LITTLE NITROGEN TO ESTABLISH GAS LOCK	MORE LIKELY TO GET AN INADVERTENT SHUTDOWN, INADVERTENT ADDITION OF POOL WATER				
	TOO LITTLE BLEED	TOO MUCH NITROGEN				
WRONG ACTION	NA					
NO ACTION	FAIL TO ESTABLISH GAS LOCK	MAY KEEP THEM FROM STARTING UP				
INDICATIONS	WE DON'T KNOW THE INDICATIONS					
STEP TOO SOON	ESTABLISH GAS LOCK BEFORE STARTING PUMPS	PC				
STEP TOO LATE	FAILING TO ESTABLISH GAS LOCK BEFORE DEBORATING	POSTPONE APPROACH TO CRITICALITY				
MORE TIME	PC					
LESS TIME	PC					
WRONG TIME	PC					
NO TIME	PC					
TOO FAST	PC					
TOO SLOW	PC					

Project: STARTUP OF PIUS - 5	Revision:	Section: 001	Page: 1
	Date: 04/02/92	Time: 13:39:16	
	Session:	Team: (see below)	
Section Description: START TO DEBORATE @ 60 KG/SEC (file: deborate)	Leader: RY	Recorder: CSC	
	Organization: BNL		
	Location: BLDG 130		
	P&ID's: Revision:		
Design Intention: INJECT DISTILLED WATER AT 60 KG/SEC THROUGH TWO INJECTION LINES, DEBORATE IN ORDER TO APPROACH CRITICALITY; ESTABLISH ENOUGH LETDOWN CAPACITY TO ACCOMMODATE THIS THROUGH THE NORMAL WITHDRAWAL SYSTEM			
Team Members: R YOUNGBLOOD B SHIER R MORANTE P SOO			

Project: STARTUP OF PIUS - 5			Section: 001		Page: 2	
Section Desc: START TO DEBORATE @ 60 KG/SEC (file: deborate)						
GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
MORE ACTION	ESTABLISH TOO MUCH LETDOWN (FOR EXAMPLE THROUGH THE TEMPORARY DISCHARGE SYSTEM)	EVENTUALLY THE POOL WOULD COME IN THROUGH THE UPPER DENSITY LOCK AND DELAY CRITICALITY				
LESS ACTION	START TO DEBORATE AT LESS THAN 60 KG/SEC	TAKES TOO LONG TO GO CRITICAL TAKES TOO LONG TO HEAT UP				
	FEED ONE INJECTION LINE, INSTEAD OF TWO INJECTION LINES	PRESENT CONFUSING BORON CONCENTRATION SIGNALS TO MONITORING? IF ALL PUMPS ARE RUNNING, CONSEQUENCES WOULD APPEAR TO BE MINIMAL				
	FAILURE TO ESTABLISH SUFFICIENT LETDOWN	DIFFERENTIAL EXCESS WILL GO TO THE POOL THROUGH THE UPPER DENSITY LOCK				
WRONG ACTION	PUT IN BORATED WATER INSTEAD OF DISTILLED WATER	DELAY CRITICALITY, NUISANCE	RECOMMEND HAZOPING REACTIVITY CONTROL SYSTEM			
	PUT IN FILTERED WATER INSTEAD OF DISTILLED WATER	PC				
NO ACTION	FAILURE TO INITIATE DEBORATION	PC				
	FAILURE TO ESTABLISH LETDOWN	PC				
INDICATIONS	NOTHING SIGNIFICANT					
STEP TOO SOON	START DEBORATING BEFORE STARTING MAIN COOLANT PUMPS	PC				
	START DEBORATING BEFORE ESTABLISHING GAS LOCK	PC				
STEP TOO LATE	INITIATING DEBORATION AFTER PLACING REACTIVITY CONTROL AND POWER CONTROL SYSTEMS IN SERVICE	MAY NOT BE REALISTIC BUT CONSEQUENCES WOULD BE WORTH EXAMINING FURTHER IF POSSIBLE				
MORE TIME	DEBORATES TOO LONG	GIVEN A NORMAL FUEL LOADING, THIS WOULD HAVE TO GO ON FOR A LONG TIME IN ORDER TO HAVE AN EFFECT				CONSEQUENCES OF THIS DEVIATION MAY DEPEND STRONGLY ON THE POSSIBILITY OF ANOMALOUS FUEL LOADING

Section Desc: START TO DEBORATE @ 60 KG/SEC (file: deborate)

GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
MORE TIME (con't)	DEBORATES TOO LONG	IT WOULD APPEAR THAT AN UNCHECKED DEBORATION WITH NO CONTROL SYSTEMS ACTIVE MIGHT LEAD TO SHUTDOWN THROUGH THE DENSITY LOCKS (ASSUMING THAT THE GAS LOCK DOES NOT PREVENT THIS)				
	LETDOWN GOES ON TOO LONG	FLUSH THE UPPER DENSITY LOCK AND THE LOWER DENSITY LOCK				
LESS TIME	DEBORATE FOR LESS TIME	DELAY APPROACH TO CRITICALITY				
	LETDOWN FOR LESS TIME	MINIMAL CONSEQUENCES				
WRONG TIME	AUTOMATIC REACTIVITY CONTROL PLACED IN SERVICE BEFORE DEBORATION IS COMPLETE	DELAYS APPROACH TO CRITICALITY				
NO TIME	DEBORATION DOES NOT OCCUR	DELAYS APPROACH TO CRITICALITY				
TOO FAST	NA					
TOO SLOW	NA					

Project: STARTUP OF PIUS - 6	Revision:	Section: 001 Page: 1
Section Description: START CONDENSATE PUMP IN PREPARATION FOR STARTUP OF SECONDARY SYSTEM NO REAL EFFECT ON PRIMARY, DETAILED HAZOP NOT PERFORMED	Date: 04/06/92	Time: 08:24:36
Design Intention:	Session:	Team: (see below)
	Leader: RY	Recorder: CSC
	Organization: BNL	
	Location: BLDG 130	
	P&ID's:	Revision:
Team Members: <div style="display: flex; justify-content: space-between;"> R YOUNGBLOOD B SHIER </div> <div style="display: flex; justify-content: space-between;"> R MORANTE B CHAN </div>		

Project: STARTUP OF PIUS - 7	Revision:	Section: 001	Page: 1
	Date: 04/06/92	Time: 10:16:35	
	Session:	Team: (see below)	
Section Description: WHEN THE REACTOR GOES CRITICAL, CHANGE THE DEBORATION RATE FROM MAXIMUM (60 KG/SEC) TO A LEVEL CONTROLLED BY THE POWER CONTROL SYSTEM; THE POWER LEVEL IS SET MANUALLY (file: critical)	Leader: RY	Recorder: CSC	
	Organization: BNL		
	Location: BLDG 130		
Design Intention: PUT DEBORATION UNDER CONTROL OF AN AUTOMATIC SYSTEM RATHER THAN BEING MAXIMUM; POWER LEVEL SHOULD BE SET AT A LOW LEVEL, BUT SUFFICIENT TO HEATUP TO 100C	P&ID's: Revision:		
Team Members: <div style="display: flex; justify-content: space-between;"> R YOUNGBLOOD B SHIER </div> <div style="display: flex; justify-content: space-between;"> R MORANTE B CHAN </div>			

Section Desc: WHEN THE REACTOR GOES CRITICAL, CHANGE THE DEBORATION RATE FROM MAXIMUM (60 KG/SEC) TO A LEVEL CONTROLLED BY THE POWER CONTROL SYSTEM; THE POWER LEVEL IS SET MANUALLY

GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
MORE ACTION	N/A					
LESS ACTION	N/A					
WRONG ACTION	PUT IN MODE A, NORMAL OPERATION	IF THE SYSTEM TRIES TO CONTROL OUTLET TEMP, SYSTEM WILL TRY TO DEBORATE AT A HIGH RATE				
	PUT IN MODE B	DITTO				
	PUT IN MODE C	DITTO				
	PUT IN MODE E	DITTO				
	PUT IN MODE F	SHUTDOWN BY BORATING FROM THE BORATED WATER SYSTEM (SUBSYSTEM 3123)				
	SETTING TOO HIGH A POWER LEVEL	SINCE PUMPS ARE STILL AT 20% SPEED, ACHIEVING TOO HIGH A POWER LEVEL WOULD LEAD TO CORE HEATUP; THIS COULD CHALLENGE THE LOWER DENSITY LOCK WITH THE GAS LOCK IN PLACE				ASSUMPTION THAT MANUAL OPERATING MODE PERMITS SETTING TOO HIGH A POWER LEVEL
	SETTING TOO LOW A POWER LEVEL	SLOWDOWN TO STARTUP PROCEDURE				
	DIAL IN TOO MUCH LETDOWN	WATER INGRESS THROUGH UPPER DENSITY LOCK AND/OR LOWER DENSITY LOCK FILTERED WATER FLOW WOULD COMPENSATE FOR THIS				
	DIAL IN TOO LITTLE LETDOWN	FILTERED WATER FLOW WOULD COMPENSATE FOR THIS				
NO ACTION	FAILURE TO PUT REACTIVITY CONTROL SYSTEM UNDER THE CONTROL OF THE REACTOR POWER CONTROL SYSTEM	LEAVE SYSTEM AT MAXIMUM DISTILLED WATER RATE; PC				
	FAILURE TO SET POWER LEVEL	PC				
INDICATIONS	INDICATIONS OF CRITICALITY AND BORON CONCENTRATION NEED TO BE VERY EXPLICIT TO RULE OUT LOCAL FUEL ANOMALIES					
STEP TOO SOON	PC					

Section Desc: WHEN THE REACTOR GOES CRITICAL, CHANGE THE DEBORATION RATE FROM MAXIMUM (60 KG/SEC) TO A LEVEL CONTROLLED BY THE POWER CONTROL SYSTEM; THE POWER LEVEL IS SET MANUALLY

GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
STEP TOO LATE	FAILURE TO PLACE REACTIVITY CONTROL UNDER POWER CONTROL BEFORE GAS LOCK DISESTABLISHED	LIKELY THAT DENSITY LOCKS WILL FLUSH				
MORE TIME	NA					
LESS TIME	NA					
WRONG TIME	NA					
NO TIME	NA					
TOO FAST	NA					
TOO SLOW	NA					

Project: STARTUP OF PIUS - 8	Revision:	Section: 001 Page: 1
	Date: 04/06/92	Time: 13:25:17
	Session:	Team: (see below)
Section Description: DEACTIVATE GAS LOCK BY STOPPING NITROGEN FLOW IN AND BLEEDING NITROGEN OUT OF GAS LOCK (file: deactiv)	Leader: RY	Recorder: CSC
	Organization: BNL	
	Location: BLDG 130	
	P&ID's: Revision:	
Design Intention: RESTORE NORMAL LOWER DENSITY LOCK FUNCTION INCLUDING BYPASS FLOW THROUGH LOWER DENSITY LOCK PLENUM SUCTION PIPE		
Team Members: <div style="display: flex; justify-content: space-between;"> R YOUNGBLOOD B SHIER </div> <div style="display: flex; justify-content: space-between;"> R MORANTE P KROEGER </div>		

Section Desc: DEACTIVATE GAS LOCK BY STOPPING NITROGEN FLOW IN AND BLEEDING NITROGEN OUT OF GAS LOCK (file: deactiv)

GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
MORE ACTION	N/A					
LESS ACTION	N/A					
WRONG ACTION	INCREASE NITROGEN SUPPLY INSTEAD OF DECREASING NITROGEN SUPPLY	GAS LOCK WILL REMAIN FUNCTIONAL POSSIBILITY OF LOCAL VOIDING IN SELECTIVE CHANNELS LEADING TO LOCAL FUEL DAMAGE				
	OPEN ALL THE WAY RATHER THAN CLOSE THE BLEEDOFF LINE	APPEARS TO CREATE THE POSSIBILITY OF A SMALL PRIMARY COOLANT LOSS WHICH IS COMPENSATED FOR EITHER BY INFUX OF POOL WATER OR MAKEUP				
NO ACTION	GAS LOCK REMAINS	PC				
INDICATIONS	FAULTY PRIMARY SYSTEM TEMPERATURE INDICATION COULD CAUSE PREMATURE OR DELAYED DEACTIVATION OF THE GAS LOCK	EARLY REMOVAL OF GAS LOCK COULD CAUSE INGRESS OF BORATED POOL WATER LATE REMOVAL OF GAS LOCK - PC				
STEP TOO SOON	PC (SEE STEP TOO LATE, STEP 7)					
STEP TOO LATE	DEACTIVATION OF GAS LOCK COULD BE DELAYED UNTIL PRIMARY SYSTEM IS HEATED UP TO 260C	ANY POTENTIAL UPSET CONDITIONS WOULD FLUSH OUT GAS LOCK				
MORE TIME	N/A					
LESS TIME	N/A					
WRONG TIME	PC					
NO TIME	PC					
TOO FAST, TOO SLOW	DEACTIVATION RATE CAN BE CONTROLLED BY MOTOR OPERATED VALVES	NO IDENTIFIED CONSEQUENCES				

Project: STARTUP OF PIUS - 9	Revision:	Section: 001	Page: 1
	Date: 04/06/92	Time: 15:31:10	
	Session:	Team: (see below)	
Section Description: ACCOMPLISH PRIMARY SYSTEM HEATUP TO 260C AND HOT STANDBY CONDITIONS, SIMULTANEOUS HEATUP AND PRESSURIZATION OF SECONDARY SYSTEM; INITIATION OF FEEDWATER FLOW (file: hotstand)	Leader: BS	Recorder: CSC	
	Organization: BNL		
	Location: BLDG 130		
	P&ID's: Revision:		
Design Intention: BRING REACTOR TO HOT STANDBY CONDITIONS IN PREPARATION FOR POWER GENERATION			
Team Members: <div style="display: flex; justify-content: space-between;"> <div> B SHIER R MORANTE </div> <div> P KROEGER R YOUNGBLOOD </div> </div>			

Section Desc: ACCOMPLISH PRIMARY SYSTEM HEATUP TO 260C AND HOT STANDBY CONDITIONS, SIMULTANEOUS HEATUP AND PRESSURIZATION OF SECONDARY SYSTEM; INITIATION OF FEEDWATER FLOW (file: hotstand)

GUIDEWORDS	CAUSE	CONSEQUENCE	RECOMMEND.	INDV	ACTION	COMMENTS
MORE ACTION	NONE IDENTIFIED					
LESS ACTION	NONE IDENTIFIED					
WRONG ACTION	INCORRECT OUTLET PLENUM TEMP SETPOINT (GREATER THAN 260C)	CONSEQUENCE OF THIS DEVIATION IS THAT LATER, AS POWER IS INCREASED, YOU COULD GO TO SATURATION CONDITIONS				ASSUMES THAT OPERATOR HAS FREEDOM IN THE CHOICE OF SELECTED OUTLET TEMPERATURE SETPOINT VALUES
NO ACTION	NO SIGNIFICANT ACTIONS IDENTIFIED					
INDICATIONS						
STEP TOO SOON						
STEP TOO LATE						
MORE TIME						
LESS TIME	HEATUP ACHIEVED IN TOO SHORT A TIME	EXCEED 40C/HR HEATUP RATE; HIGHER THERMAL STRESSES				

APPENDIX C

SYSTEM INTER-RELATIONSHIPS AND SAFETY CLASSIFICATIONS

APPENDIX C - SYSTEM INTER-RELATIONSHIPS AND SAFETY CLASSIFICATIONS

Appendix C.1 Safety Classification of Systems

Section 8 discussed the PIUS system dependencies and the safety classification of systems. This Appendix presents some of the same information but also goes into more detail than Section 8. This Appendix incorporates comments, as appropriate, from the May 11 ABB Atom letter.

Section 3.2 of the PSID discusses the "Classification of structures, components, and systems." The PSID states that the safety classification for PIUS is based on the same principles as for present day LWRs. Mention is made of ANSI/ANS 51.1 (1983), "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," however no commitment to following this standard is given. Most (but not all) of the various classification schemes used for PIUS, are defined in the PSID, are similar to those in ANSI/ANS 51.1, and will be summarized below. In section 3.2 of the PSID, Table 3.A (which is similar to Table C-1 below) correlates some of the classifications used. The notes to Table 3.A use the undefined term safety-related. Additionally, in the PSID "Systems Chapters" (Chapters 100 - 800 of Volume III), a slightly different terminology is used. In a number of these system and subsystem chapters, the PSID designates that system as "safety grade" or "not safety grade." Other of these chapters are silent relative to the designation as safety grade. Further, the PSID does not appear to provide a definition of safety grade, as used in these chapters, or a correlation between safety grade and the designations in Table C-1 below.

Table C-1 Summary of PIUS Safety Classifications			
PIUS Safety Class (SC-)	Seismic Class	Mechanical Class	Electrical Class
1	I	A	IE
2	I	B	IE
3	I	C	IE
N	N	D	C

The four PIUS classifications of this Table are discussed in more detail below, using quotes from the PSID:

Class 1 "shall apply to pressure-retaining portions and supports of mechanical equipment that form part of the RCPB, whose failure could cause a loss of reactor coolant in excess of the reactor coolant normal makeup capability and whose requirements are within the scope of the ASME Boiler and Pressure Vessel Code, Section III." This means the PCR/V up to and including isolation valves. However consideration must be given to PIUS where the upper and lower parts have very different safety implications. The lower PCR/V (e.g. PCR/V monolith) is assigned to a higher safety class (1) than the upper parts (2).

Class 2 "shall apply to pressure-retaining portions and supports of primary containment and other mechanical equipment, requirements for which are within the scope of the ASME Boiler and Pressure Vessel Code, Section III, that is not included in SC-1, and is designed and relied upon to accomplish the following nuclear safety functions:"

- Fission product barrier, primary containment, radioactive material holdup or isolation,
- Emergency heat removal of the primary atmosphere to an intermediate heat sink, or emergency removal of radioactive material from the primary containment atmosphere,
- Introduce negative reactivity into the reactor, or restrict the addition of positive reactivity,
- Emergency core cooling,
- Maintain coolant inventory for emergency core cooling.

PIUS does not have many of the systems mentioned in the code. The PIUS interpretation is:

- The upper parts of the PCRV including containment isolation valves,
- Reactor internals that are important for maintaining core geometry and coolant flow,
- Safety valve function,
- Reactor shutdown via coolant pump trip,
- Containment.

Safety Class 3 "shall apply to equipment, not included in SC-1 or SC-2, that is designed and relied upon to accomplish the following nuclear safety functions:"

- a) Provide for functions of SC-2 equipments or portions are not within the scope of ASME Section III,
- b) Provide secondary containment radioactive material holdup, isolation or heat removal,
- c) Ensure hydrogen concentration control,
- d) Removal of radioactive material from spaces outside of the containment,
- e) Introduce negative reactivity to achieve or maintain subcriticality,
- f) Provide or maintain reactor coolant inventory,
- g) Maintain geometry within the reactor to ensure reactivity control and core cooling,
- h) Structurally load-bear or protect SC-1, -2 or -3 equipment,
- i) Radiation shielding of the control room and offsite personnel.
- j) Cooling of liquid-stored fuel,
- k) Functionability of SC-1, -2, -3 equipment,
- l) Power for SC-1, -2, -3 equipment,
- m) Signals or power for SC-1, -2, -3 equipment,
- o) Manual or automatic interlock for functionability of SC-1, -2, -3 equipment,
- p) Acceptable environment for SC-1, -2, -3 equipment.

The two seismic classes of Regulatory Guide 1.29, I (fully seismic) and N (not seismic), are supplemented with a Seismic Class P (passive). This means that the passive function and structural integrity of the equipment must be maintained, but there is no requirement for active function i.e. a pressure boundary must be maintained but pumps need not work. It should be noted that PIUS considers the combined occurrence of SSE and LOCA.

Mechanical classifications correspond to the mechanical quality classes, and electrical equipment classification correspond to IEEE 308 (1974). However, PIUS uses the terminology IE rather than 1E for safety related electrical items. An electrical classification of "C" indicates conventional electrical equipment and a "-" indicates that there is no electrical function for that equipment.

The simplified Table (C-1) shows the relationships between these four classification systems. Table C-2, which was adapted from Table 3.A of the PSID, provides a detailed breakdown of PIUS subsystems, showing their various safety classifications. Non-safety equipment or subsystems (that is, equipment having a safety designation N, seismic classification N, mechanical quality D, and electrical classification C) are not included in this table.

Table C-2 Safety Classification of PIUS Structures, Components and Systems					
System No.	Description	Safety	Seismic	Mechanical	Electrical
100	Buildings, External Plant Systems				
121	Reactor (Rx) building (bldg) containing safety-related parts	3	I	C	-
125	Control systems bldg, main control room	3	I	C	-
151	Concrete vessel monolith	1	I	A	-
152	Concrete containment	2	I	B	-
157	Concrete vessel liner cooling system	N	P	D	-
159	Concrete vessel liner venting system	3	I	C	-
200	Rx & Rx Auxiliary Systems				
211	Pressure vessel upper parts ¹	2	I	B	IE
213	Reactor internals - for shutdown and core cooling (sd&cc)	2	I	B	-
215	Reactor assembly supports and restraints for sd&cc	2	I	B	-
216	Vessel internal instrumentation support equipment for sd&cc	2	I	B	-
231	Refueling parts to prevent dropping of fuel elements	3	P	C	-
233	Refueling handling parts to prevent equipment dropping on reactor	3	P	C	-
241	Fresh fuel storage equipment	N	P	D	-
243	Spent fuel storage equipment	N	P	D	-
261	Fuel assemblies	2	I	B	-
300	Reactor Plant Process Systems				
312	Reactivity control system integrity ¹	2	I	B	-
313	Recirculation system integrity	2	I	B	-
314	Pressure relief system (PRS) integrity	2	I	B	-
314	PRS safety valves	2	I	B	-
314	PRS other parts	3	P	B	C
316	Condensation system	2	P	B	IE
321	Reactor pool	2 (only waterfill of 151 cavity)			
322	Pool loop inactivation system integrity ¹	2	I	B	-

Table C-2 Safety Classification of PIUS Structures, Components and Systems					
System No.	Description	Safety	Seismic	Mechanical	Electrical
323	Reactor pool cooling integrity ¹	2	I	B	-
325	Reactor pool water replacement system ¹	2	I	B	-
328	Pressurizer system integrity ¹	2	I	B	-
331	Reactor water cleanup system ²	3	P	C	C
332	Plant shutdown reactor pool cleanup integrity ¹	2	I	B	-
341	Offgas delay system ²	3	P	C	C
342	Liquid waste system ²	3	P	C	C
343	Solid waste system ²	3	P	C	C
345	Monitored floor drains, parts for safety-related room	3	I	C	IE
349	Offgas filter systems (if excess radioactivity)	3	P	C	C
400	Steam & Power Conversion Systems				
500	Control Equipment				
511	Main control room	N	P	D	C
515	Control cables ³	3	I	C	IE
516	Trip and interlock system	3	I	-	IE
518	Emergency control room	3	I	C	IE
520	Computer Systems ³	3	I	C	IE
531	Neutron flux monitoring systems ³	3	I	C	IE
533	Water chemistry control system, boron monitoring ³	3	I	C	IE
536	Reactor, pool, concrete vessel monitoring ³	3	I	C	IE
554	Steam generator tube rupture mon. sys.	3	I	C	IE
555	Room radiation monitoring systems ³	3	I	C	IE
583	Containment monitoring systems ³	3	I	C	IE
600	Electrical Equipment				
649	MCP power supply system ³	3	I	C	IE
673	Safety grade 110 VDC aux. power supply	3	I	C	IE
680	Electrical power system, control system ³	3	I	C	IE
690	Power cable systems ³	3	I	C	IE

Table C-2 Safety Classification of PIUS Structures, Components and Systems					
System No.	Description	Safety	Seismic	Mechanical	Electrical
700	Service Systems				
714	Cooling towers for reactor pool coolers	3	I	C	IE
721	Diesel-backed closed cooling system for reactor pool ⁴	3	I	C	IE
722	Diesel-backed closed cooling system for component cooling ⁴	3	I	C	IE
724	Closed cooling system for reactor pool coolers	3	I	C	IE
727	Local air coolers ³	3	I	C	IE
733	Demineralized water distribution system ⁴	3	I	C	IE
734	High pressure distilled water system ⁴	3	I	C	IE
741	Containment ventilation system ⁴	3	I	C	IE
742	Reactor building ventilation dampers	3	I	C	IE
746	Control building ventilation system	3	N	C	IE
749	Emergency ventilation system	3	N	C	IE
751	Bottled air system	3	I	C	IE
753	Compressed air system ⁴	3	I	C	IE
800	Other Plant Equipment				
810	Transport equipment, reactor service room overhead crane	N	P	C	C

Notes to Table:

1. RCPB portion of system up to and including containment isolation valves (CIVs).
2. These systems may contain significant amounts of radioactive material. If the consequences of a rupture may be close to regulatory limits, then classify as shown, otherwise system is non-safety related (N/N/D/C).
3. Safety-related parts of these systems only.
4. These systems may have lines penetrating the containment. Only the CIVs and the lines between the CIV and containment wall are classified as shown.

C.2 System Dependencies and Safety Classifications

Table C-3, below, presents PIUS systems for which dependencies have been identified. This Table allows one to see, in a fairly compact form, the relationships between the various PIUS systems. One comment worthy of noting is that PIUS tends to divide systems into a more detailed subsystem structure than current US NPPs. Table C-4, which follows Table C-3, lists the systems for which no dependencies were identified. The system safety classifications used in these two Tables are: ND - not designated, NS - non-safety, PS - passive safety, and SG - safety grade. These classifications are taken from the PSID "Systems Chapters" (Chapters 100 - 800 of Volume III). In a number of these system and subsystem chapters, the PSID designates that system as "safety grade" or "not safety grade." Other of these chapters are silent relative to the designation as safety grade. Further, the PSID does not appear to provide a definition of safety grade, as used in these chapters, or a correlation between safety grade and the safety class designations used in Table C-1 and C-2 above. Another difficulty is that Table 3.A of the PSID has a number of instances where portions of a system are designated as safety class 1, 2, or 3, but other portions are non-safety. The systems chapters of the PSID generally do not make such distinctions and just classify a system as safety grade, not safety grade, or remain silent on the issue.

All of the safety classifications are located to the left of the system identifications. Table C-3 has three other columns, for three types of systems: the parent system, systems that support the parent, and systems that are supported by the parent. These are designated as: System, Supporting Systems, and Fed Systems, respectively. These systems are identified by the column in which they are located. The table also contains some descriptive matter. If descriptions are too long for one line, they are continued, indented from the column that designates the type of system.

Table C-3 illustrates quite a non-uniform combination of safety and non-safety grade systems and subsystems associated with each other. It is believed that a Table such as this will be useful for evaluating the appropriateness of safety classification of support systems. However, before one can make any reasonable conclusions about this, the designation and documentation will need to be improved.

Table C-3			
Systems with Identified Dependencies			
Safety Class	Fed System	System	Supporting Systems
ND		System 121 Reactor Building	
NS			345 Monitored Floor Drain Systems
ND			727 Local Air Cooler Systems
PS			742 Reactor Building Ventilation
SG			749 Emergency Ventilation
ND		System 122 Turbine Building	
ND			727 Local Air Cooler Systems
ND			744 Turbine Building Ventilation
ND		System 124 Chemical Systems Building	
NS			345 Monitored Floor Drain Systems
ND			727 Local Air Cooler Systems
ND			743 Chemical Systems Building Ventilation ND
ND		System 125 Control Systems Building	
NS			345 Monitored Floor Drain Systems
ND			745 Other Controlled Ventilation Systems
SG			746 Control Building Ventilation
ND			748 Other Non-controlled Ventilation Systems

Table C-3

Safety Class	Fed System	System	Supporting Systems
ND			765 Non Controlled Drains
ND		System 126 Electrical Systems Building	
ND			727 Local Air Cooler Systems
ND			747 Electrical & Diesel Building Ventilation
ND			765 Non Controlled Drains
ND		System 127 Diesel Building	
ND			727 Local Air Cooler Systems
ND			747 Electrical & Diesel Building Ventilation
ND			765 Non Controlled Drains
NS		System 157 Concrete Vessel Liner Cooling System	
NS			722 Diesel Backed Closed Cooling Sys. for Components and Pool Water Tank Supply
NS		System 3121 Filtered Water Supply System (1) Supplies Primary System with clean water of equal boron content (2) Replaces water withdrawn by System 3125 (3) Controls hot/cold interface in System 3222	
NS			533 Water Chemistry Systems to adjust chemistry using 3333 & 3335
ND			3311 Primary Water Filter System and Water Supply
ND			3333 Lithium Hydroxide Supply System
ND			3335 Hydrogen Supply System
ND			535 Reactor Power Control System
ND	3124 Inlet-Outlet System		
ND		System 3122 Distilled Water Supply System supplies Primary System with water which contains less than 10 ppm boron	
ND			3312 Boron Concentrating System supplies water from system 331 tank 22
NS			533 Water Chemistry Systems to adjust chemistry
ND			3333 Lithium Hydroxide Supply System
ND			3335 Hydrogen Supply System
ND			535 Reactor Power Control System
ND	3124 Inlet-Outlet System		
ND		System 3123 Borated Water Supply System supplies Primary System with 7000 ppm borated water to adjust power or for shutdown	
ND			3312 Boron Concentrating System water from System 331 Tank 23
ND		System 3124 Inlet-Outlet System (1) supplies and preheats water to Primary System from Systems 3121, 3122, & 3123, (2) withdraws water from the Primary System to the Reactor Water Cleanup System	
NS			3121 Filtered Water Supply System supplies and preheats water,
ND			3122 Distilled Water Supply System supplies and preheats water,

Table C-3

Safety Class	Fed System	System	Supporting Systems
ND			3123 Borated Water Supply System supplies and preheats water,
ND	3125 Normal Withdrawal System		supplies and cools water to Reactor Water Cleanup System 331
ND		3131 Hot and Cold Leg Pipes	
ND		System 3125 Normal Withdrawal System	cools, depressurizes and measures boron content for System 331
ND			535 Reactor Power Control System controlling system
ND			3124 Inlet-Outlet System
NS	331 Reactor Water Cleanup System		
ND		System 3126 Temporary Discharge System	supports Discharge System 3125 during transient situations
ND			3124 Inlet-Outlet System
SG	316 Condensation System		discharged water
ND		System 3131 Hot and Cold Leg Pipes	
ND		3124 Inlet-Outlet System	provides injection/withdrawal and heat exchange
ND		System 3133 Main Coolant Pumps	
NS			649 Main Coolant Pump Power Supply
NS			723 Non Diesel Backed Closed Cooling System
NS			534 Main Coolant Pump Control Speed Control ND
			734 High Pressure Distilled Water System which prevent ingress of Primary Water
SG		System 314 Pressure Relief System	contains 4 safety valves, 2 relief valves, blowdown pipes in containment walls
SG	316 Condensation System		receives and condenses steam discharged from pressurizer dome, via System 314.
SG		System 316 Condensation System	receives and condenses steam,
SG			314 Pressure Relief System
ND			3126 Temporary Discharge System
ND			3132 Steam Generators
NS			355 Containment Pool Cooling System cools Condensation System Pool Water,
NS			324 Fuel Pool Water Cooling and Cleanup adjusts boron concentration and water inventory
SG		System 321 Reactor Pool	provides heat sink for one week.
SG			314 Pressure Relief System if all residual heat removal is lost,
SG			316 Condensation System, if all residual heat removal is lost,
NS			323 Reactor Pool Cooling System - loss of both pumps will yield shutdown

Table C-3

Safety Class	Fed System	System	Supporting Systems
NS			325 Reactor Pool Water Replacement System via Cleanup System 331,
NS			712 Diesel Backed Service Water System
NS			721 Diesel Backed Closed Cooling System for reactor pool,
SG			724 Passive Closed Cooling System for Reactor Pool if System 323 is not available,
SG			714 Cooling Tower For Reactor Pool.
ND		322 Pool Loop Inactivation System - flow is provided through this system from pool, when a main coolant pump is not operating,	
ND		3281 Pressurizer, interface exists between Pzr and the pool.	
NS		System 323 Reactor Pool Cooling System	
ND			313 Recirculation System,
NS			661 Diesel Backed 660 VAC Power Supply System,
NS			721 Diesel Backed Closed Cooling System for reactor pool,
NS			722 Diesel Backed Closed Cooling System for components, pool, and motors.
SG		321 Reactor Pool	
NS		System 324 Fuel Pool Water Cooling and Cleanup cools Spent Fuel Pool, and Fuel Service Pool	
NS			722 Diesel Backed Closed Cooling System for components and pool.
NS		System 325 Reactor Pool Water Replacement System	
NS			331 Reactor Water Cleanup System withdraws pool water for cleanup and boron recirculation in System 331,
ND			332 Plant Shutdown Reactor Pool Cleanup during refueling System
NS			533 Water Chemistry Control System monitors boron concentration in withdrawn/injected water,
			333 Chemical Addition System
			723 Non-diesel-backed Closed Cooling System
ND		System 3251 Supply Pump	
ND			3312 Boron Concentrating System,
ND			3313 Pool Water Filter System,
NS			533 Water Chemistry Control System - measures chemistry variables
ND			3333 Lithium Hydroxide Supply System,
ND			3335 Hydrogen Supply System,
SG		321 Reactor Pool	

Table C-3

Safety Class	Fed System	System	Supporting Systems
ND		System 3253 Withdrawal System	
SG			321 Reactor Pool System 3253 withdraws water from pool just below System 313,
NS			723 Non Diesel Backed Closed Cooling System
NS	533 Water Chemistry Systems,		
ND	3313 Pool Water Filter System feeds 3313 via two pipes: "A" pipe 4 kg/s, and "B" pipe		
ND		System 3255 Pool Level Instrumentation if the system detects too large or too small rates of flow from System 3251 the reactor will be scrammed and isolation will occur.	
SG	516 Trip and Interlock System.		
ND		System 3281 Pressurizer (PZR)	
NS	538 Reactor Pressure Control System		
ND			3283 PZR Heating System
ND			3285 PZR Offgassing system
ND			3288 PZR Spray System
NS			538 Rx Pressure Control System
NS			723 Non-diesel-backed Closed Cooling System
	PZR Level & Pressure Measurements		
	Control signals	NS	
	Alarm signals	NS	
	Scram signals	SG	
NS		System 331 Reactor Water Cleanup System receives and cleans water from Primary Loop & Reactor Pool,	
ND	312 Reactivity Control System prepares clean water and borated/deborated feed water,		
NS	325 Reactor Pool Water Replacement System prepares clean water and borated feed water		
ND		System 3311 Primary Water Filter System degasses and removes particulates	
ND		3125 Normal Withdrawal System which obtains the water for filtering.	
ND		System 3312 Boron Concentrating System concentrates borated water to 7000 ppm boron concentration,	
ND			3311 Primary Water Filter System,
ND			3313 Pool Water Filter System.
ND		System 3313 Pool Water Filter System - scram stops withdrawal via system 325,	
NS		325 Reactor Pool Water Replacement System, with- draws pool water for filtering.	
NS		System 333 Chemical Additives System supplies chemicals to maintain proper makeup of Primary System and Pool	
ND			3339 Supply System For Water,
ND			591 Plant Power Overall Control sets operating mode for controller,
NS			533 Water Chemistry Systems analysis instruments,
ND			312 Reactivity Control System flow meters,

Table C-3

Safety Class	Fed System	System	Supporting Systems
NS			325 Reactor Pool Water Replacement System flow-meters,
ND	312 Reactivity Control System,		
NS	325 Reactor Pool Water Replacement System,		
NS	331 Reactor Water Cleanup System.		
ND		System 3331 Boron Supply System operated manually, or on orders, prepares 7000 ppm borated water,	
NS	331 Reactor Water Cleanup System	supplies tank in system.	
ND		System 3333 Lithium Hydroxide Supply System	
ND	312 Reactivity Control System,		
NS	325 Reactor Pool Water Replacement System.		
ND		System 3337 Hydrazine Supply System	
ND	3122 Distilled Water Supply System,		
ND	3251 Supply Pump.		
ND		System 3339 Supply System For Water - supplies other subsystems with distilled water.	
ND		733 Demineralized Water Distribution System,	
ND		3312 Boron Concentrating System using tank in System 331.	
NS		System 336 Sampling System samples water, steam, sludge, and gas,	
NS	338 Controlled Drains System.		
NS		System 342 Liquid Waste System	
NS		324 Spent Fuel Pool Cooling System	
NS		331 Reactor Water Cleanup System	
NS		338 Controlled Drains System	
NS		345 Monitored Floor Drain System	
NS		733 Demin Water Distribution System	
NS		753 Compressed Air System	
NS	331 Reactor Water Cleanup System,		
NS	343 Solid Waste System sludge.		
NS		System 345 Monitored Floor Drain Systems collects drain water from plant, NS	
	342 Liquid Waste System for treatment,		
ND	112 Cooling Water Outlet Channels.		
NS		System 349 Off-Gas Filter System	
NS		341 Off-Gas Release System.	
NS		344 Decontamination Systems	
NS		745 Other Controlled Area Ventilation Systems	
NS		System 355 Containment Pool Cooling System, cools Condensation Pool and replaces water,	

Table C-3

Safety Class	Fed System	System	Supporting Systems
SG ND ND NS NS	316 Condensation System,		3312 Boron Concentrating System for makeup and adjustment of boron for the Condensation Pool, 3313 Pool Water Filter System for makeup and adjustment of boron for the Condensation Pool. 324 Spent Fuel Pool Cooling System 722 Diesel-backed Closed Cooling System for Component Cooling
ND ND PS SG		System 4111 Steam Generator Output Steam Lines during turbine transients prevents system pressure from exceeding design pressure.	3132 Steam Generators,
	411 Main Steam Line System Steam Drum Tank #31,		
	316 Condensation System on overpressure opens up to three safety relief valves and dumps steam to Pool.		
ND PS NS NS NS ND ND NS ND		System 4113 Turbine Admission Steam Lines - four steam lines from drum to Turbine Admission.	411 Main Steam Line System Steam Drum Tank #31
	413 Steam Bypass System during Hot Standby, plant startup, power reductions, and after turbine trips,		
	415 Turbine Sealing System, approx 1 kg/s,		
	419 Live Steam Feedwater Heating, approx 30 kg/s,		
	422 Steam Admission System, steam admission up to 950 kg/s,		
	423 Moisture Separator and Steam Reheat System, 66 kg/s,		
	442 Condenser Gas Removal System steam driven air ejector, 1 kg/s,		
	4632 Feed Pump Drive Turbine System.		
NS PS ND ND		System 413 Steam Bypass System blows steam from the Steam Line System 411 directly to the Turbine Condenser manages 60% flow continuously, 100% for short periods,	411 Main Steam Line System, includes relief valves, 597 Turbine Control System,
	441 Turbine Condenser.		
NS NS ND		System 415 Turbine Sealing System	411 Main Steam Lines
	421 Main Turbine seals shaft for leakage of steam out of turbine.		
	4632 Feed Pump Drive Turbine System, also various valves.		
NS PS ND		System 419 Live Steam Feedwater Heating	411 Main Steam Line System,
	4635 High Pressure Feedwater Heaters last high pressure feedwater heater.		
ND NS		System 422 Steam Admission System,	
	421 Main Turbine.		
ND		System 426 Plant Control and Trip Oil System protects turbine against	

Table C-3

Safety Class	Fed System	System	Supporting Systems
NS		overspeed, and high pressure in moisture separator	425 Lubrication and Jacking Oil Systems for EHC.
NS		System 4281 Turbine Turning Gear starts turning turbine at startup,	
NS			661 Diesel Backed 660 VAC Power Supply System.
NS		System 4283 Exhaust Hood Water Spray System, sprays turbine casing and exhaust steam when air pressure is too high in turbine condenser	NS
			461 Condensate System.
ND		System 429 Turbine Protection System against overspeed, vibration, and vacuum breaker,	
ND			426 Plant Control and Trip Oil System.
ND		System 441 Turbine Condenser	
NS			442 Condenser Gas Removal System for non-condensable gases.
NS		System 446 Condenser Makeup Water System,	
ND	441 Turbine Condenser,		
NS	461 Condensate System,		
ND	463 Feedwater System,		
NS	467 Condensate Cleanup System.		
NS		System 451 Condenser Cooling Pump System supplies cooling water	
NS			161 Cooling Tower,
NS			1121 Water Intake Channel
NS			456 Circulating Water Makeup Water System,
NS			642 Auxiliary Power 6KV AC System,
ND	441 Turbine Condenser evacuates Water Box air, cleans tubes,		
NS	453 Cooling Tower Pump System via System 441.		
NS		System 453 Cooling Tower Pump System - supplies Cooling Tower System 160 with water from System 441 when power is available via System 642,	
ND			441 Turbine Condenser,
NS			642 Auxiliary Power 6KV AC System,
ND	112 Cooling Water Outlet Channels,		
ND	160 Cooling Tower System		
NS		System 456 Circulating Water Makeup Water System replaces water spent by cooling tower and letdown to river,	
NS			1121 Water Intake Channel,
NS			711 Cooling Water Intake System
NS	451 Condenser Cooling Pump System,		
NS	457 Water Quality Control System.		
NS		System 461 Condensate System,	
ND			441 Turbine Condenser,
NS			446 Condenser Makeup Water System along with

Table C-3

Safety Class	Fed System	System	Supporting Systems
NS			system 461 controls water level in turbine condenser,
NS			461 Condensate System,
NS	413 Steam Bypass System,		467 Condensate Cleanup System,
NS	4283 Exhaust Hood Water Spray System,		
ND	447 Seal Water System,		
ND	463 Feedwater System,		
ND		System 463 Feedwater System	the feedpumps control the plants power output,
NS			461 Condensate System,
ND	313 Recirculation System		to steam generator.
ND		System 4631 Main Feed Pump	feeds steam generators when their need exceeds the capacity of the Auxiliary Feed System
NS			461 Condensate System feedwater tank
ND	4635 High Pressure Feedwater Heaters,		
ND	4636 System Inlet Pipe System,		
ND	3132 Steam Generators		via Systems 4635 and 4636.
ND		System 4634 Auxiliary Feedpump System	feeds Steam Generators until main feedpump drive can be used,
NS			461 Condensate System,
NS			643 Auxiliary 660 VAC System powered by A and B buses,
ND	3132 Steam Generators,		
ND	4636 System Inlet Pipe System		by bypassing the high pressure feedwater heater system 4635
ND		System 4636 System Inlet Pipe System,	
ND			4634 Auxiliary Feedpump System,
ND			4635 High Pressure Feedwater Heaters,
ND	3132 Steam Generators,		
ND		System 472 Turbine Plant Leakage and Drain Systems,	
NS	342 Liquid Waste System.		
SG		System 5161 Reactor Scram Trip System	scrams the reactor if a runback or shutdown is not sufficient. System creates a scram alarm System 5161 will be tripped by events which also trip System 5162,
SG			5162 Hot Reactor Shutdown Trip System,
NS	3121 Filtered Water Supply System		interrupts power to system,
ND	3122 Distilled Water Supply System		interrupts power to system, also closes shutoff valves in inlet lines,
ND	3133 Main Coolant Pumps		one pump is tripped,
ND	517 Alarm Display System		scram alarm is tripped,
NS	527 Safety Parameter Display System		Scram Alarm Survey is tripped,
NS	649 Main Coolant Pump Power Supply,		

Table C-3

Safety Class	Fed System	System	Supporting Systems
SG		System 5162 Hot Reactor Shutdown Trip System	shuts down reactor for certain incidents which include manual scram,
NS			325 Reactor Pool Water Replacement System - too high and too low pressurizer pool level,
ND			328 Pressurizer Systems - too high vessel pressure
NS	421 Main Turbine - turbine trip, without steam bypassed to system 441 via system 413,		
SG		5163 Isolation Trip System	trip of Isolation Trip System,
SG		531 Neutron Flux Monitoring System	too high or PS
		536 Reactor, Pool, and Vessel Monitoring	too high reactor outlet temperature
		Note: ABB has stated in their comments on this report that even though a system above (e.g. system 325) may be designated as NS or ND, all signals to system 5162 are SG.	
ND	463 Feedwater System	orders rapid reduction of feedwater by throttling steam	
NS	3121 Filtered Water Supply System	isolates power supply for system,	
ND	3122 Distilled Water Supply System	isolates power supply for system,	
ND	3123 Borated Water Supply System	injection of borated water,	
ND	517 Alarm Display System	trips alarm,	
NS	527 Safety Parameter Display System	trips plant shutdown survey.	
SG		System 5163 Isolation Trip System	limits loss of reactor pool water and isolates pipes if there is a rupture
NS		325 Reactor Pool Water Replacement System	activates on low level signal,
NS		3412 Half Hour Delay Off Gas System	activates on high radiation level signal,
SG		546 Pipe Rupture Monitoring System	measuring channels
SG		554 Steam Generator Tube Rupture Monitoring	activates on high radiation,
PS		583 Containment Atmosphere Monitoring	activates on high temp or pressure measurement,
ND	312 Reactivity Control System	closes isolation valves,	
NS	325 Reactor Pool Water Replacement System	isolation,	
ND	328 Pressurizer Systems	isolation,	
PS	411 Main Steam Line System	isolates and trips alarms,	
ND	592 Turbine Protection System	trips,	
SG	5161 Reactor Scram Trip System,		
SG	5162 Hot Reactor Shutdown Trip System,		
ND	517 Alarm Display System,		
NS	527 Safety Parameter Display System,		
NS	843 Safety Parameter Display System.		

Table C-3

Safety Class	Fed System	System	Supporting Systems
SG		System 5165 Startup Interlock System prevents criticality in stages of refueling and for service shutdowns; manual trip can be initiated in main control room,	
ND			328 Pressurizer Systems low pressure signal from vessel,
ND	313 Recirculation System	interrupts power to one main coolant pump,	
NS	649 Main Coolant Pump Power Supply	interrupts power to one main coolant pump,	
NS	3121 Filtered Water Supply System	interrupts power to water injection system,	
ND	3122 Distilled Water Supply System	interrupts power to water injection system,	
SG		System 518 Emergency Control Room monitors vessel and containment when central control room is inoperable,	
SG			673 110 V DC Auxiliary Power,
SG			516 Trip and Interlock System,
SG			546 Pipe Rupture Monitoring System,
PS			583 Containment Atmosphere Monitoring.
SG		System 531 Neutron Flux Monitoring System,	
ND	525 Core Management Computer System	supplies core parameters, monitors startup sequence,	
NS		System 534 Main Coolant Pump Control - controls speed of Main Coolant Pumps, pumps are limited to 20% power on administrative order.	
ND			3221 Lower Density Lock temperature Interface Monitors,
PS			536 Reactor, Pool, and Vessel Monitoring mixed reactor outlet temperature data.
ND			591 Plant Power Overall Control gives operating Mode Selection orders,
NS	649 Main Coolant Pump Power Supply		
ND		System 535 Reactor Power Control System,	
ND			312 Reactivity Control System,
ND			322 Pool Loop Inactivation System,
SG			531 Neutron Flux Monitoring System,
PS			536 Reactor, Pool, and Vessel Monitoring gets current value,
ND			591 Plant Power Overall Control gets operating mode, and mixed outlet temperature set values,
ND	312 Reactivity Control System	all subsystems of 312 receive operating orders	
ND	3222 Upper Density Lock	controls hot-cold water interface	
SG		System 5361 Outlet Water Temperature Monitoring	
SG	516 Trip and Interlock System	provides signals for scram,	
NS	534 Main Coolant Pump Control	(during startup),	
ND	535 Reactor Power Control System		
SG		System 5362 Inlet Water Temperature Monitoring	
ND	535 Reactor Power Control System,		
ND	520 Computer Systems.		

Table C-3

Safety Class	Fed System	System	Supporting Systems
NS		System 5363 Fuel Assembly Outlet Water Temperature	
ND	525	Core Management Computer System provides margin for net boiling in hot channel information.	
SG		System 5364 Reactor Pool Temperature Monitoring	
NS	323	Reactor Pool Cooling System for possible startup of standby pump	
NS	712	Diesel Backed Service Water System if needed,	
NS	721	Diesel Backed Closed Cooling System for Reactor Pool if needed.	
NS		System 537 Pressurizer Pool Level Control,	
NS		325 Reactor Pool Water Replacement System,	
ND		591 Plant Power Overall Control,	
ND	312	Reactivity Control System, via System 535,	
ND	328	Pressurizer Systems via System 325.	
NS		System 538 Reactor Pressure Control System	
NS		325 Reactor Pool Water Replacement System,	
ND		328 Pressurizer Systems,	
ND		524 Control Processors micro-computer,	
ND		591 Plant Power Overall Control,	
SG	314	Pressure Relief System activates system,	
ND	3281	Pressurizer keeps dome pressure constant,	
ND	3283	Pressurizer Heating System initiates heating,	
ND	3288	Pressurizer Spray System	
NS	648	Pressurizer Heater Supply System Switchgear	
ND		System 542 Process Control Principles - system only specifies principles,	
NS	521	Main Process Computer,	
ND	524	Control Processors does Process Data Handling,	
SG		System 546 Pipe Rupture Monitoring System monitors primary and secondary lines,	
SG	5163	Isolation Trip System.	
SG		System 554 Steam Generator Tube Rupture Monitoring - self contained system, provides alarm	
PS	411	Main Steam Line System initiates isolation.	
PS		System 555 Room Radiation Monitors - in case of a refueling accident,	
ND	517	Alarm Display System sets off alarms,	
NS	527	Safety Parameter Display System starts up system,	
PS	742	Reactor Building Ventilation shuts down system,	
SG	749	Emergency Ventilation - starts emergency ventilation,	
NS	843	Alarm System	
ND		System 591 Plant Power Overall Control selects operating mode, & generates system variables,	
ND		523 Man-Machine Interface With Control Room,	
ND		524 Control Processors	

Table C-3

Safety Class	Fed System	System	Supporting Systems
NS	534 Main Coolant Pump Control,		
ND	535 Reactor Power Control System,		
ND	597 Turbine Control System,		
ND	598 Feedwater Control System,		
ND		System 592 Turbine Protection System	
NS	413 Steam Bypass System,		
ND	422 Steam Admission System,		
SG	516 Trip and Interlock System.		
ND		System 594 Turbine Plant Supervisory System,	
ND		592 Turbine Protection System measures vibrations	
NS		4291 Turbine Overspeed Protection System,	
NS		4293 Turbine Vibration Protection System,	
ND	592 Turbine Protection System - trips system,		
ND		System 595 Turbine Plant Automatic Operations System	
ND		591 Plant Power Overall Control,	
ND	593 Turbine Plant Process Control System.		
ND		System 597 Turbine Control System	
ND		591 Plant Power Overall Control gets operating mode selection and pressure,	
PS	411 Main Steam Line System - maintains pressure in system		
NS	413 Steam Bypass System - orders steam bypass control valves to open or close,		
ND	422 Steam Admission System - orders steam admission valves to open or close.		
ND		System 598 Feedwater Control System - In mode A it maintains level in steam generator constant (feedwater flow equal to steam flow); in mode B keeps reactor outlet temperature at 290 degrees C,	
ND		591 Plant Power Overall Control,	
ND	4631 Main Feed Pump,		
ND	4632 Feed Pump Drive Turbine System,		
ND	4634 Auxiliary Feedpump System,		
ND	4636 System Inlet Pipe System,		
NS		System 611 Main Transformer feeds power to Grid, 400 KV system or from grid to system 642,	
ND		621 Main Grid Switchyard circuit breaker opens on abnormal conditions,	
NS		631 Generator Bus System at startup circuit breaker is open,	
NS		642 Auxiliary Power 6KV AC System feeds power to 6 KV auxiliary power system,	
ND	630 Generator Bus Systems at loss of load separates bus from grid,		
ND	640 Auxiliary AC Power Supply System,		
NS	660 Diesel Backed AC Power Systems,		
PS	670 Battery Backed Power Supplies.		

Table C-3

Safety Class	Fed System	System	Supporting Systems
NS		System 631 Generator Bus System at startup, separates system 431 from 611 and 642,	ND 431 Main Generator,
ND	610 High Voltage Systems		
ND	621 Main Grid Switchyard		
ND	640 Auxiliary AC Power Supply System,		
NS	660 Diesel Backed AC Power Systems,		
PS	670 Battery Backed Power Supplies,		
NS		System 642 Auxiliary Power 6KV AC System A and B Buses,	
NS		631 Generator Bus System,	
ND		622 Startup Grid Switchyard	
NS		642 6-KV Auxiliary Power AC System	
NS	643 Auxiliary 660 VAC System plant loads exceeding 350 KW,		
ND	648 PZR Heater Power Supply		
ND	649 Main Coolant Pump Power Supply		
NS	661 Diesel-backed AC Power Supply System		
SG	673 Safety Grade 110 VDC Auxiliary Power Supply System, fed through rectifier from system 661		
NS	677 440 V Battery-backed Auxiliary Power AC System		
NS		System 648 Pressurizer Heater Supply System,	
ND	3283 Pressurizer Heating System supplies power to system.		
NS		System 649 Main Coolant Pump Power Supply - thyristors and control, wet motor pumps,	
SG		516 Trip and Interlock System tripping signal goes to frequency converter in system 649.	
NS		534 Main Coolant Pump Control, in order to maintain hot/cold interface in system 3221 at constant level	
ND	3133 Main Coolant Pumps - at startup each pump is started individually to 20%.		
NS		System 651 Diesel Plant - this is to protect the capital investment. The system has two 660 VAC Diesel Sets,	
NS	661 Diesel Backed 660 VAC Power Supply System Main Bus.		
NS		System 661 Diesel Backed 660 VAC Power Supply System - protects plant instrumentation, when loss of power is noted, major loads are disconnected,	
NS		643 Auxiliary 660 VAC System - when regular power from system 642 is interrupted,	
NS		651 Diesel Plant	
ND	312 Reactivity Control System,		
NS	323 Reactor Pool Cooling System,		
NS	355 Containment Pool Cooling System,		
NS	425 Lubrication and Jacking Oil Systems,		
ND	428 Other Turbine Auxiliaries,		
NS	467 Condensate Cleanup System,		

Table C-3

Safety Class	Fed System	System	Supporting Systems
PS	670 Battery Backed Power Supplies,		
SG	673 110 V DC Auxiliary Power,		
NS	677 440 V Battery Backed Auxiliary AC Power - when regular supply fails,		
NS	712 Diesel Backed Service Water System,		
NS	721 Diesel Backed Closed Cooling System for Reactor Pool,		
NS	722 Diesel Backed Closed Cooling Sys. for Components and Pool,		
ND	747 Electrical and Diesel Building Ventilation.		
SG		System 673 110 V DC Auxiliary Power, plant can be still be monitored with loss of channels in system 673,	
NS			661 Diesel Backed 660 VAC Power Supply System
ND		System 686 Power Measuring Systems	
ND			520 Computer Systems.
NS		System 711 Cooling Water Intake System, principally removes trash,	
NS			456 Circulating Water Makeup Water System,
NS	712 Diesel Backed Service Water System,		
NS	713 Non Diesel Backed Service Water System,		
NS		System 712 Diesel Backed Service Water System,	
NS	651 Diesel Plant		
NS	721 Diesel Backed Closed Cooling System for Reactor Pool,		
NS	722 Diesel Backed Closed Cooling Sys. for Components and Pool,		
NS	451 Condenser Cooling Pump System - secondary option,		
NS	456 Circulating Water Makeup Water System - secondary option,		
NS		System 713 Non Diesel Backed Service Water System (1) Cools systems which can use river water, (2) Does not need Diesel backed cooling (3) From which no radiation leakage needs to be assumed	
ND			112 Cooling Water Outlet Channels - input from Cold Water Channel,
NS			451 Condenser Cooling Pump System - output to Warm Water Channel,
ND	400 Steam and Power Conversion - various minor loads,		
NS	425 Lubrication and Jacking Oil Systems,		
NS	723 Non Diesel Backed Closed Cooling System,		
NS	726 Chilled Water System.		
NS		System 721 Diesel Backed Closed Cooling System for Reactor Pool,	
NS			712 Diesel-backed Service Water System
NS	323 Reactor Pool Cooling System,		
SG	321 Reactor Pool,		
NS		System 722 Diesel Backed Closed Cooling System for Components and Pool,	
			NS 355 Containment Pool Cooling System,
NS	157 Concrete Vessel Liner Cooling System,		
ND	313 Recirculation System,		

Table C-3

Safety Class	Fed System	System	Supporting Systems
SG	316	Condensation System,	
NS	323	Reactor Pool Cooling System,	
NS	324	Fuel Pool Water Cooling and Cleanup.	
NS	355	Containment Pool Cooling System	
ND	2111	RPV Anchor Flange Cooling	
ND		System 7221 Wall Cooler System	
ND			7225 Pump System for 722,
NS	157	Concrete Vessel Liner Cooling System,	
ND	2111	RPV Anchor Flange Cooling	
ND		System 7222 Pump Cooler System,	
ND			7225 Pump System for 722,
ND	313	Recirculation System,	
NS	323	Reactor Pool Cooling System,	
NS		System 723 Non-Diesel Backed Closed Cooling System - system does not need Diesel backup	
NS		713 Non Diesel Backed Service Water System - as heat sink (transmits 10 to 15 MW),	
ND	312	Reactivity Control System,	
NS	325	Reactor Pool Water Replacement System,	
ND	328	Pressurizer Systems,	
NS	331	Reactor Water Cleanup System,	
ND	727	Local Air Cooler Systems,	
ND	740	Ventilation System,	
ND	741	Containment Ventilation.	
SG		System 724 Passive Closed Cooling System for Reactor Pool - eight loops and heat exchangers with isolation valves,	
SG			321 Reactor Pool,
ND	714	Cooling Tower For Reactor Pool - heat sink for Pool Cooling	
NS		System 726 Chilled Water System - can be switched to outdoor cooling mode	
ND	727	Local Air Cooler Systems,	
ND	740	Ventilation System.	
ND		System 731 Raw Water Treatment,	
ND	732	Water Demineralization System,	
NS	761	Tap Water System.	
ND		System 732 Water Demineralization System	
ND			731 Raw Water Treatment,
NS			761 Tap Water System,
ND	733	Demineralized Water Distribution System Storage Tank - demand for water is signalled via level monitors.	

Table C-3

Safety Class	Fed System	System	Supporting Systems
ND		System 733 Demineralized Water Distribution System.	
ND			732 Water Demineralization System,
NS			753 Compressed Air Non Diesel Backed System,
ND	121 Reactor Building,		
ND	122 Turbine Building,		
ND	123 Air Cooling Systems Pump Building,		
ND	124 Chemical Systems Building,		
ND	129 Workshops,		
ND	135 Water Treatment Building.		
ND		System 7343 High Pressure Purge Water System,	
ND	313 Recirculation System,		
NS	323 Reactor Pool Cooling System.		
SG		System 749 Emergency Ventilation - services reactor service room filters exhaust into containment,	
ND	144 Main Plant Stack,		
SG		System 751 Bottled Air Diesel Backed System - power to safety grade pneumatic valves if system 643 fails	
NS			753 Compressed Air Non Diesel Backed System local switching,
NS		System 761 Tap Water System - for drinking water, toilets etc.	
ND			731 Raw Water Treatment
ND	732 Water Demineralization System.		

C.3 Systems with No Identified Dependencies

Table C-4 presents systems for which dependencies were not identified. Some of the systems have no active dependencies; other do but the dependencies are not regarded to have safety significance, or the dependencies are presented in Table C-3.

Table C-4 Systems with No Identified Dependencies

Safety Class	System Identification
SG	System 151 Vessel Monolith
SG	System 152 Containment
SG	System 1521 Containment Vessel
SG	System 1522 Elevated Cooling Tower Compartments
SG	System 1523 Service Pool & Fuel Pool Structures
SG	System 1524 Containment Access Penetrations

Table C-4 Systems with No Identified Dependencies

Safety Class	System Identification
SG	System 1525 Containment Vessel Pipe Penetrations
SG	System 159 Concrete Vessel Liner Venting System
ND	System 160 Cooling Tower System
ND	System 170 Temporary Buildings
ND	System 180 Hangers & Restraints
PS	System 231 Refueling Systems
ND	System 260 Reactor Fuel
ND	System 290 Thermal Insulation
ND	System 291 Wet Thermal Insulation for Primary System
ND	System 293 Wet Thermal Insulation for Pressure Vessel
ND	System 295 Thermal Insulation Outside Pressure Vessel
ND	System 312 Reactivity Control System
ND	System 3124 Inlet-Outlet System
ND	System 313 Recirculation System (1) Operates the Primary Loop in such a way that the Pool Loop is not activated, (2) Delivers power to the steam generators
ND	System 3132 Steam Generators
ND	System 322 Pool Loop Inactivation System - Separates the Primary Loop from the Pool
ND	System 3221 Lower Density Lock
ND	System 3222 Upper Density Lock
ND	System 3225 Gas Lock For Startup - Separates Primary Loop from Pool by means of a Gas Bubble
ND	System 328 Pressurizer Systems
ND	System 3283 Pressurizer Heating System
ND	System 3285 Pressurizer Offgassing System - Removes Hydrogen & Nitrogen from Hydrolizers
NS	System 338 Controlled Drains System
NS	System 341 Off-Gas Release System - For decay of radiated gas

Table C-4 Systems with No Identified Dependencies

Safety Class	System Identification
NS	System 343 Solid Waste System
NS	System 344 Decontamination System - Storage area for radiated components, Decon area.
PS	System 411 Main Steam Line System parts inside Containment up to Isolation valves are safety grade; other parts are non-safety grade: (1) Brings steam from SG to Steam Admission System of Main Turbine, (2) Brings steam to Turbine Sealing System, (3) Brings steam to Air Ejectors of System 442, (4) At startup sends steam to the Bypass, (5) Sends steam to Feedwater Heating System
NS	System 421 Main Turbine - One high pressure and three low pressure sections
ND	System 423 Moisture Separator and Steam Reheat System
NS	System 425 Lubrication and Jacking Oil Systems supplies lubrication and cooling for turbine and generator bearings,
ND	System 427 Steam Extraction System
ND	System 428 Other Turbine Auxiliaries
ND	System 431 Main Generator
ND	System 432 Exciter and Voltage Control
ND	System 433 Seal Oil System
ND	System 435 Generator Cooling Water System
ND	System 437 Hydrogen and Carbon Dioxide Systems
NS	System 457 Water Quality Control System - Adds chemicals and keeps impurities at a low level
ND	System 4632 Feed Pump Drive Turbine System - Controls Plant Power when SG need exceeds Auxiliary feedpump delivery
ND	System 4635 High Pressure Feedwater Heaters
NS	System 467 Condensate Cleanup System - Cleanup system for particulates etc.
ND	System 473 Turbine Plant Drying System - For components drained during Shutdown to protect them against corrosion
NS	System 511 Main Control Room - Some actions can be taken from Emergency System 518, or Security Center 858
ND	System 517 Alarm Display System
NS	System 521 Main Process Computer

Table C-4 Systems with No Identified Dependencies

Safety Class	System Identification
ND	System 523 Man-Machine Interface With Control Room
ND	System 524 Control Processors
ND	System 525 Core Management Computer System - has a 3D model of core. Core management on line, fuel management off line
NS	System 527 Safety Parameter Display System - Displays operating status, and forecasts future situations
NS	System 533 Water Chemistry Systems
ND	System 5331 Boron Concentration Monitoring
ND	System 5333 PH Control System
ND	System 5335 Hydrogen Gas Control
ND	System 5337 Conductivity Monitoring System
PS	System 536 Reactor, Pool, and Vessel Monitoring
SG	System 5366 Concrete Vessel Temperature Monitoring
NS	System 5367 Primary Loop Flow Monitoring
ND	System 541 Process Measurement Principles - methods for measuring channels such as transducers, transmitters, etc.
ND	System 545 Leak Detection Principles
ND	System 547 Process Monitoring System - Initiates alarms for small fractures, and alarms for leakage from pools and insufficient cooling
NS	System 551 Extended Delay Off-Gas System Monitoring
NS	System 552 Half Hour Delay Off-Gas System Monitoring - Warns of small steam generator leaks
NS	System 553 Stack Radiation Monitors
ND	System 556 Portable Radiation Monitors
ND	System 557 Environment Radiation Monitors
PS	System 583 Containment Atmosphere Monitoring
ND	System 584 Earthquake Monitoring System
ND	System 588 Meteorological Instrumentation

6 of 6

Table C-4 Systems with No Identified Dependencies

Safety Class	System Identification
ND	System 593 Turbine Plant Process Control System
ND	System 621 Main Grid Switchyard
ND	System 622 Startup Grid Switchyard
NS	System 643 Auxiliary 660 VAC System - Two buses for all 660 V loads
NS	System 677 440 V Battery Backed Auxiliary AC Power - System 677 is provided with its own DC Power, loss of power activates the PIUS self protection features
ND	System 682 Synchronizing System
ND	System 683 Feeder Switching
ND	System 684 Motor Sequence Starting
ND	System 685 Relay Protection
ND	System 689 Energy Logging System
ND	System 691 Power Cables
ND	System 714 Cooling Tower For Reactor Pool - Cools the Reactor Pool
ND	System 7211 Cooling Units
ND	System 7215 Pump System for 721
ND	System 7223 Condensation Pool Cooling System
ND	System 7224 Spent Fuel Pool Cooling System - On Blackout systems 323, 712, 721, and system 722 are lost
ND	System 7225 Pump System for 722 - Feeds all 722 sub-systems
ND	System 7261 Ventilation System
ND	System 727 Local Air Cooler Systems
ND	System 734 High Pressure Distilled Water System - For wet motor high pressure pumps
ND	System 7341 Pump System - Feeds system 7343
ND	System 741 Containment Ventilation
PS	System 742 Reactor Building Ventilation
ND	System 743 Chemical Systems Building Ventilation

Table C-4 Systems with No Identified Dependencies

<u>Safety Class</u>	<u>System Identification</u>
ND	System 744 Turbine Building Ventilation
SG	System 746 Control Building Ventilation - Control Room habitability
ND	System 747 Electrical and Diesel Building Ventilation
ND	System 748 Other Non-controlled Ventilation Systems
NS	System 753 Compressed Air Non Diesel Backed System - Supplies most plant needs for pneumatic systems and valves instrument and chemical systems
ND	System 763 Plant Heating System
ND	System 765 Non Controlled Drains
ND	System 766 Sanitary Drains
ND	System 767 Ground Water Drain
ND	System 768 Surface Water Drain
ND	System 866 Gas Extinguishing System
ND	System 869 Fire Detection and Trip

References

Letter of ABB Atom to Dino Scaletti, USNRC, May 11,1993, Subject : Comments on BNL Draft Report.

APPENDIX D

SCRAM

APPENDIX D - SCRAM

If the main cooling of any reactor is lost, the chain reaction must be immediately stopped because no reactor, PIUS, included is designed to remove full reactor power using the cooling systems that are available after loss of the main cooling. Reactivity is controlled by: 1) varying the amount of neutron absorber dissolved in the primary coolant, or 2) causing entry of the reactor pool water already containing high concentrations of neutron absorber. The first mechanism is comparatively slow because of the relative flow rates of system 312 compared with the volume of the primary loops. The second, comparatively fast, mechanism is actuated by an imbalance between the primary pumped flow and the thermally induced flow through the core. Loop flow may be actuated by upsetting this balance. The active way is to remove power from one primary pump either automatically by instruments indicating a process upset and commanding a scram through the main computer control of a designated scram primary pump (active scram - AS) or by a manual actuation of a scram switch that causes the main computer to command a scram using the same designated scram primary pump. If the primary pumps are operating normally but with the reactor power increases or decreases with respect to the rate of heat removal, the flow balance is disturbed to cause activation of the pool loop. This is the passive scram (PS) mechanism of the PIUS design.

It is clear that the three identified scram mechanisms are not independent and diverse. The AS and MS scram through the same computer and same primary pump. Once the scram pump is tripped, they become the same as the PS. Hence the PIUS design is not in accord with NRC requirements for two diverse and independent scram systems. Whether or not a passive scram system is equivalent to two diverse and independent active systems remains to be decided.

All three of the scram mechanisms require activation of the pool loop thus the basic concerns are mechanisms that could result in failure of the passive scram. Identified failure mechanisms are:

1) low pool water, 2) insufficient boration of the pool water, and 3) blocked flow.

D.1 Fueling or Refueling

When fueling PIUS, it would be convenient to have the water level as low as possible, but if the boration is insufficient, the core could go critical during the loading process. If criticality occurs, workmen will be irradiated but there will be little or no offsite effects. Criticality while fueling or refueling is avoided by measuring the reciprocal neutron multiplication factor ($1/M$) for determining how close to critical the core is.

For PIUS, the core instrumentation appears to be removed when refueling. Presumably other instrumentation will be used for protection from accidental criticality.

When PIUS is refuelled, the pool level cannot be lowered because: 1) there is no provision in the plant design for the storage of such a large quantity of contaminated poisoned water, and 2) it is necessary to keep irradiated fuel immersed in water at all times to prevent overheating. Pool level must be as designed in order to perform the pool transfer. Similarly sufficient boration must be in the pool water to prevent accidental criticality at refueling.

The PSID does not identify a system for boration of the reactor pool. Presumably containers of boric acid are released and dissolved into initially pure pool water before the first core load and remains in the water throughout the plant lifetime unless it is necessary to drain the pool for repair work. If this is done, the core would be removed and restart would be the same as the first core load. The pool may be borated using System 312 to provide borated water flow from the primary to the pool through the density locks.

Some boration may be lost about valve and pump seals, and leaks but such would constitute small loss compared with the quantity of boron in the pool.

The boration of the pool is determined by: 1) the amount of boric acid that is dissolved in a known volume of water, 2) the measurement of the neutron multiplication of the core, 3) the boration monitoring system based on neutron

absorption, and 4) chemical laboratory methods which are not described in the PSID but are presumably available to calibrate the boration monitoring system.

It should be noted that more boration is needed to prevent criticality at the water temperature of refueling than at reactor operating temperature. Thus that that is sufficient at refueling is more than sufficient for operation.

Thus insufficient pool water and insufficient boration at fueling or refueling requires multiple failures. If criticality were to occur at this time, the accident would have little offsite effects (mainly noble gas release) with little direct radiation effects on the workmen because of the shielding provided by the pool.

D.2 Power Operation

Assuming that the pool level and pool water boration concentration were sufficient for fuel transfer and to prevent criticality, then they are proper for power operation. The question arises, what can happen during normal operation that can prevent activation of the pool loop and scram the reactor?

Loss of Pool Water

The pool water level in PIUS is automatically regulated to maintain the level in the pressurizer. Should the level drop below the entrance to the upper density locks, the pool loop would not be able to activate. However before this can occur the vacuum breaker would allow steam into the primary to disrupt primary flow and cause a passive scram. (Probably the pumps would also trip on cavitation.) Thus it does not appear that sufficient water to disrupt the pool loop flow can be lost before the reactor will scram. This provides automatic protection against low water caused failure.

The question arises as to how the pool water can be lost. It may be lost through the PCRV or by removal by System 312. Loss through the PCRV requires leakage through both the pool liner and the diaphragm. The water leaking into this intermediate space is collected by a monitoring system that alarms on excessive leakage. Therefore leakage through the PCRV requires failure of: the liner, diaphragm, and water collection system. If the leak is large, reactor operating pressure cannot be maintained thus causing a flow imbalance and passive scram.

Pool water can be removed by System 312. The rate at which it draws off water is 134 gpm. The volume of pool water is about 800,000 gallons hence 100 hours would be required to remove this water but of course only a small amount must be removed before the operation of the upper density lock is disrupted. However, this is contaminated water and cannot be discharged. For this reason, significant loss of pool water through System 312 does not seem possible.

Loss of Boration

Given that at refueling, the pool boration is proper, the question is, how can boration be reduced sufficiently that activation of the pool loop will not shutdown the reactor?

The Handbook of Physics and Chemistry indicates that 6.35 grams of boric acid will dissolve in 100 grams of cold water. This corresponds to 1 boric acid molecule for each 56 water molecules which is far from the 2200 parts per million concentration of boric acid in the pool. Thus no precipitation such that the pool water would not be able to shut down the reactor is expected. (Freezing could be a cause of boron removal but this could not occur at power.)

The maximum boron concentration in the primary is 375 ppm and in the pool is 2200 ppm. The amount of boric acid that must be removed from the pool water to prevent scram is 42,000 lbs. Such a mass cannot be removed accidentally. Moreover, the return, low boration stream does not pass directly into the reactor pool but goes to the primary loop. If this were done, the reactor would go on a reactivity excursion that would indicate the low

boration.

Loop Flow Blockage

The third way passive scram could fail is the most credible - a flow blockage. Such a blockage could occur any place in the loop circuit but blockage of flow between the upper and lower density lock is very unlikely because of the size of the blockage that would be necessary to block the flow. Blockage of flow through the upper density lock could be the result of a core degrading event. It would not be the initiating event but could prevent decay heat removal and exacerbate the accident. An obstruction that has to pass through the lower density lock, and the core to block the upper density lock does not seem credible - such an obstruction would simply block the lower density lock.

The lower density lock could be blocked by a flexible sheet material that enters the PCRV during refueling when the flow structure is disassembled. It would be carried by the flow, when the pool loop is activated, to cover the density lock.

Flow through the lower density lock could also be prevented by: 1) a gas bubble¹, 2) structural failure e.g. the gas cap falls and blocks the lower density lock, 3) reactor core support structure failure that drops material onto the lower density lock.

D.3 Conclusions

Based on this discussion, Table D-1 is prepared to present the failure probability categories for the scram failure.

Table D-1 Summary of Failure Probabilities Scram

<u>Type of Scram</u>	<u>Failure Probability Class</u>
Active Scram (AS)	III
Manual Scram (MS)	II
Passive Scram (PS)	IV

The estimate for AS is based on considerations of failure probabilities in instrumentation. It is believed that the doubly redundant main computer will not contribute significantly. The MS class is based on human error assessments and the explicitness of the information provided the operators. The assessment for PS is strongly influenced by the HFIR assessment² for loose objects that could block the flow. It is believed that other effects such as low pool water and low boration may be subsumed under this category.

¹ Gas bubble blockage may be dependent on the amount of heat being generated by the core. It may be that shutdown from full power will generate sufficient heat to eject the bubble while a colder core may not. While this is favorable for the initial scram, it might cause problems for long-term cooling.

² D.H. Johnson et al., "The High Flux Isotope Reactor Probabilistic Risk Assessment, Interim Report (I)," PLG-573, August 1987.

APPENDIX E

THERMAL-FLUID FLOW ANALYSIS,

USING THE PIPA CODE

APPENDIX E - THERMAL-FLUID FLOW ANALYSIS, USING THE PIPA CODE

In support of the FMECA and HAZOP studies of Section 4 and 5, a simplified thermal-fluid flow model of the primary loop of the PIUS reactor was set up, using the PC-based PIPA-Code. The model was applied to several basic transients and accident scenarios, to enhance our understanding of the behavior of the reactor under such conditions. This appendix provides a brief description of the model and summarizes the results obtained. It ought to be emphasized, that the PIPA model, as applied to PIUS here, represents a simplified model of the reactor, to be used only for preliminary and scoping evaluations, in support of the above mentioned FMECA and HAZOP studies.

Figure E-1 shows the basic PIUS design. Much of the primary system is contained in a large pool of highly (2000 ppm) borated water. The separation between the lightly (375 ppm) borated primary water and the pool water is maintained at two "density locks" only as long as the system remains near equilibrium operating conditions. Because the density locks are always open, any significant change in the pressure balance will generally result in the ingress of highly borated pool water into the reactor core, thus providing a passive shutdown mechanism.

For the PIUS 600 design being evaluated by the U.S. NRC, the steam generators and main coolant pumps are outside the pool region. Consequently, a pipe break can cause a partial loss of inventory. For this reason, the PIUS 600 design includes a "siphon breaker" which links the cold leg and the pressurizer, and, thereby, assures that vapor will interrupt the siphon effect so that most of the pool inventory remains.

E.1 Description of PIPA Model

The PC-based model was set up, to simulate the response of the PIUS primary system to various postulated scenarios. The model is called the PIUS Interactive Plant Analyzer (PIPA) code. It includes the reactor core, the primary coolant system with steam generators and coolant pumps, the pressurizer, the lower and upper density locks, the large borated pool, and the control and plant protection systems, as illustrated in Figure E-2. The important physical processes modeled include the two-phase flow thermal-hydraulics, neutron kinetics, and boron transport.

Dynamics of the Two-Phase Flow Coolant

Two-phase flow conditions must be modeled because PIUS is designed to operate at 9 MPa, which is not far from 7 MPa at which bulk boiling will occur at normal operating temperatures. Thus, the dynamics of the coolant are modeled via the four-equation drift flux model (Cheng et al, 1986, Ishii 1977). To reduce the numerical stiffness that results when pressure waves are tracked locally, an integrated form of the mixture momentum equation is used, integrating along each closed flow loop.

There are four main coolant loops in the PIUS 600 design. In addition, there are two pool loops: one between the lower and upper density locks and the other between the lower density locks and the pressurizer pool (Figure E-2). Thus, with a two-channel representation for the reactor core, there are a total of seven flow loops that must be modeled for PIUS. To accommodate the model on a PC, we lumped the four coolant loops into one, reducing the total number of flow loops to four.

The fluid properties of the two phases are broken into two void regimes: (i) For void fraction below 0.7, the vapor phase is assumed to be at saturation, and the liquid phase may be sub-cooled, saturated, or super-heated; and (ii) For void fraction above 0.7, the liquid phase is assumed to be at saturation, and the vapor phase may be sub-cooled, saturated, or super-heated. The NBS (National Bureau of Standards) water/steam property library (Haar et al, 1984) was used to generate the property function tables for the simulation via high-order polynomial interpolation (choice of order 1 to 10).

Density Locks

In the PIUS 600 design, there are lower and upper density locks whose functions are essential to its inherent safety. PIPA represents the two density locks as two separate computational cells, as shown in Figure E-2. They are treated the same as the other computational cells in regard to the conservation of mass, energy, momentum, and boron. However, the hot/cold interface level in the two density locks are tracked via the volumetric flows at the interfaces, which are determined from the volumetric flow rates at the cell boundaries of the two density locks. These flow rates are obtained from the volumetric flux distribution, as described above.

The hot/cold interface levels then are used to determine the average fluid properties of the density locks from those of the neighboring cells.

Boron Transport

The modeling of boron transport is very important because it is the sole shutdown mechanism for PIUS. Boron tracking is done by solving the boron mass balance equation in each of the computational cells depicted in Figure E-2. The model includes a flow-dependent boron mixing efficiency, which takes into account the degradation of boron mixing and the potential stratification of boron as liquid flow decreases.

Thermal Conduction in Fuel Elements

The transient heat conduction in the fuel is modelled, using a radially integrated mean fuel element temperature, evaluated by explicit integration for each axial node (Wulff et al, 1984).

Neutron Kinetics

The neutron kinetics are prescribed by the conventional point kinetics equations. The effects of thermal-hydraulic feedback and boron come into play in the point kinetics equations via the total reactivity as a function of time, where the total reactivity is represented by the sum of the moderator temperature reactivity, Doppler reactivity, void reactivity, boron reactivity, and accident reactivity (a user-defined external reactivity function).

Limitations of PIPA Model

The major limitations of the PIPA model, in its current form, are the following:

1. The flow and boron concentration model is one dimensional.
2. The acoustic effects are not accounted for, therefore, the model is not applicable to rapid depressurization events such as a large break LOCA.
3. The use of point kinetics ignores the spatial effect of time-dependent neutron flux distribution.
4. The secondary side system or balance of plant (BOP) is not modelled, but its effects are accounted for via boundary conditions.
5. The siphon breaker is not modelled because it is only important in the event of a large break LOCA on a cold leg, for which the PIPA code is not applicable (see Point 2, above).

E.2 Simulations of PIUS System Transients

Full Scram - One Pump Trip

In PIUS, there are no control rods and the emergency shutdown is accomplished by ingress of highly borated pool water via natural circulation by tripping one of the four main coolant pumps. This process is proposed to be equivalent to a full scram in conventional LWRs. Thus, analysis of the full scram in PIUS is of fundamental importance to safety. The PC-model PIPA was used to simulate the full scram of the PIUS 600 design.

In accordance with the sequence of events described in PSID, the transient analysis was initiated at $t=0$ from hot full power conditions, by tripping one coolant pump, followed by a feed-water trip at 18 seconds. Figure E-3 shows the average relative speed of the four pumps, which coast down from 100% to 73% in 10 seconds. As a consequence, the core and hot leg flow-rates decrease as shown in Figure E-4, which break the pressure balance in the pool loop and activate the natural circulation inflow of the highly borated pool water, as evidenced in the responses of the lower and upper density locks (Figure E-5). The resulting ingress of highly concentrated boron is demonstrated in the average boron concentration in the core (Figure E-6). This boron ingress eventually shuts down the reactor after about 80 seconds, as shown in the reactor power response in Figure E-7. Note that there is a power spike of 86% at 20 sec. The corresponding behavior with time of various reactivities are presented in Figure E-8.

The oscillatory behavior of this transient is the consequence of boron transport phenomena, with a period of approximately 20 seconds. The boron concentration in the core increases every time the highly borated pool water passes through the core in cycles, and decreases when the lightly borated primary coolant loop water (which has a five times higher flow rate than the pool loop water) enters the core between the flow cycles of the pool loop (See Figure E-6). In addition, the strongly negative moderator temperature coefficient contributes to the oscillations, as the cooler water pushes the power up and hotter water reduces the power.

The responses of the primary system to this transient are in reasonably good agreement with the vendor's analyses, as given in the PSID. The responses are shown in Figure E-9 for reactor power, in Figure E-10 for boron concentration, in Figure E-11 for core flow, and in Figure E-12 for the lower density lock flow.

The above calculation was done assuming perfect boron mixing. The flow-dependent boron mixing model, with a prescribed characteristic curve, was also used to simulate the full scram. The model was equivalent to a boron transport delay of a few seconds. This model predicted an initial power spike of 118% at 4.5 seconds, followed by a larger spike of 129% at 21 seconds, as shown in Figure E-13. However, it still led to a hot shutdown within 2 minutes. Since many key parameters are still BNL estimates, the exact timing and size of these power spikes are not well known.

Both the pump inertia and the efficiency of boron mixing have significant effects on the outcome of this transient. However, in all cases, there was a hot shutdown within two minutes, thus demonstrating the success of the passive shutdown feature of the PIUS 600 design.

Loss of Heat Sink

A feed-water trip or a steam line break generally leads to a loss of heat sink event. Such an event was analyzed with PIPA, by specifying a time-dependent heat sink function as a boundary condition, which is a linear ramp-down of the heat sink across the steam generator in 0.2 second (a bounding case).

The simulated results indicate that the response of the primary system to such an event is the over-pressurization of the primary system (Figure E-14). The system pressure reached 9.25 MPa at about 17 seconds into the transient, started to decrease afterward, then settled down at the initial pressure of 9 MPa, due to the pressure control system trying to maintain the system's pressure. The control system reduces the pump speed to 80% of its initial value,

as shown in Figure E-15, in response to the increasing pressure and the downward movement of the hot/cold interface in lower density lock (See Figure E-16). This movement helped reduce the reactor power, as shown in Figure E-17, as the highly borated pool water enters the reactor core (Figure E-18). The corresponding response boron concentration is presented in Figure E-19. The reactor was nearly in hot shutdown at the end of 100 seconds.

Uncontrolled Boron Dilution

An uncontrolled boron dilution event was simulated via a continuous injection of clean water at the rate of 60 kg/s on the hot leg as a result of failure of the reactivity control system and the inadvertent start-up of the backup distilled water pump. (There are two parallel pumps for the distilled water system, each with 30 kg/s flow capacity). As a multiple failure event, an additional scram failure (due to a total blockage of the density locks) was assumed.

The immediate consequence of the distilled water injection was a reduction of boron concentration in the core (Figure E-20). As a result, the reactor power increased steadily, after a few initial small-amplitude oscillations, and reached 105% at 250 seconds (Figure E-21). The power increase was limited by the negative moderator temperature feedback and Doppler feedback, as shown in Figure E-22. Because of the postulated scram failure, the pump speed remained essentially constant during the transient (Figure E-23). The pressure experienced small oscillations in the first 100 seconds, (Figure E-24), then stabilized at the initial pressure of 9 MPa due to the pressure control system.

The increase in reactor power is expected to continue and would reach the scram set-point of 120%, at which a pump trip would occur, thereby reducing the core flow to help limit the power increase. No serious consequences are expected from this multiple failure event.

Inadvertent Opening of Safety/Relief Valves (IORV)

To test the PIPA model for moderate depressurization transients such as an IORV, we simulated a stuck open SRV of the largest flow area. This is equivalent to a small break LOCA. In this simulation, the normal scram function (via pump trip) was assumed.

Figure E-25 shows pressure response of the primary system to the IORV. The pressure decreased steadily in the first 25 seconds and settled down to about 8.5 MPa afterward due to the steam injection system and the pressure control system. The initial reduction of the system pressure created pressure imbalance in the pool loop and caused the pump speed controller to reduce the pump's speed to about 79% of its initial value, as shown in Figure E-26. This led to a reduction in the primary loop, as shown in Figure E-27, and resulted in the natural circulation inflow of the highly borated pool water (Figure E-28). The corresponding boron ingress into the core is evident in Figure E-29. The reactor power decreased rapidly after the boron ingress, as shown in Figure E-30, although there were a few power spikes of small amplitude. The time behavior of feedback reactivities are presented in Figure E-31. The reactor was in hot shutdown after 125 seconds and there is no serious consequence for this event.

E.3 References

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Figure E-1 Plant Schematic of PIUS 600 Design

CR1=Core Channel 1	CEN1=Core Entrance 1	LPL=Lower Plenum
CR2=Core Channel 2	CEN2=Core Entrance 2	UPL=Upper Plenum
UDC=Upper Downcomer	UPU=Lower Riser	LPO=Lower Pool
LDC=Lower Downcomer	RSR=Upper Riser	UPO=Upper Pool
UDL=Upper Density Lock	SGR=Steam Generator	SGI=SG Inlet Plenum
LDL=Lower Density Lock	PZR=Pressurizer Funnel	SGE=SG Exit Plenum
SRV=Safety/Relief Valve	PZP=Pressurizer Pool	UHD=Steam Dome
ORS=Outer Riser		

Figure E-2 PIPA Representation of PIUS Primary System

PIUS Full Scram at Hot Full Power One Pump Trip

Figure E-3 Average Pump Speed Response to a PIUS Full Scram

PIUS Full Scram at Hot Full Power One Pump Trip

Figure E-4 Core Inlet and Hot Leg Inlet Flow Response to a PIUS Full Scram

PIUS Full Scram at Hot Full Power One Pump Trip

Figure E-5 Lower and Upper Density Lock Flow Response to a PIUS Full Scram
(Lower Density Lock Flow Positive to Core;
Upper Density Lock Flow Positive from Core)

PIUS Full Scram at Hot Full Power One Pump Trip

Figure E-6 Core Average Boron Concentration Response to a PIUS Full Scram

PIUS Full Scram at Hot Full Power One Pump Trip

Figure E-7 Reactor Power Response to a PIUS Full Scram

PIUS Full Scram at Hot Full Power One Pump Trip

Figure E-8 Reactivity Behavior during a PIUS Full Scram

PIUS Full Scram at Hot Full Power PIPA (BNL) vs. RIGEL (ABB Atom)

Figure E-9 PIPA vs. RIGEL (ABB Atom) for Reactor Power During a Full Scram

PIUS Full Scram at Hot Full Power PIPA (BNL) vs. RIGEL (ABB Atom)

Figure E-10 PIPA vs. RIGEL for Boron Concentration During a Full Scram

PIUS Full Scram at Hot Full Power PIPA (BNL) vs. RIGEL (ABB Atom)

Figure E-11 PIPA vs. RIGEL for Core Flow During a Full Scram

PIUS Full Scram at Hot Full Power PIPA (BNL) vs. RIGEL (ABB Atom)

Figure E-12 PIPA vs. RIGEL for Lower Density Lock Flow During a Full Scram

PIUS Full Scram at Hot Full Power Effect of Boron Mixing

Figure E-13 Effect of Boron Mixing on Reactor Power During a Full Scram

Loss of Heat Sink
Time Constant = 0.2 sec, IBM RISC 6000

Figure E-14 Primary System Pressure Response to a Loss of Heat Sink Event

Loss of Heat Sink
Time Constant = 0.2 sec, IBM RISC 6000

Figure E-15 Pump Speed Response to a Loss of Heat Sink Event

Loss of Heat Sink

Time Constant = 0.2 sec, IBM RISC 6000

Figure E-16 Lower Density Lock Level Response to a Loss of Heat Sink Event

Loss of Heat Sink
Time Constant = 0.2 sec, IBM RISC 6000

Figure E-17 Reactor Power Response to a Loss of Heat Sink Event

Loss of Heat Sink

Time Constant = 0.2 sec, IBM RISC 6000

Figure E-18 Density Locks Flow Response to a Loss of Heat Sink Event
(Lower Density Lock Flow Positive to Core;
Upper Density Lock Flow Positive from Core)

Loss of Heat Sink
Time Constant = 0.2 sec, IBM RISC 6000

Figure E-19 Boron Concentration Response to a Loss of Heat Sink Event

Uncontrolled Boron Dilution With Postulated Scram Failure, 60 kg/s

Figure E-20 Boron Concentration Response to an Uncontrolled Boron Dilution Event

Uncontrolled Boron Dilution With Postulated Scram Failure, 60 kg/s

Figure E-21 Reactor Power Response to an Uncontrolled Boron Dilution Event

Uncontrolled Boron Dilution With Postulated Scram Failure, 60 kg/s

Figure E-22 Reactivity Behavior During an Uncontrolled Boron Dilution

Uncontrolled Boron Dilution With Postulated Scram Failure, 60 kg/s

Figure E-23 Pump Speed Response to an Uncontrolled Boron Dilution

Uncontrolled Boron Dilution With Postulated Scram Failure, 60 kg/s

Figure E-24 Primary System Pressure Response to an Uncontrolled Boron Dilution Event

Inadvertent Opening of SRV 1st SRV Stuck Open

Figure E-25 Primary System Pressure Response to an IORV Event

Inadvertent Opening of SRV 1st SRV Stuck Open

Figure E-26 Pump Speed Response to an IORV Event

Inadvertent Opening of SRV 1st SRV Stuck Open

Figure E-27 Primary Loop Flow Response to an IORV Event

Inadvertent Opening of SRV 1st SRV Stuck Open

Figure E-28 Density Locks Flow Response to an IORV Event
(Lower Density Lock Flow Positive to Core;
Upper Density Lock Flow Positive from Core)

Inadvertent Opening of SRV 1st SRV Stuck Open

Figure E-29 Boron Concentration Response to an IORV Event

Inadvertent Opening of SRV 1st SRV Stuck Open

Figure E-30 Reactor Power Response to an IORV Event

Inadvertent Opening of SRV 1st SRV Stuck Open

Figure E-31 Reactivity Behavior During an IORV Event

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