FERMI 2 - 91-150

POWER UPRATE SAFETY ANALYSIS

Approved by: _____
M. K. Deora
Detroit Edison Co.

Approved by: _____
G. L. Sozzi
GE Nuclear Energy

Approved by: _____
D. J. Robare
GE Nuclear Energy

Approved by: _____
T. B. Madden
Stone & Webster Engineering Corp.

September 1991

9212100033 920622
PDR FOIA
GUNTER91-579 PDR

PROPRIETARY INFORMATION NOTICE

Parts of this document contain proprietary information of the General Electric Company (GE) and are furnished to Detroit Edison Co. in confidence solely for the purpose or purposes stated in the transmittal letter. No other use, direct or indirect, of the document or the information it contains is authorized. The recipient shall not publish or otherwise disclose it or the information to others without the consent of GE, and shall return the document at the request of GE.

GE proprietary information is noted by "GE Proprietary Information" on those pages so designated. Non-proprietary information is noted by "bars" drawn in the margin of the text of this report or is on a page not designated proprietary.

The technical bases for the information provided by GE in this report is contained in DRF No. A00-03770.

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

The only undertakings of the General Electric Company (GE) respecting information in this document are contained in the contract between Detroit Edison Co. and GE, Plant Services Contract, effective December 15, 1989, as amended to the date of transmittal of this document, and nothing contained in this document shall be construed as changing the contract. The use of this information by anyone other than Detroit Edison Co., or for any purpose other than that for which it is intended, is not authorized; and with respect to any unauthorized use, GE makes no representation or warranty, express or implied, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document, or that its use may not infringe privately owned rights.

TABLE OF CONTENTS

Section	Title	Page
EXECUTIVE SUMMARY		i
1.0 OVERVIEW		1-1
1.1 INTRODUCTION		1-1
1.2 PURPOSE AND APPROACH		1-1
1.3 UPRATED PLANT OPERATING CONDITIONS		1-4
1.3.1 Reactor Heat Balance		1-4
1.3.2 Reactor Performance Improvement Features		1-5
1.4 SUMMARY AND CONCLUSIONS		1-5
1.5 REFERENCES		1-6
2.0 REACTOR CORE AND FUEL PERFORMANCE		2-1
2.1 FUEL DESIGN AND OPERATION		2-1
2.1.1 Lead Fuel Assemblies		2-1
2.2 THERMAL LIMITS ASSESSMENT		2-2
2.2.1 Minimum Critical Power Ratio Operating Limit		2-2
2.2.2 Maximum Average Planar Linear Heat Generation Rate and Maximum Linear Heat Generation Rate Operating Limits		2-2
2.3 REACTIVITY CHARACTERISTICS		2-2
2.3.1 Power/Flow Operating Map		2-3
2.4 STABILITY		2-3
2.5 REACTIVITY CONTROL		2-3
2.5.1 Control Rod Drive Hydraulic System		2-3
2.6 REFERENCES		2-4
3.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS		3-1
3.1 NUCLEAR SYSTEM PRESSURE RELIEF		3-1

TABLE OF CONTENTS (Cont'd)

Section	Title	Page
3.2	REACTOR OVERPRESSURE PROTECTION	3-1
3.3	REACTOR VESSEL AND INTERNALS	3-1
3.3.1	Reactor Vessel Fracture Toughness	3-2
3.3.2	Reactor Internals and Pressure Differentials	3-2
3.3.3	Reactor Vessel Integrity	3-3
3.4	REACTOR RECIRCULATION SYSTEM	3-3
3.5	REACTOR COOLANT PIPING	3-4
3.6	MAIN STEAMLINE FLOW RESTRICTORS	3-5
3.7	MAIN STEAM ISOLATION VALVES	3-5
3.8	REACTOR CORE ISOLATION COOLING	3-5
3.9	RESIDUAL HEAT REMOVAL SYSTEM	3-5
3.10	REACTOR WATER CLEANUP SYSTEM	3-6
3.11	MAIN STEAM, FEEDWATER, AND BALANCE-OF-PLANT PIPING	3-8
3.11.1.	Pipe Stress Evaluation	3-8
3.11.2	Pipe Support Evaluation	3-8
3.12	REFERENCES	3-9
4.0	**ENGINEERED SAFETY FEATURES**	4-1
4.1	CONTAINMENT SYSTEM PERFORMANCE	4-1
4.1.1	Containment Pressure and Temperature Response	4-1
4.1.1.1	Long-Term Suppression Pool Temperature Response	4-2
4.1.1.2	Containment Gas Temperature Response	4-2
4.1.1.3	Short-Term Containment Pressure Response	4-3
4.1.2	Containment Dynamic Loads	4-3
4.1.2.1	LOCA Containment Dynamic Loads	4-3
4.1.2.2	Safety/Relief Valve Containment Dynamic Loads	4-3
4.1.2.3	Subcompartment Pressurization	4-4
4.1.3	Containment Isolation	4-4

TABLE OF CONTENTS (Cont'd)

Section	Title	Page
4.2	EMERGENCY CORE COOLING SYSTEMS (ECCS)	4-4
4.2.1	High Pressure ECCS (High Pressure Coolant Injection)	4-5
4.2.2	RHR System (Low Pressure Coolant Injection)	4-5
4.2.3	Low Pressure Core Spray System	4-5
4.2.4	Automatic Depressurization System	4-5
4.3	EMERGENCY CORE COOLING SYSTEM PERFORMANCE EVALUATION	4-5
4.4	STANDBY GAS TREATMENT SYSTEM	4-6
4.5	OTHER ESF SYSTEMS	4-6
4.5.1	Main Steam Isolation Valve Leakage Control System	4-6
4.5.2	Post-LOCA Combustible Gas Control	4-6
4.5.3	Emergency Cooling Water System	4-6
4.5.4	Emergency Core Cooling Auxiliary Systems	4-6
4.5.5	Main Control Room Atmosphere Control System	4-7
4.5.6	Standby Power System	4-7
4.5.7	Main Steamline Flow Restrictors	4-7
4.5.8	CRD Velocity Limiters	4-7
4.6	REFERENCES	4-8
5.0	INSTRUMENTATION AND CONTROL	5-1
5.1	NUCLEAR STEAM SUPPLY SYSTEM	5-1
5.1.1	Control Systems Evaluation	5-1
5.1.1.1	Neutron Monitoring System	5-1
5.1.2	Instrument Setpoints	5-2
5.1.2.1	RPV High-Pressure Scram	5-2
5.1.2.2	High-Pressure Recirculation Pump Trip	5-2
5.1.2.3	Pressure Regulator	5-3
5.1.2.4	Safety/Relief Valve	5-3
5.1.2.5	Neutron Monitoring System	5-3
5.1.2.6	Main Steam High Flow Isolation	5-3
5.1.2.7	Main Steamline High Radiation Scram	5-3
5.1.2.8	Turbine Stop Valve Closure and Turbine Control Valve Fast Closure Scram Bypass	5-4

TABLE OF CONTENTS (Cont'd)

Section	Title	Page
5.2	BALANCE-OF-PLANT - POWER CONVERSION AND AUXILIARY SYSTEMS	5-5
5.2.1	Control Systems Evaluation	5-5
5.2.1.1	Pressure Control System	5-5
5.2.1.2	Turbine Control System	5-5
5.3	REFERENCES	5-6
6.0	**ELECTRICAL POWER AND AUXILIARY SYSTEMS**	6-1
6.1	AC POWER	6-1
6.1.1	Offsite Power System	6-1
6.1.2	Onsite Power Distribution System	6-1
6.2	DC POWER	6-2
6.3	FUEL POOL COOLING	6-2
6.4	WATER SYSTEMS	6-3
6.4.1	Service Water Systems	6-3
6.4.1.1	Safety-Related Loads	6-3
6.4.1.1.1	Emergency Equipment Service Water System	6-3
6.4.1.1.2	Diesel Generator Service Water System	6-4
6.4.1.1.3	Residual Heat Removal Service Water System	6-4
6.4.1.2	Nonsafety-Related Loads	6-4
6.4.2	Main Condenser/Circulating Water/Cooling Tower Performance	6-5
6.4.2.1	Discharge Limits	6-5
6.4.3	Reactor Building Closed Cooling Water System	6-5
6.4.4	Turbine Building Closed Cooling Water System	6-5
6.4.5	Ultimate Heat Sink	6-6
6.5	STANDBY LIQUID CONTROL SYSTEM	6-7
6.6	POWER DEPENDENT HVAC	6-8
6.7	FIRE PROTECTION	6-8
6.8	SYSTEMS NOT IMPACTED BY POWER UPRATE	6-9
6.8.1	Systems With No Impact	6-9

TABLE OF CONTENTS (Cont'd)

Section	Title	Page
6.8.2	Systems With Insignificant Impact	6-10
6.9	REFERENCES	6-11
7.0	**POWER CONVERSION SYSTEMS**	7-1
7.1	TURBINE/GENERATOR	7-1
7.2	CONDENSER AND STEAM JET AIR EJECTORS	7-1
7.3	TURBINE STEAM BYPASS	7-1
7.4	FEEDWATER AND CONDENSATE SYSTEMS	7-2
7.4.1	Condensate Demineralizers	7-3
7.4.2	Standby Feedwater System	7-3
8.0	**RADWASTE SYSTEMS AND RADIATION SOURCES**	8-1
8.1	LIQUID WASTE MANAGEMENT	8-1
8.2	GASEOUS WASTE MANAGEMENT	8-1
8.2.1	Offgas System	8-2
8.3	RADIATION SOURCES IN THE REACTOR CORE	8-2
8.3.1	Operation	8-2
8.3.2	Post-Operation	8-2
8.4	RADIATION SOURCES IN THE COOLANT	8-3
8.4.1	Coolant Activation Products	8-3
8.4.2	Activated Corrosion Products	8-3
8.4.3	Fission Products	8-3
8.5	RADIATION LEVELS	8-4
8.5.1	Normal Operation	8-4
8.5.2	Post-Operation/Accident	8-4
8.5.3	Offsite Doses (Normal Operation)	8-4

TABLE OF CONTENTS (Cont'd)

Section	Title	Page
9.0	**REACTOR SAFETY PERFORMANCE EVALUATIONS**	9-1
9.1	REACTOR TRANSIENTS	9-1
9.2	DESIGN BASIS ACCIDENTS	9-2
9.2.1	Other Accidents Addressed In USAR	9-2
9.3	SPECIAL EVENTS	9-2
9.3.1	Anticipated Transients Without Scram (ATWS)	9-2
9.3.2	Station Blackout (SBO)	9-3
9.4	REFERENCES	9-4
10.0	**ADDITIONAL ASPECTS OF POWER UPRATE**	10-1
10.1	HIGH-ENERGY LINE BREAK	10-1
10.1.1	Temperature, Pressure, and Humidity Profiles	10-1
10.1.1.1	Main Steam System Line Break	10-1
10.1.1.2	Feedwater System Line Break	10-1
10.1.1.3	High Pressure Coolant Injection System Line Break	10-2
10.1.1.4	Reactor Core Isolation Cooling System Line Break	10-2
10.1.1.5	Reactor Water Cleanup System Line Break	10-2
10.1.2	Pipe Whip and Jet Impingement	10-2
10.1.3	Moderate Energy Line Crack	10-2
10.2	EQUIPMENT QUALIFICATION	10-3
10.2.1	Equipment Qualification of Electrical Equipment	10-3
10.2.1.1	Inside Containment	10-3
10.2.1.2	Outside Containment	10-3
10.2.2	EQ of Mechanical Equipment with Non-Metallic Components	10-4
10.2.3	Mechanical Component Design Qualification	10-4
10.3	REQUIRED TESTING	10-5
10.4	REFERENCES	10-6

TABLE OF CONTENTS (Cont'd)

Section	Title	Page
11.0	**LICENSING EVALUATIONS**	11-1
11.1	EVALUATION OF OTHER APPLICABLE LICENSING REQUIREMENTS	11-1
11.1.1	Generic Communications	11-1
11.1.2	Plant Unique Items	11-1
11.1.2.1	Deviation Event Reports	11-2
11.1.2.2	Regulatory Action Commitment Tracking	11-2
11.1.2.3	Safety Evaluations	11-2
11.1.2.4	Temporary Modifications	11-2
11.1.3	Emergency Operating Procedures	11-3
11.2	IMPACT ON TECHNICAL SPECIFICATIONS	11-3
11.3	ENVIRONMENTAL ASSESSMENT	11-3
11.4	SIGNIFICANT HAZARDS CONSIDERATION ASSESSMENT	11-4
11.4.1	Introduction	11-4
11.4.2	Discussion of Issues Being Evaluated	11-5
11.4.3	Assessment Against 10CFR50.92 Criteria	11-13
11.5	REFERENCES	11-18

LIST OF TABLES

Number and Title		Page
Table 1-1	GLOSSARY OF TERMS	1-7
Table 1-2	CURRENT AND UPRATED PLANT OPERATING CONDITIONS	1-11
Table 3-1	OVERPRESSURIZATION EVENT ANALYSIS	3-11
Table 3-2	REACTOR INTERNALS AND PRESSURE DIFFERENTIAL ANALYSIS	3-12
Table 4-1	CONTAINMENT PERFORMANCE RESULTS	4-9
Table 4-2	ECCS PERFORMANCE RESULTS	4-10
Table 5-1	ANALYTICAL LIMITS FOR SETPOINTS	5-6
Table 6-1	FUEL POOL COOLING ANALYSIS	6-13
Table 9-1	PARAMETERS USED FOR TRANSIENT ANALYSIS	9-5
Table 9-2	TRANSIENT ANALYSIS RESULTS FOR POWER UPRATE	9-6
Table 9-3	LOSS-OF-COOLANT ACCIDENT RADIOLOGICAL CONSEQUENCES	9-7
Table 9-4	MAIN STEAMLINE BREAK ACCIDENT RADIOLOGICAL CONSEQUENCES	9-8
Table 10-1	HIGH-ENERGY LINE BREAK	10-7
Table 11-1	TECHNICAL SPECIFICATIONS AFFECTED BY POWER UPRATE	11-21

LIST OF FIGURES

Number and Title

Figure 1-1 UPRATED HEAT BALANCE (NOMINAL) 100% POWER/100% CORE FLOW/1045 PSIA DOME PRESSURE

Figure 1-2 UPRATED HEAT BALANCE (NOMINAL) 100% POWER/105% CORE FLOW (ICF)/1045 PSIA DOME PRESSURE

Figure 2-1 POWER/FLOW OPERATING MAP FOR POWER UPRATE

Figure 3-1 RESPONSE TO MSIV CLOSURE WITH FLUX SCRAM

Figure 3-2 MINIMUM REACTOR PRESSURE VESSEL METAL TEMPERATURE VERSUS REACTOR VESSEL PRESSURE

Figure 6-1 SODIUM PENTABORATE SOLUTION VOLUME CONCENTRATION REQUIREMENTS

Figure 9-1 TURBINE/GENERATOR TRIP WITH BYPASS FAILURE - 100% UPRATED POWER/105% CORE FLOW

Figure 9-2 TURBINE/GENERATOR TRIP WITH BYPASS AND MOISTURE SEPARATOR REHEATER FAILURE -100% UPRATED POWER/105% CORE FLOW

Figure 9-3 FEEDWATER CONTROLLER FAILURE - MAXIMUM DEMAND - 100% UPRATED POWER/105% CORE FLOW/374.5°F FEEDWATER TEMPERATURE

Figure 9-4 FEEDWATER CONTROLLER FAILURE - MAXIMUM DEMAND W/BYPASS FAILURE/100% UPRATED POWER/105% CORE FLOW/374.5°F FEEDWATER TEMPERATURE

Figure 9-5 FEEDWATER CONTROLLER FAILURE - MAXIMUM DEMAND W/BYPASS AND MOISTURE SEPARATOR REHEATER FAILURE 100% UPRATED POWER/105% CORE FLOW/374.5°F FEEDWATER TEMPERATURE

Figure 11-1 RADIAL POWER PROFILE SCHEMATIC

Figure 11-2 PCT MARGIN SCHEMATIC

Figure 11-3 TRANSIENT MARGIN SCHEMATIC

EXECUTIVE SUMMARY

Pursuant to 10CFR50.90, Detroit Edison Company hereby proposes to amend Operating License NPF-43 for the Fermi 2 plant by changing the plant Operating License and Technical Specifications. The proposed changes will allow Fermi 2 to increase power level from 3293 MWt to 3430 MWt, a 4.2% increase in thermal power and a 5% increase in steam flow. This licensing report presents the results of the analyses and evaluations that support operation with the increase in power and steam flow. The original design power rating for Fermi 2 was 3430 Mwt and 105% steam flow; a number of transients and accidents were evaluated at that power level.

GE Nuclear Energy has published generic guidelines and evaluations for BWR power plant uprates in References 1 and 2. This document follows the generic format and addresses those issues that should be included in a Fermi 2-specific power uprate licensing report. For those issues that are evaluated on a generic basis, wherever applicable, this report refers to one of the two references.

Fermi 2-specific analyses and evaluations were performed, consistent with the generic guidelines, for systems and components that might be affected to ensure their capability to support the increase in power output and steam flow. Since data is described in detail in the USAR, the descriptive material is referenced, as appropriate. The analyses and evaluations resulted in determinations that the systems and components were either not affected by power uprate or had sufficient design capacity to accommodate uprate conditions.

In addition to the above, the effect of uprate on the environment was assessed to verify that operation of Fermi 2 at uprated power was environmentally acceptable with established NRC requirements and that consistency was maintained with Federal, State, and local regulations. As a result, no changes to the Environmental Protection Plan or to any of the non-NRC permits are required.

Continuing improvements in the analytical techniques based on more realistic assumptions, plant performance feedback, and the latest fuel designs have resulted in a significant increase in the difference between calculated safety analysis results and licensing limits. This margin, combined with the as-built systems and component capacity, provides Fermi 2 with the capability to operate at uprated power and steam flow without major modifications to the plant hardware.

The results of the analyses and evaluations contained herein show that operation of Fermi 2 at the proposed uprated power and steam flow can be accommodated with no significant hazards as set forth in 10CFR50.92 because the proposed changes will not:

(a) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(b) Create the possibility of a new or different kind of accident previously evaluated; or

(c) Create a significant reduction in a margin of safety.

This proposed amendment to the Fermi 2 Operating License meets the criteria of 10CFR51.22(c)(9) for a categorical exclusion from the requirements for an environmental impact statement since it does not:

(a) Involve a significant hazards consideration, or

(b) Result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or

(c) Result in a significant increase in individual or cumulative occupational radiation exposure.

REFERENCES

1. GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31897P-1, Class III (Proprietary), June 1991.

2. GE Nuclear Energy, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31984P, Class III (Proprietary), July 1991.

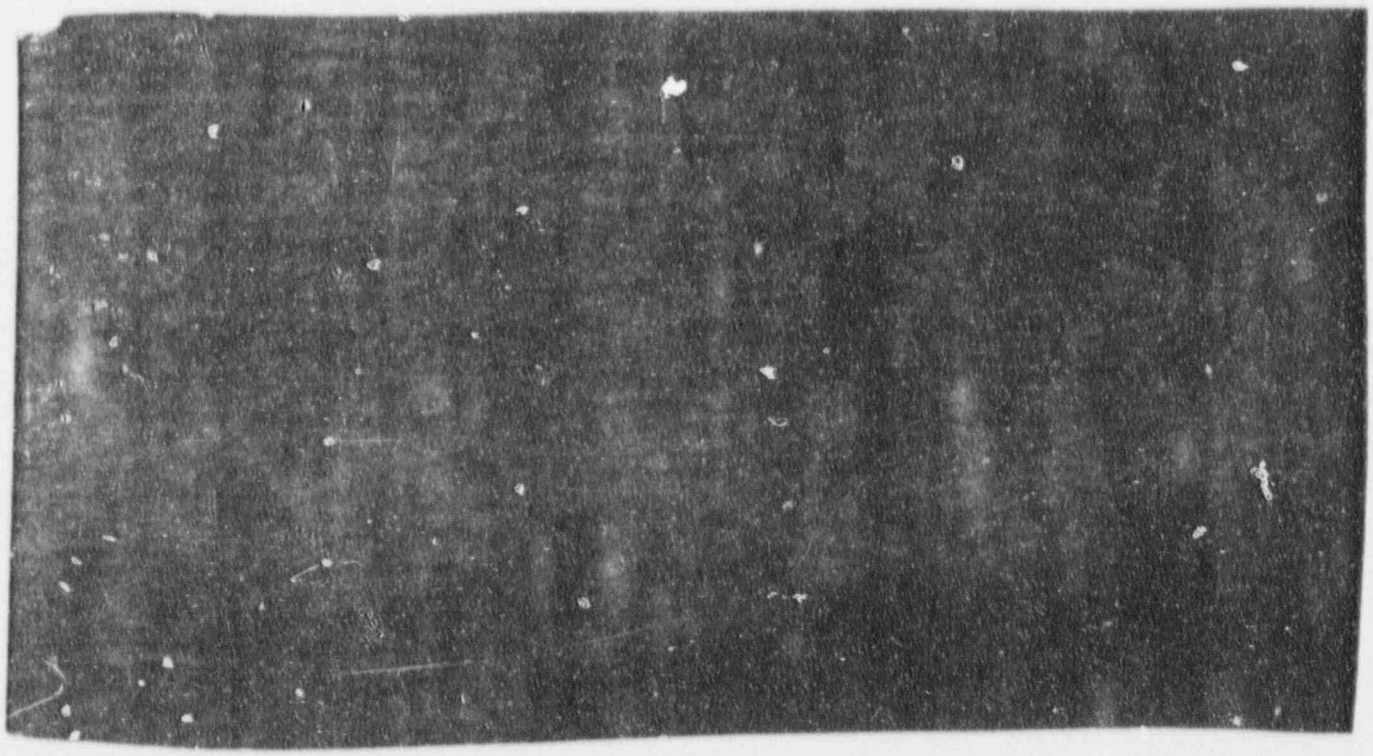
1.0 OVERVIEW

1.1 INTRODUCTION

The original design power rating for Fermi 2 was 3430 MWt and 105% steam flow. Several transient and accident analyses, including overpressurization, emergency core cooling/loss-of-coolant accident, and the radiological consequences of the design basis loss-of-coolant and main steamline break accidents were analyzed at the original design power rating. In addition, the radiological effluents resulting from normal plant operation were evaluated at 3430 MWt. Thus for the current power uprate program, the increase is only the 2% uncertainty factor added for conservatism.

A glossary of terms is provided in Table 1-1.

1.2 PURPOSE AND APPROACH



[redacted]

Uprate Analysis Basis

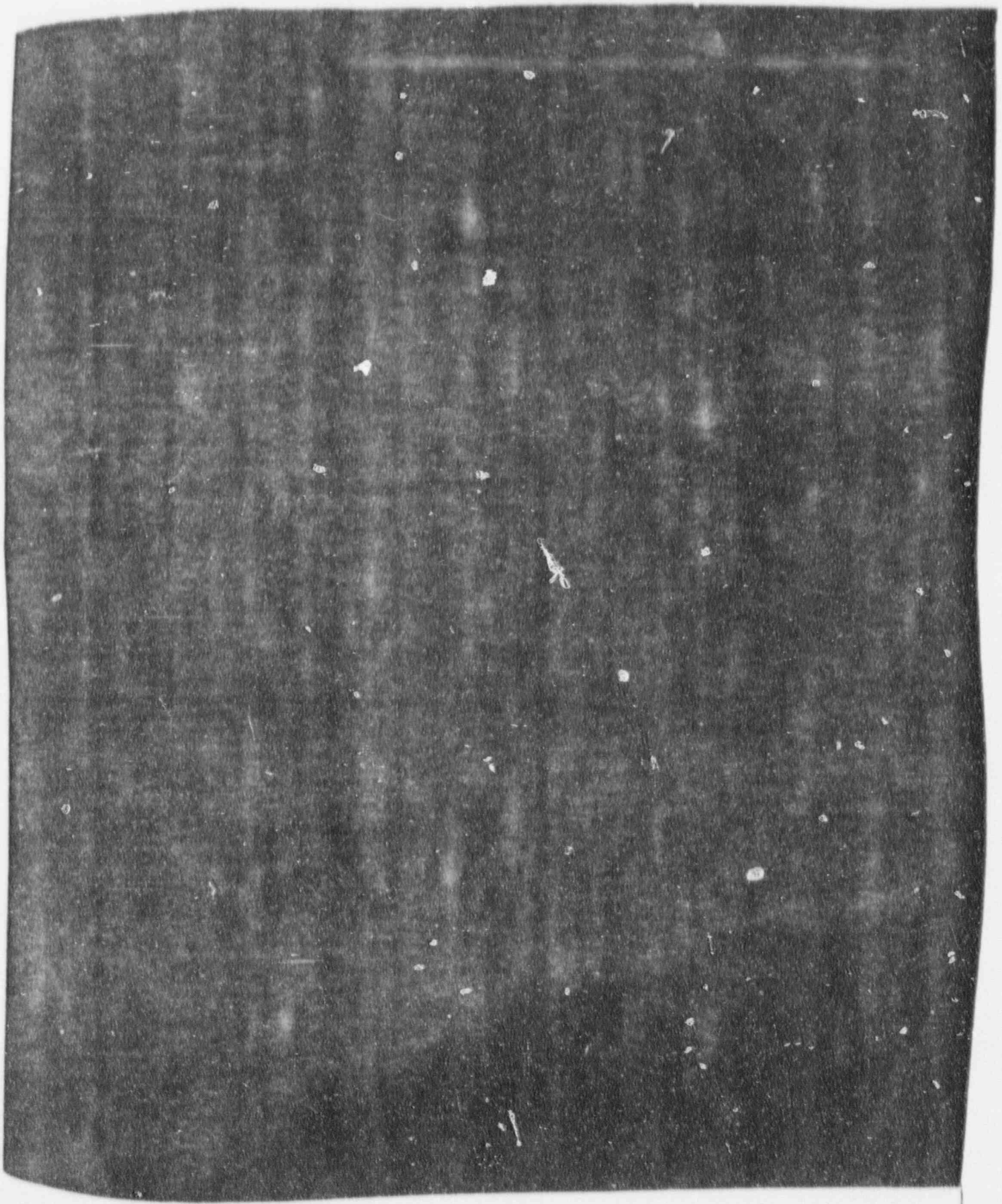
The plant is currently licensed at 3293 MWt (100%) power. The original safety analysis basis was that the reactor had been operating continuously at a power level at least 1.02 times the licensed power level; many of the original analyses had been performed at 105% steam flow. The uprated power level included in this evaluation is that power level associated with 105% steam flow which is about 104.2% power. Therefore, as described herein, the plant has been reanalyzed at a power level of at least 1.02 times the uprated power rating which corresponds to 1.063 times the current licensed power level.

[redacted]

Margin

[redacted]

[redacted]

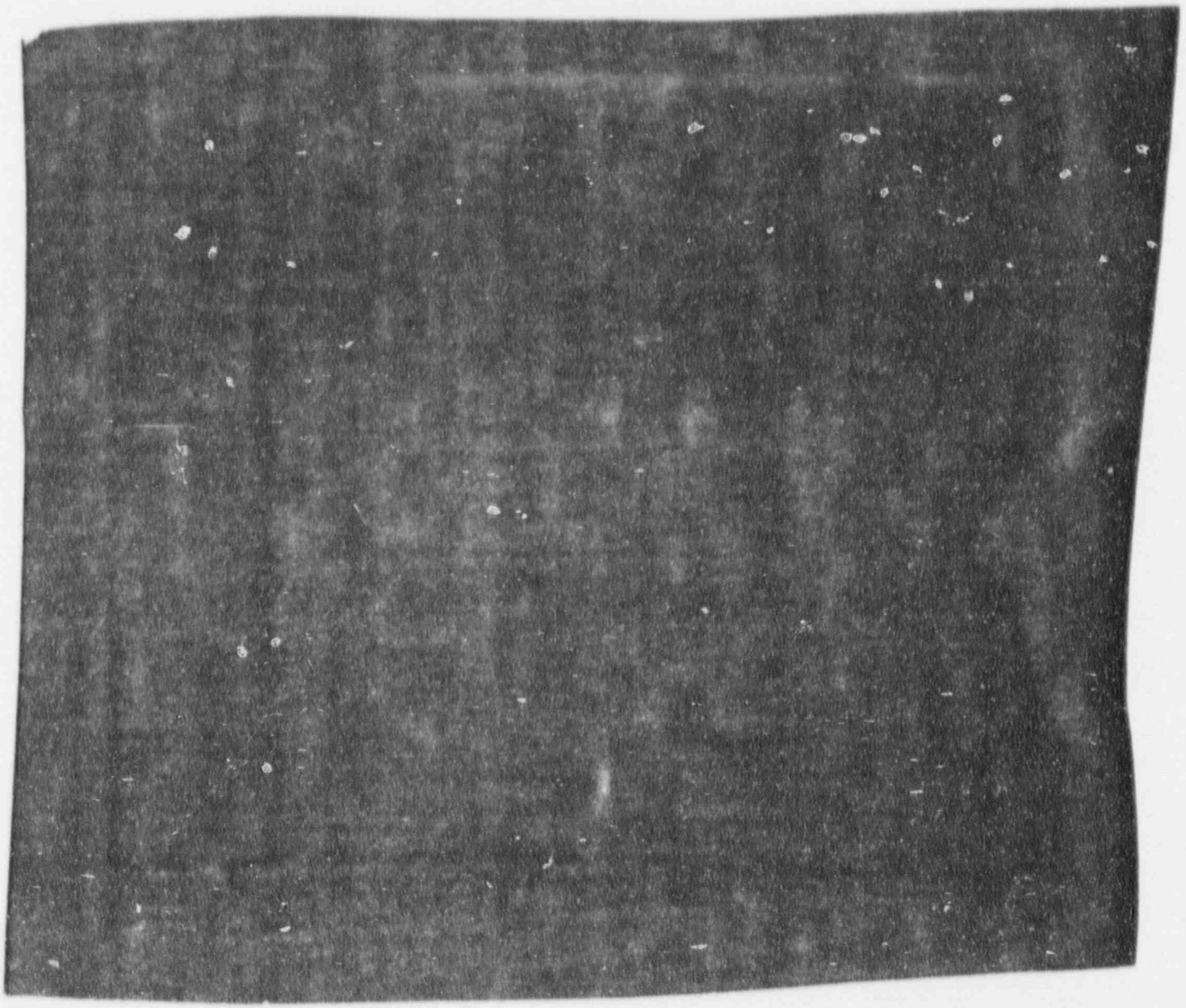


[redacted]

1.3 UPRATED PLANT OPERATING CONDITIONS

The following evaluations justify increasing the licensed reactor thermal power to 3430 MWt. The following descriptions provide information on the original and uprated plant operating conditions.

[redacted]



1.5 REFERENCES

1. GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31897P-1, Class III (Proprietary), June 1991.

2. GE Nuclear Energy, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31984P, Class III (Proprietary), July 1991.

Table 1-1

GLOSSARY OF TERMS

Term	Definition
ABBA	ASEA Brown Boveri Atom
A/C	Air Conditioning
ADS	Automatic Depressurization System
AOO	Anticipated Operating Occurrences (Moderate Frequency Transient Events)
AP	Annulus Pressurization
APRM	Average Power Range Monitor
ARTS	APRM/RBM/Technical Specifications
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transients Without Scram
ATWS-RPT	Anticipated Transients Without Scram Recirculation Pump Trip
BHP	Brake HorsePower
BOP	Balance-of-Plant
BP	Bypass
BWR	Boiling Water Reactor
CC	Collapse Criteria
CFR	Code of Federal Regulations
CLP	Containment Long-Term Program
CPD	Condensate Polishing Demineralizers
CRD	Control Rod Drive
CS	Core Spray
DBA	Design Basis Accident
DECo	Detroit Edison Company
DER	Deviation Event Report
DG	Diesel Generator
DGSW	Diesel Generator Service Water
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EECW	Emergency Equipment Cooling Water
EESW	Emergency Equipment Service Water
EFPY	Effective-Full-Power-Year
EOC	End-of-Cycle
EOOS	Equipment Out-of-Service
EQ	Equipment Qualification
FPC	Fuel Pool Cooling
FW	Feedwater
FFWTR	Final Feedwater Temperature Reduction

Table 1-1 (cont'd)

GLOSSARY OF TERMS

Term	Definition
FWCF	Feedwater Controller Failure
FWHOS	Feedwater Heater(s) Out-of-Service
GE	GE Nuclear Energy/General Electric Company
GESTAR	General Electric Standard Application for Reactor Fuel
GSW	General Service Water
HCU	Hydraulic Control Unit
HELB	High-Energy Line Break
HPCI	High Pressure Coolant Injection
HVAC	Heating, Ventilation, and Air Conditioning
ICF	Increased Core Flow
IEB	Inspection and Enforcement Bulletin
IEEE	Institute of Electrical and Electronic Engineers
IPE	Individual Plant Examination
IRM	Intermediate Range Monitor
JR	Jet Reaction
LCS	Leakage Control System
LFA	Lead Fuel Assembly
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-Coolant Accident
LPCI	Low Pressure Coolant Injection
LPRM	Low Power Range Monitor
LTR1	Licensing Topical Report No. 1 (NEDC-31897P-1)
LTR2	Licensing Topical Report No. 2 (NEDC-31984P)
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCC	Motor Control Center
MCPR	Minimum Critical Power Ratio
MELC	Moderate Energy Line Crack
MELLL	Maximum Extended Load Line Limit
MEOD	Maximum Extended Operating Domain
MG	Motor Generator
MSIV	Main Steam Isolation Valve
MSLB	Main Steamline Break
MSR	Moisture Separator Reheater
MWD/T	Megawatt Days/Tonne
MWt	Megawatt thermal
NEMA	National Electrical Manufactures Association
NPSH	Net Positive Suction Head

Table 1-1 (cont'd)

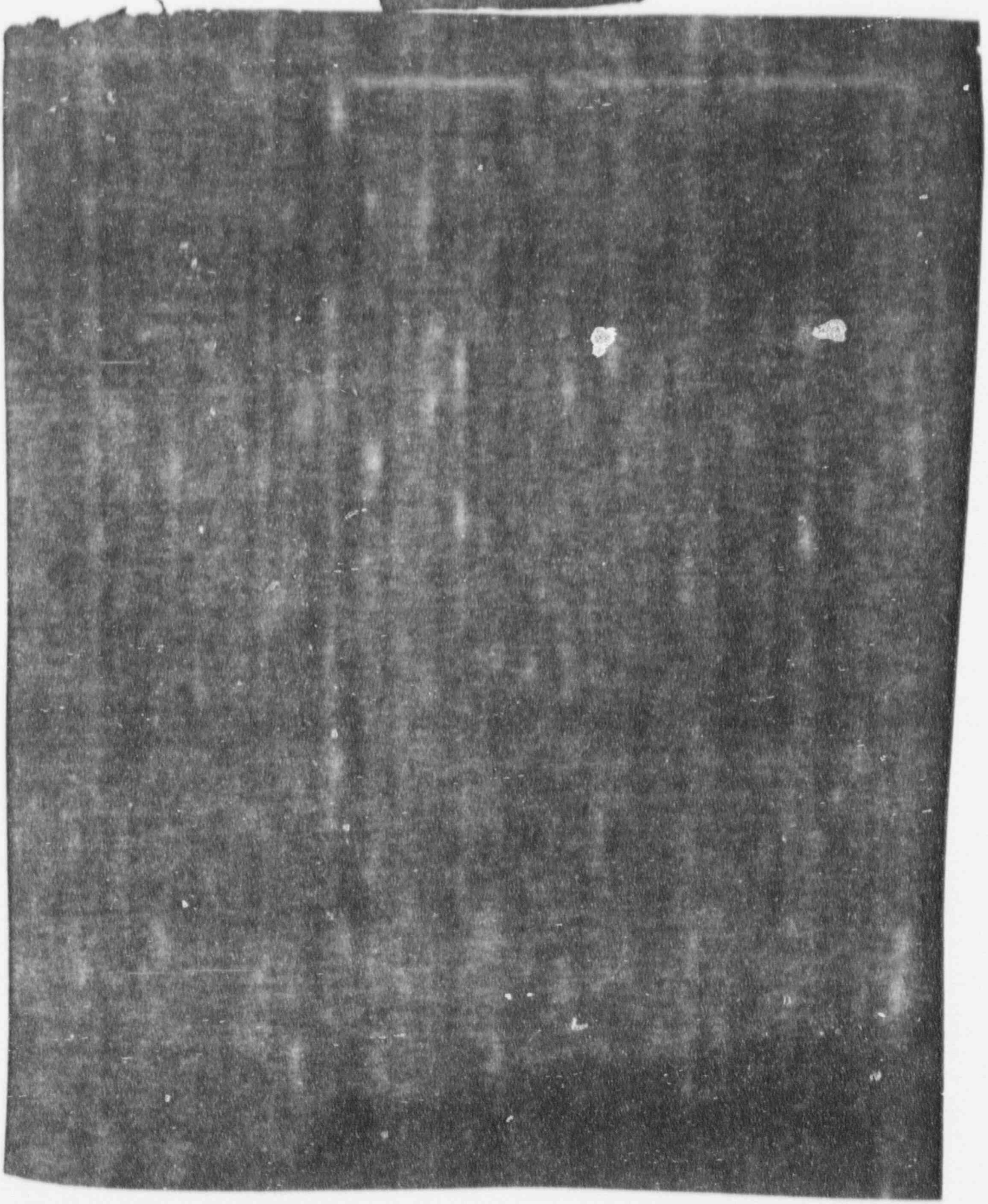
GLOSSARY OF TERMS

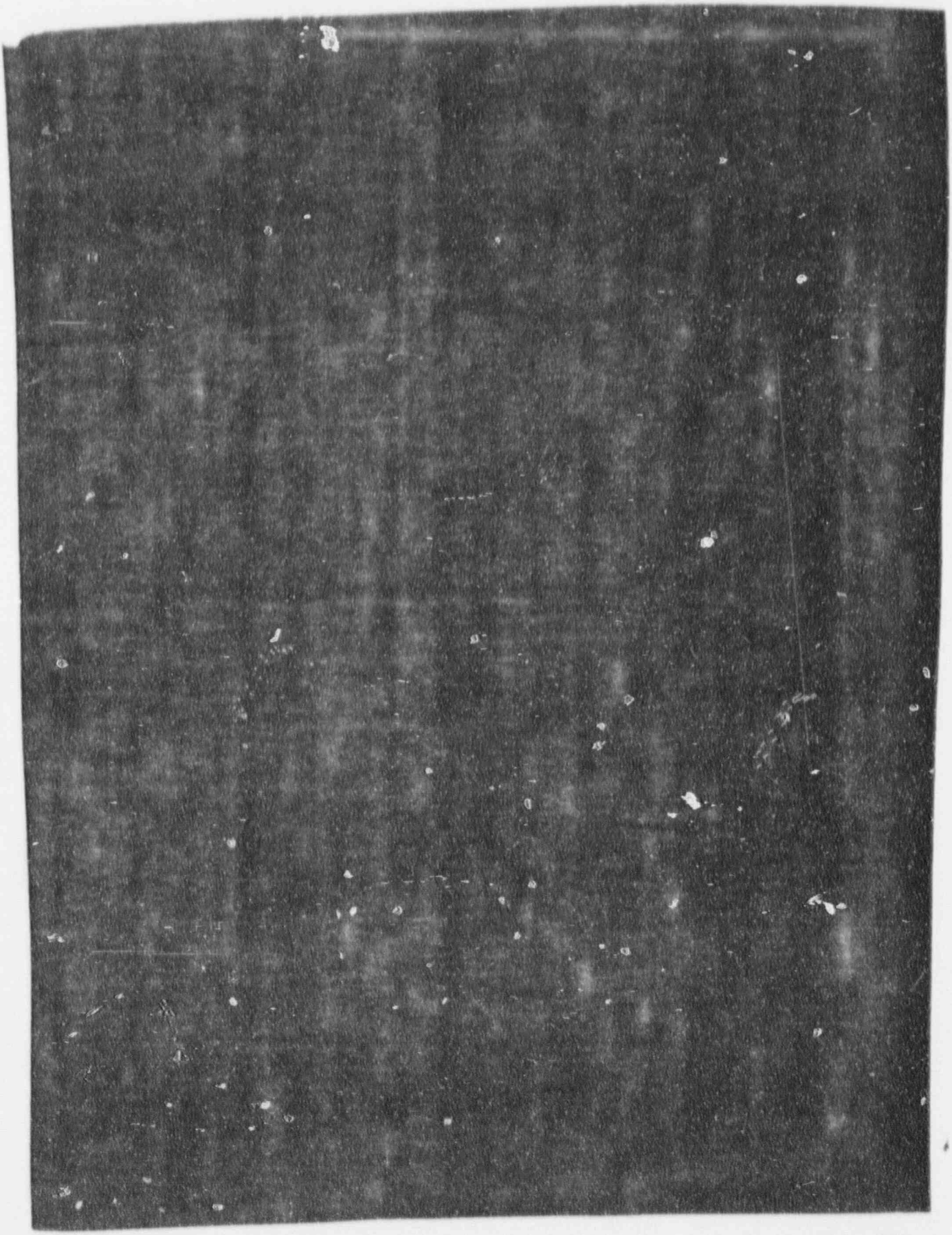
Term	Definition
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NUREG	Nuclear Regulations
OLMCPR	Operating Limit Minimum Critical Power Ratio
PCS	Pressure Control System
PCT	Peak Cladding Temperature
PFH	Partial Feedwater Heating
PRA	Probabilistic Risk Analysis
PULD	Plant Unique Load Definition
RACTS	Regulatory Action Commitment Tracking System
RBCCW	Reactor Building Closed Cooling Water
RBM	Rod Block Monitor
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
RFP	Reactor Feed Pump
RF03	Refueling Outage No. 3
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RIPD	Reactor Internals Pressure Difference
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Valve
RR&IS	Regulatory Requirements and Industry Standards
RT_{NDT}	Reference Temperature of Nil-Ductility Transition
RWCU	Reactor Water Cleanup
RWE	Rod Withdrawal Error
SAR	Safety Analysis Report
SBFW	Standby Feedwater
SBO	Station Blackout
SE	Safety Evaluation
SGTS	Standby Gas Treatment System
SIL	GE Service Information Letter
SJAE	Steam Jet Air Ejector
SLC	Standby Liquid Control
SLMCPR	Safety Limit Minimum Critical Power Ratio
SLO	Single-loop Operation
SPCM	Suppression Pool Cooling Mode
SRM	Source Range Monitor

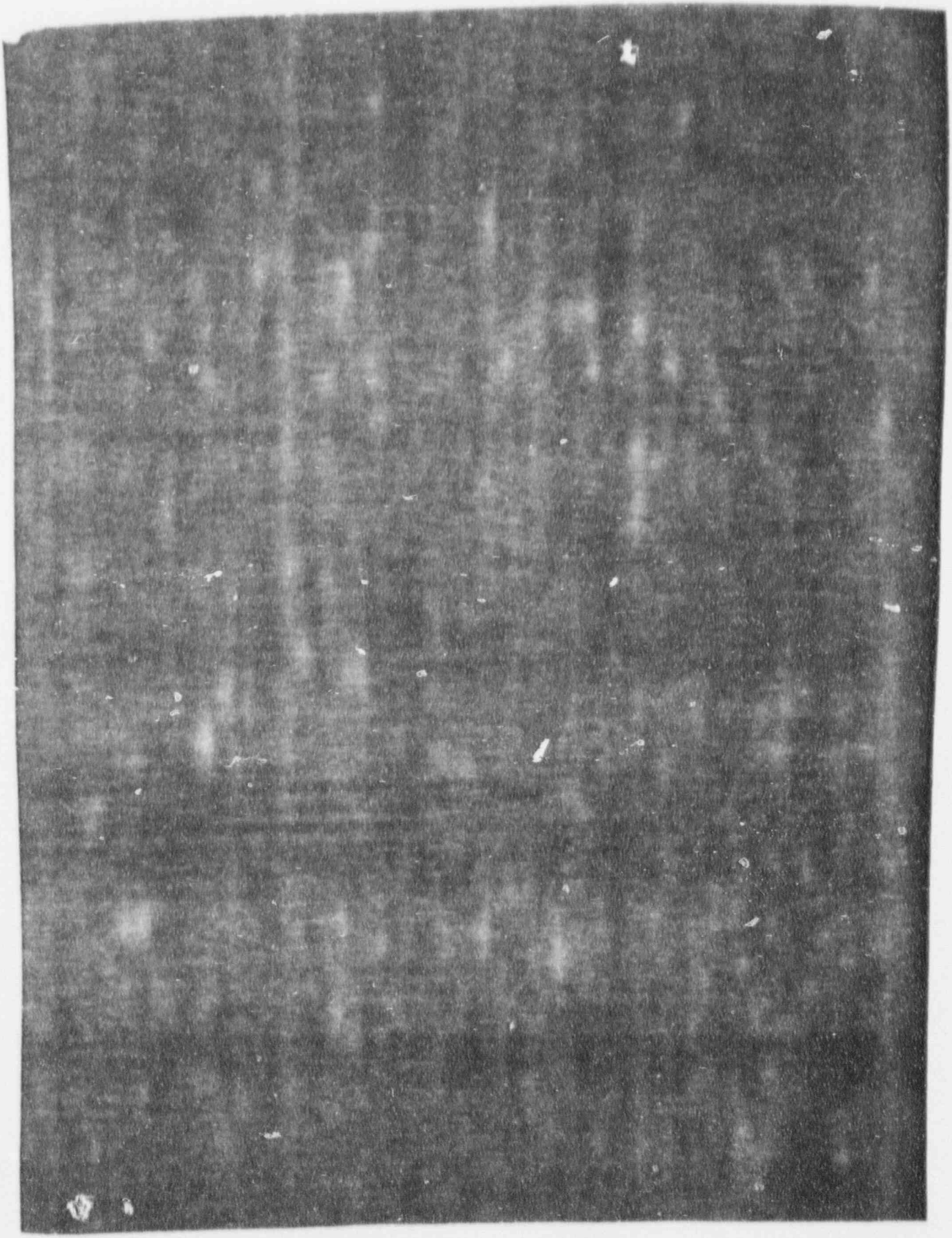
Table 1-1 (cont'd)

GLOSSARY OF TERMS

Term	Definition
SRP	Standard Review Plan
SRV	Safety/Relief Valve
SRVDL	Safety/Relief Valve Discharge Line
STPM	Simulated Thermal Power Monitor
SW	Service Water
TB	Turbine Bypass
TBCCW	Turbine Building Closed Cooling Water
TCV	Turbine Control Valve
TIP	Traversing In-Core Probe
TM	Temporary Modifications
TSV	Turbine Stop Valve
T/G	Turbine/Generator
T/GT	Turbine/Generator Trip
TPM	Thermal Power Monitor
USAR/ UFSAR	Updated Safety Analysis Report/Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
USE	Upper Shelf Energy



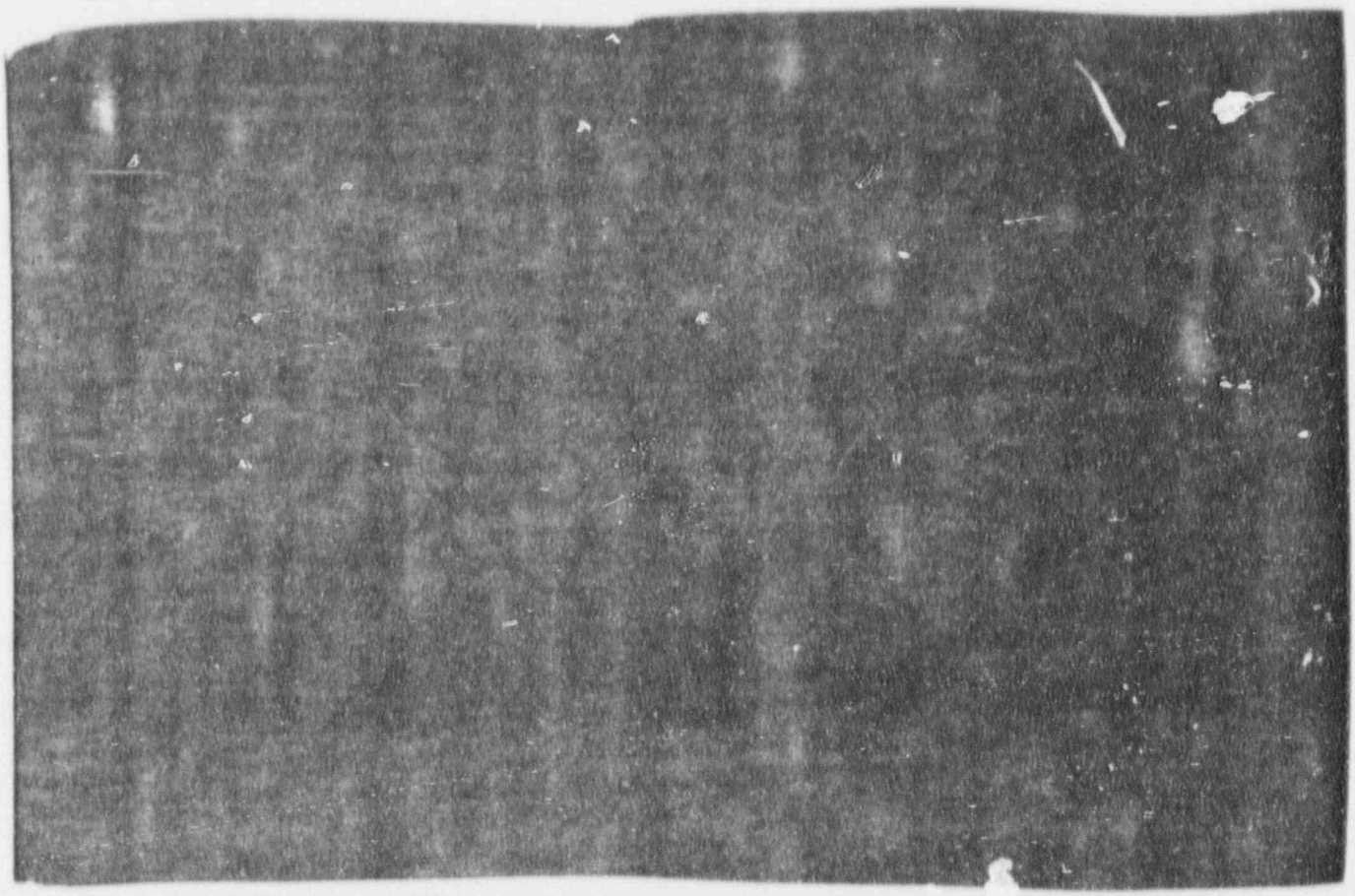




2.0 REACTOR CORE AND FUEL PERFORMANCE

This section primarily focuses on the information requested in Regulatory Guide 1.70, Chapter 4, that applies to power uprate.

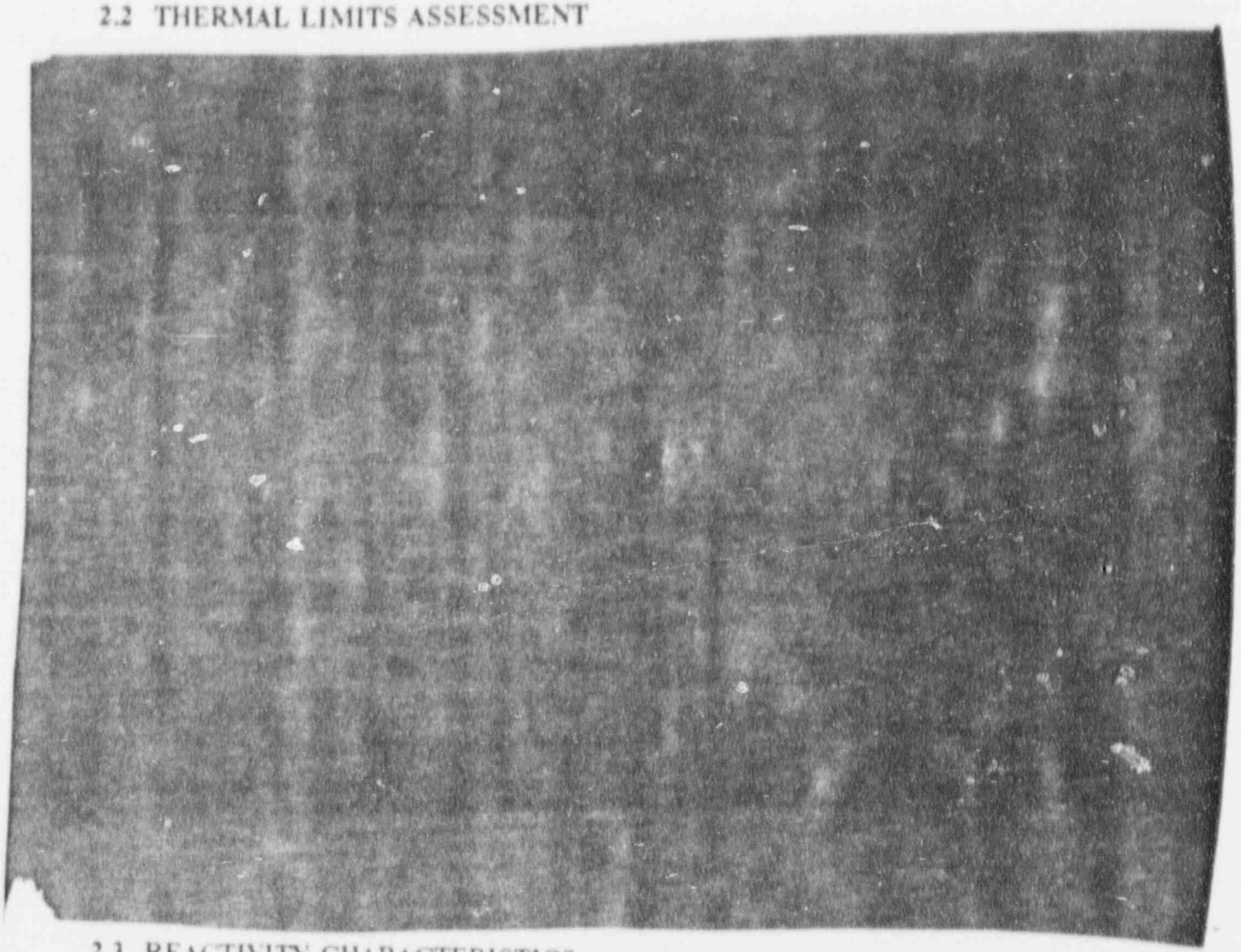
2.1 FUEL DESIGN AND OPERATION



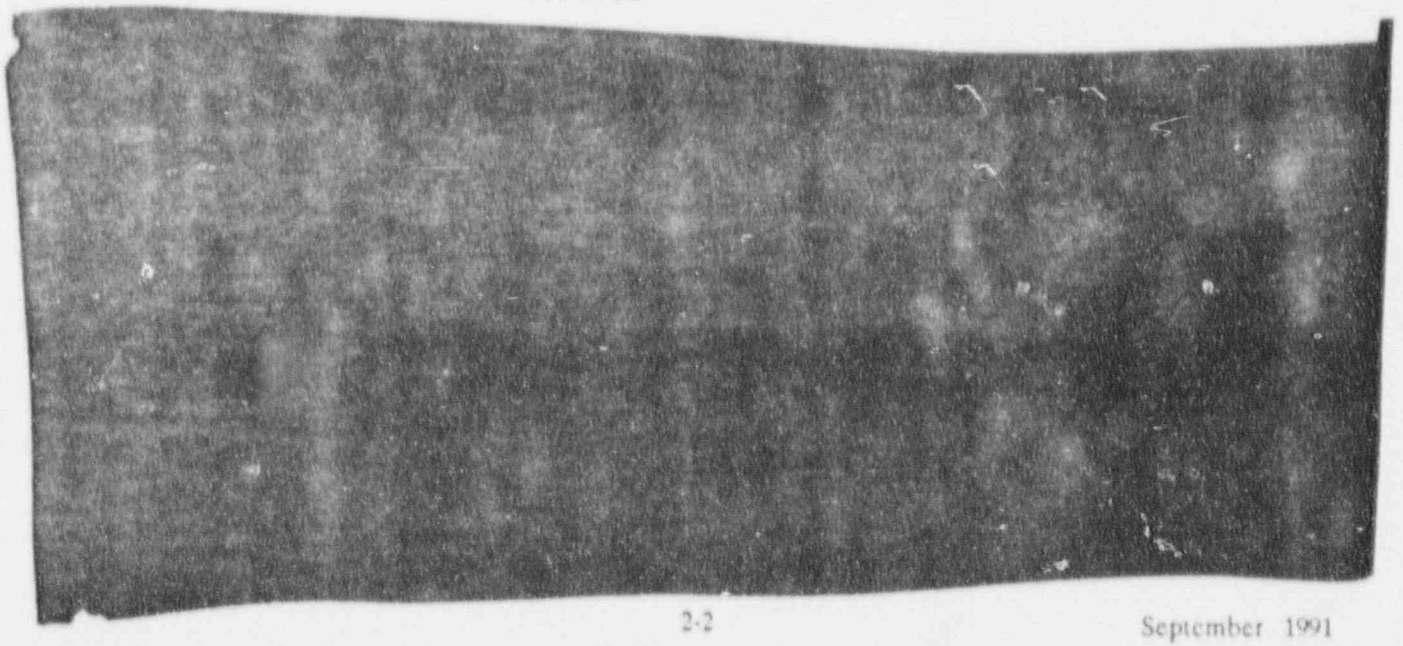
2.1.1 Lead Fuel Assemblies

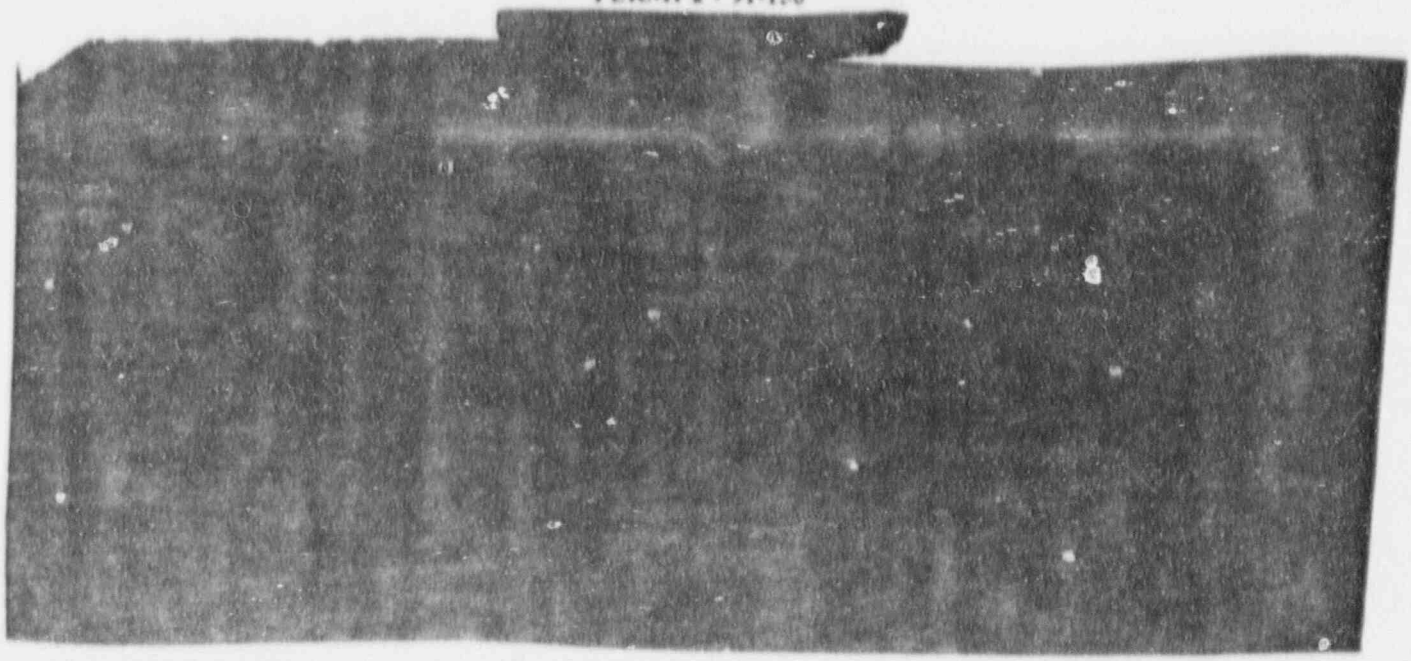
In Cycle 3, Fermi 2 inserted four ASEA Brown Boveri Atom (ABBA) SVEA-96 lead fuel assemblies (LFA). The LFAs were designed to match the GE9B reload fuel in power and thermal-hydraulic characteristics. Because of this similarity, the operating limits of the GE9B fuel bundle are used as a basis for establishing the LFA specific operating limits. In addition, the LFAs are placed in core locations where they are never expected to be the most limiting assemblies in the core on either a nodal or bundle power basis. The SVEA-96 LFA response to power uprate conditions is very similar to the GE9B bundle and, therefore, is judged to be acceptable.

2.2 THERMAL LIMITS ASSESSMENT



2.3 REACTIVITY CHARACTERISTICS

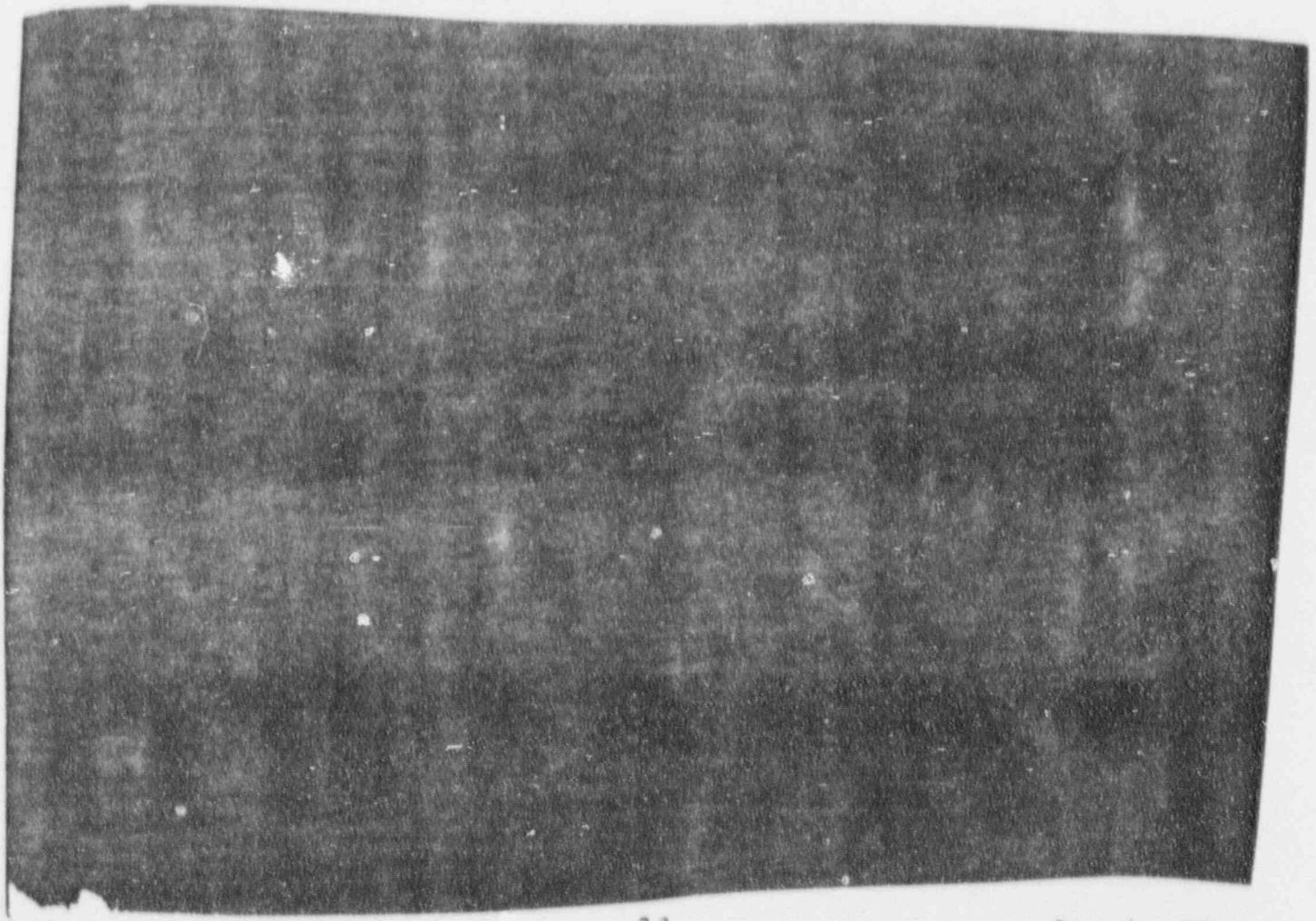




2.4 STABILITY

To maintain the same level of instability protection during uprated operation, the plant will be operated as described in Section 3.2 of Reference 1.

2.5 REACTIVITY CONTROL



2.6 REFERENCES

1. GE Nuclear Energy, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31984P, Class III (Proprietary), July 1991.

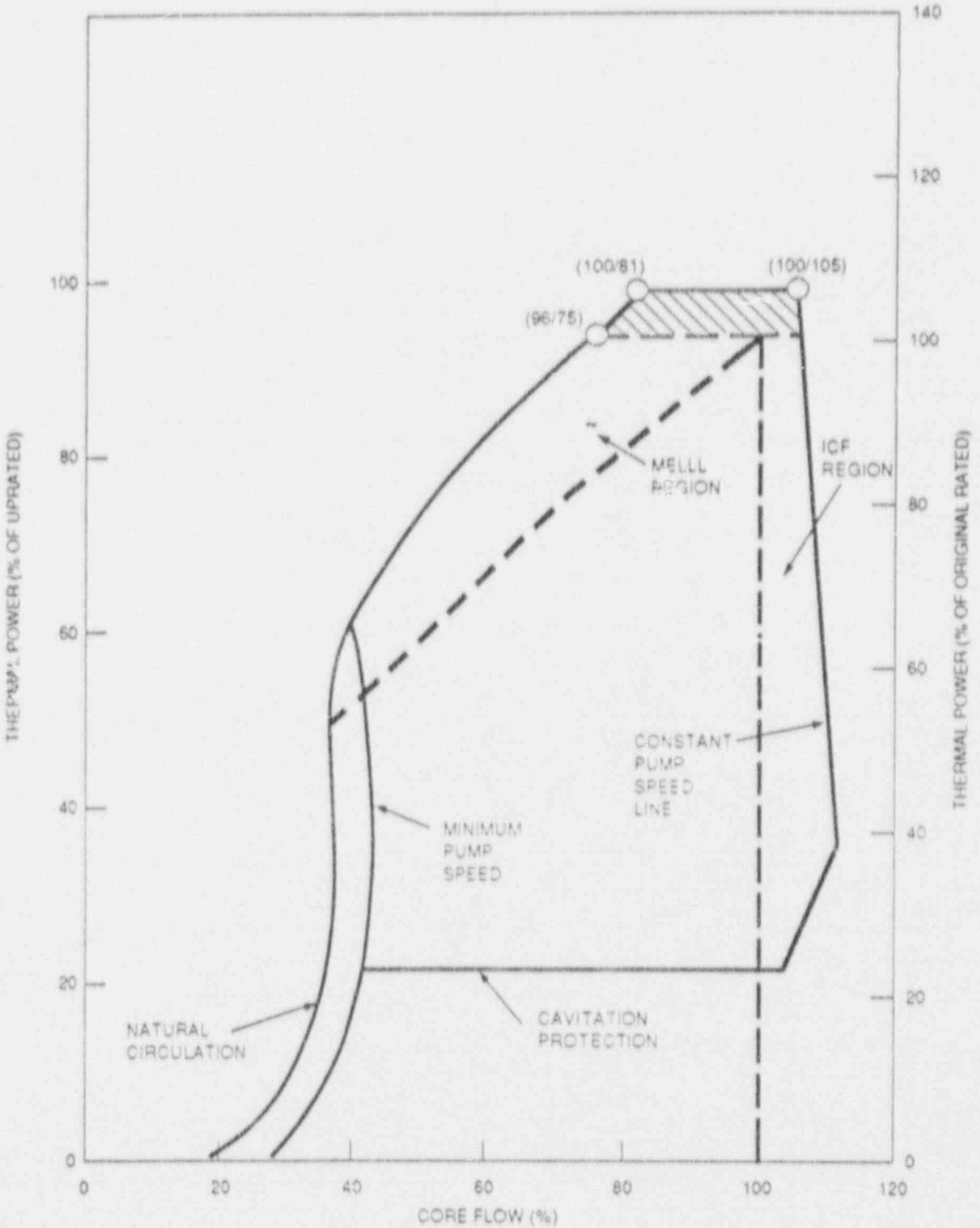
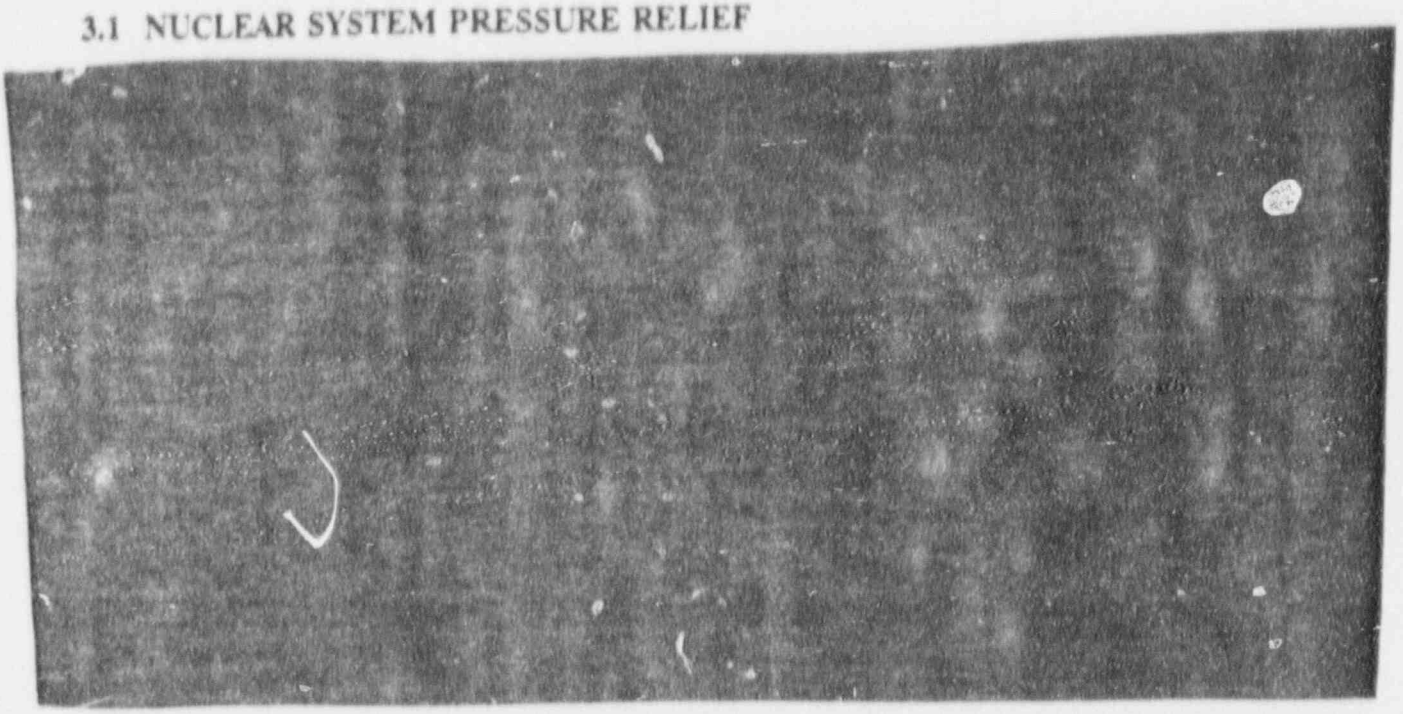


Figure 2-1. POWER/FLOW OPERATING MAP FOR POWER UPRATE

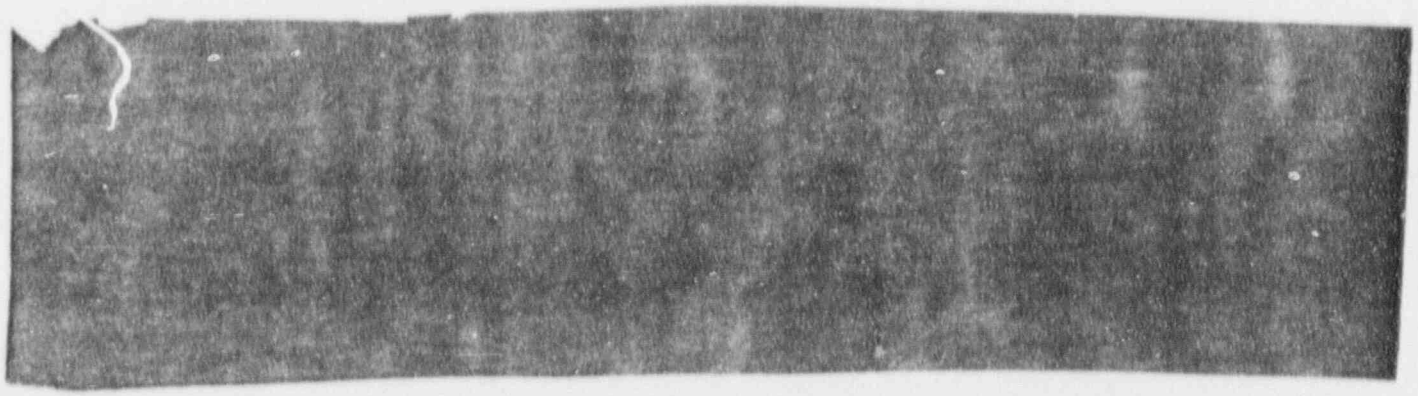
3.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

This section primarily focuses on the information requested in Regulatory Guide 1.70, Chapter 5 and to a very limited extent Chapter 3, that applies to power uprate.

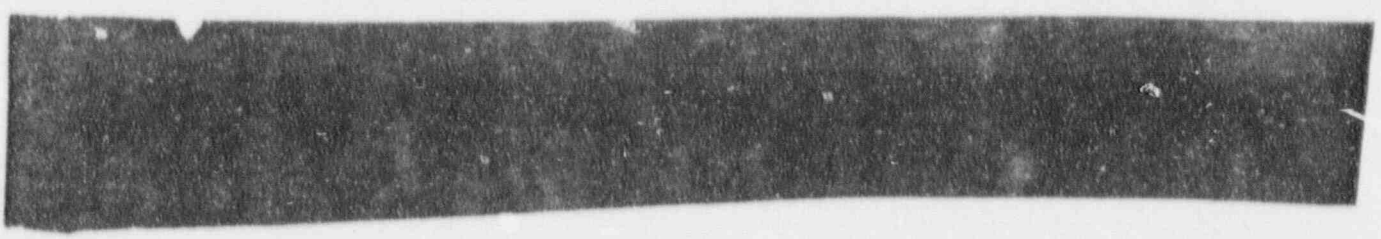
3.1 NUCLEAR SYSTEM PRESSURE RELIEF

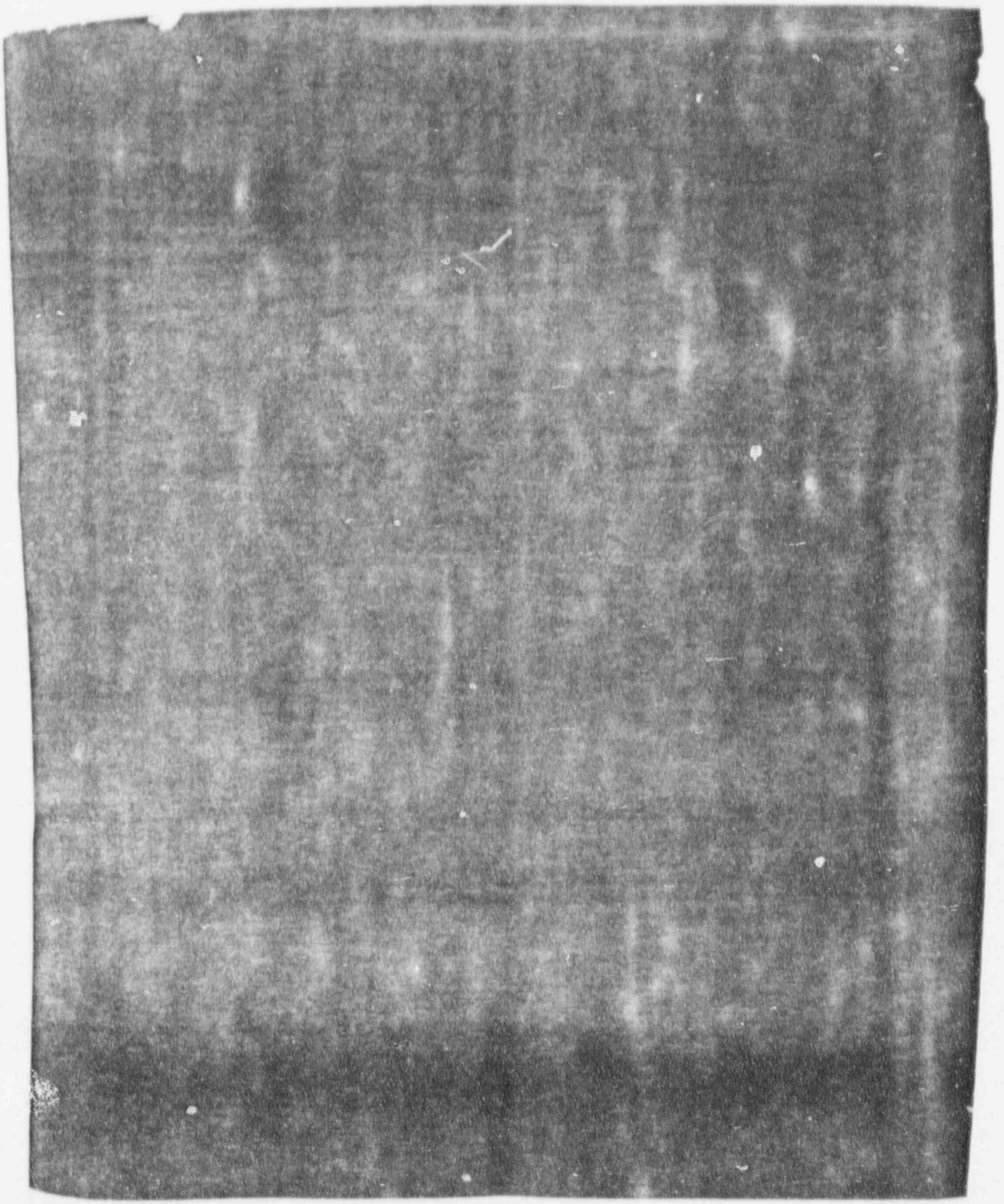


3.2 REACTOR OVERPRESSURE PROTECTION



3.3 REACTOR VESSEL AND INTERNALS

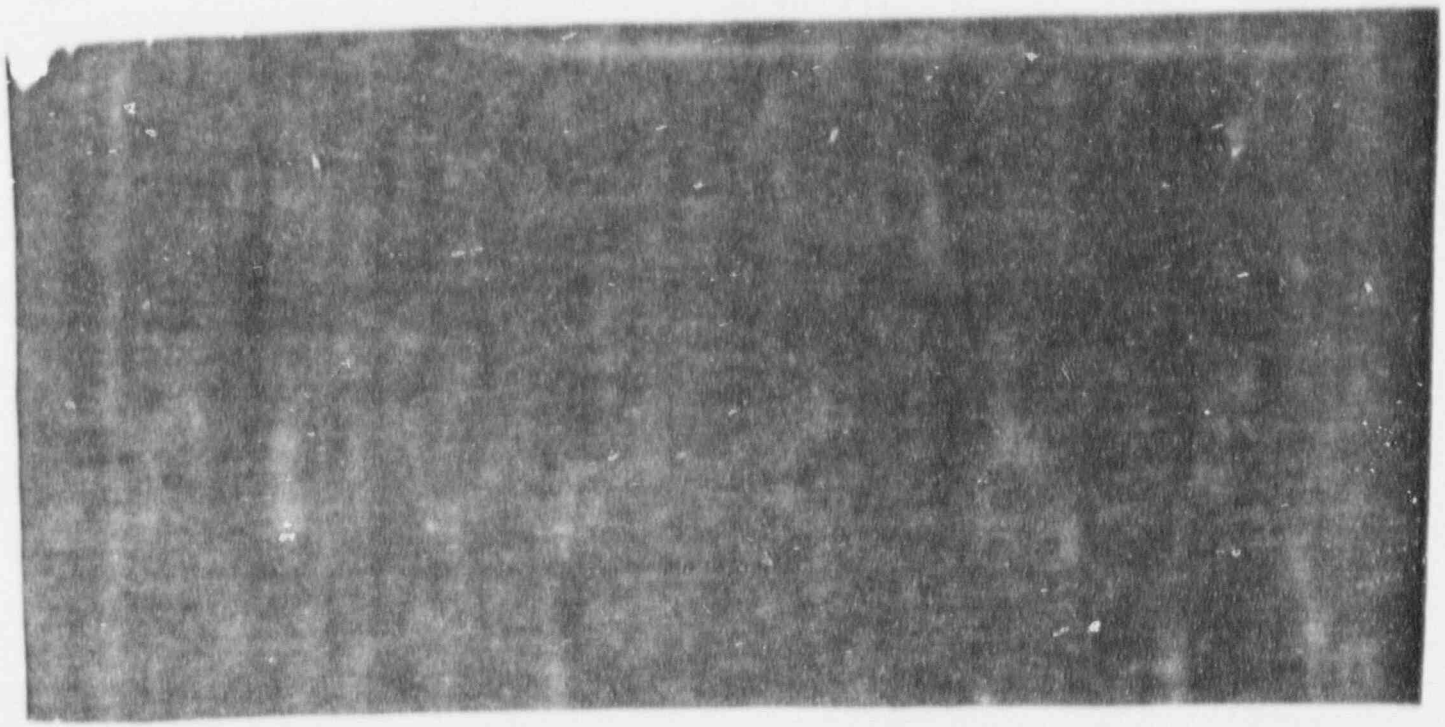




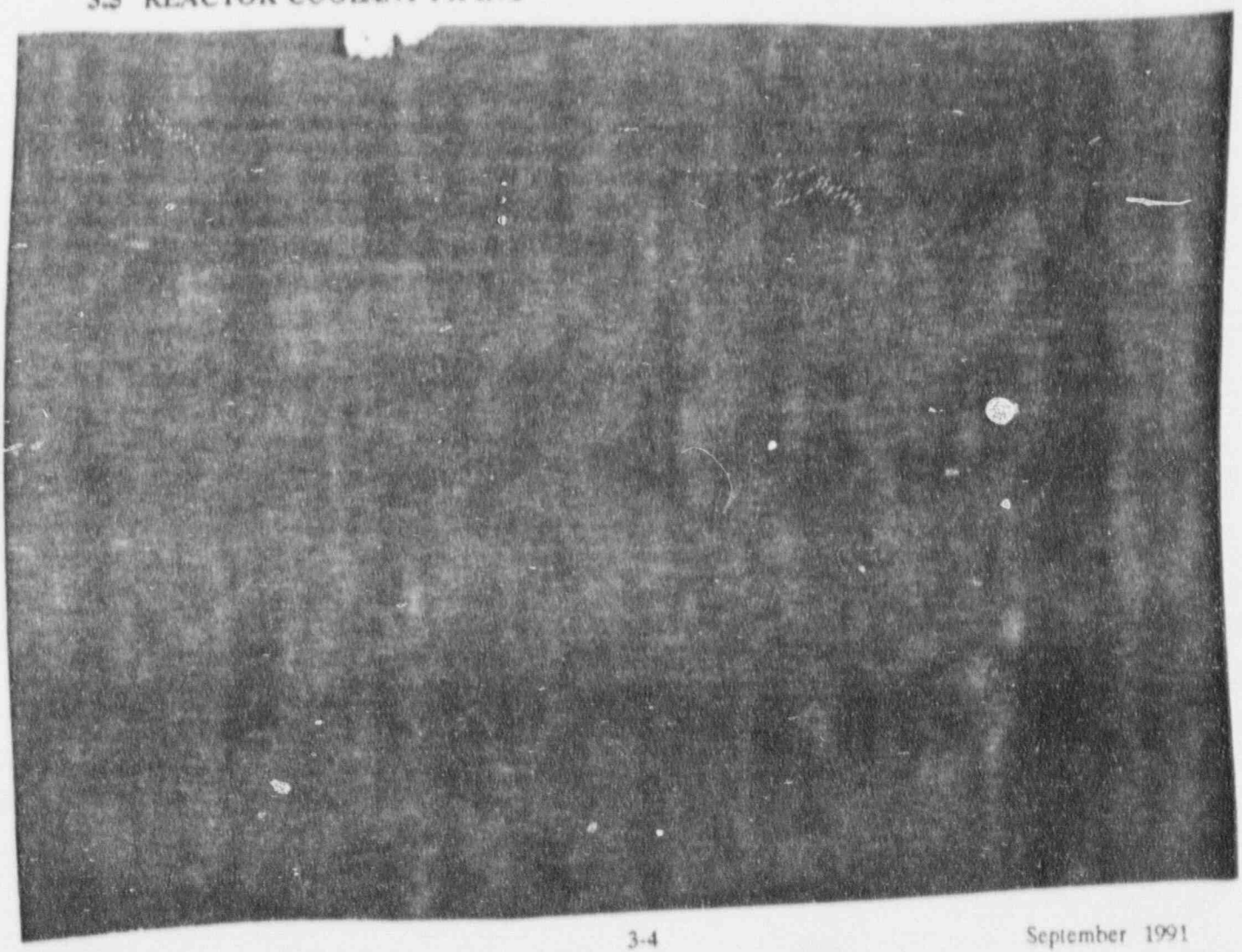
[redacted]

3.4 REACTOR RECIRCULATION SYSTEM

[redacted]



3.5 REACTOR COOLANT PIPING



[redacted]

3.6 MAIN STEAMLINE FLOW RESTRICTORS

[redacted]

3.7 MAIN STEAM ISOLATION VALVES

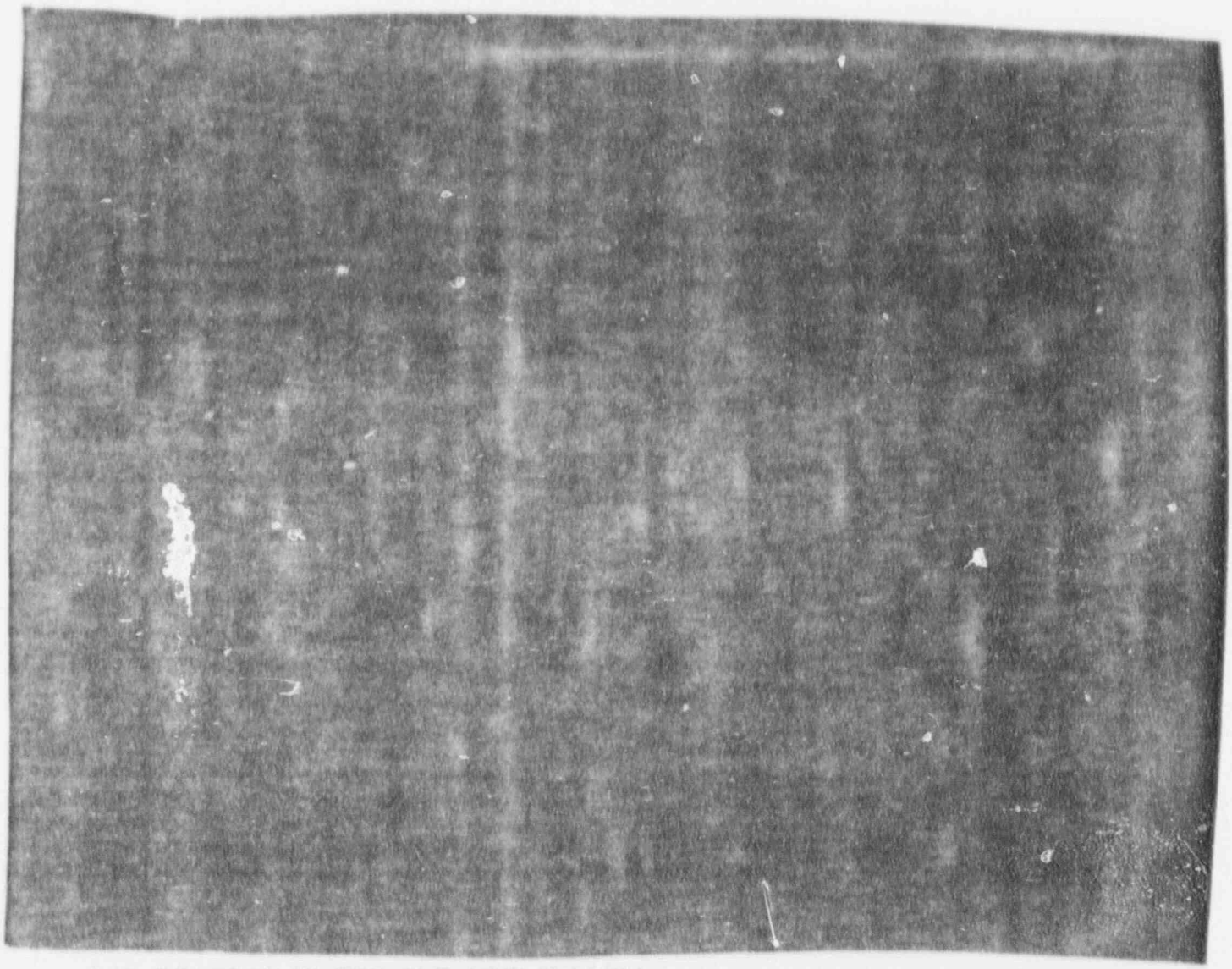
[redacted]

3.8 REACTOR CORE ISOLATION COOLING

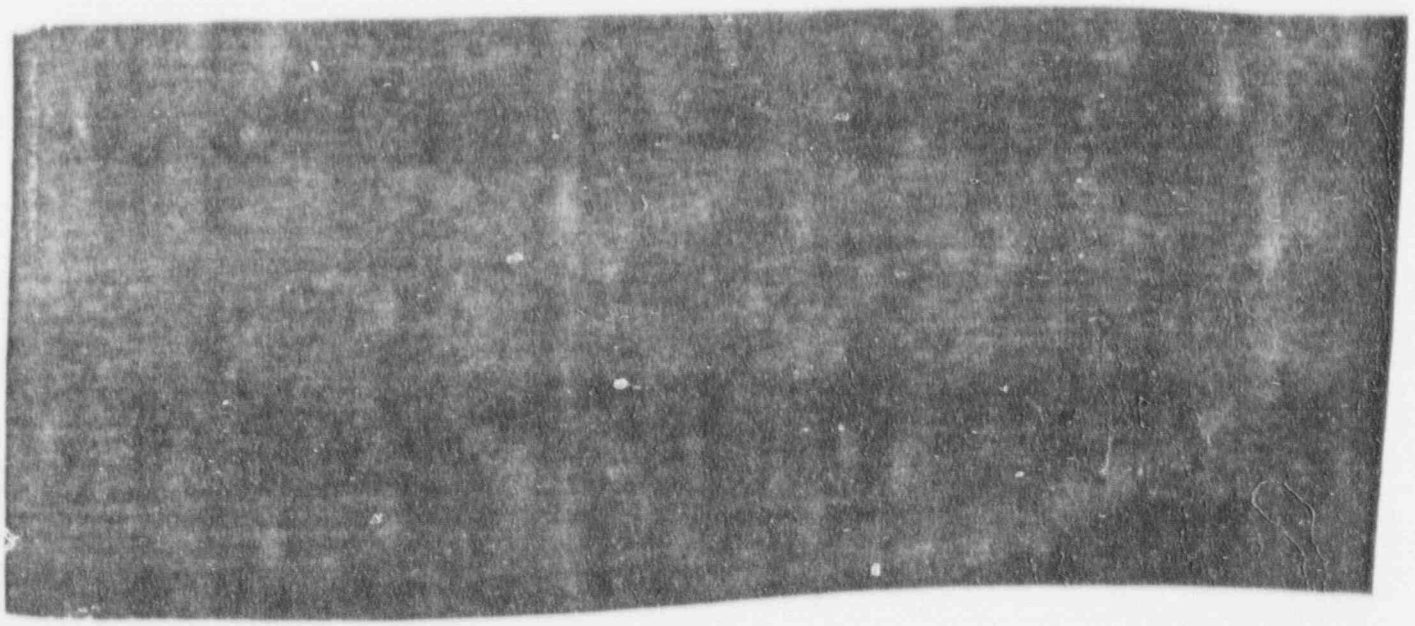
[redacted]

3.9 RESIDUAL HEAT REMOVAL SYSTEM

[redacted]



3.10 REACTOR WATER CLEANUP SYSTEM



3.11 MAIN STEAM, FEEDWATER, AND BALANCE-OF-PLANT PIPING

This section addresses the adequacy of all other piping not addressed in Section 3.5 for operation at the uprated conditions.

3.11.1. Pipe Stress Evaluation

Operation at the uprated conditions increases pipe stresses due to slightly higher operating temperatures, pressures, and flow rates internal to the pipes. For all systems, the maximum stress levels and fatigue analysis results were reviewed based on specific increases in temperature, pressure, and flow rate. These piping systems have been evaluated and meet the appropriate code criteria for the uprated conditions, based on the differences between actual stresses and code limits in the original design. Except for a few specific lines, all piping has a minimum of 5% margin relative to code allowables. No new postulated pipe break locations were identified.

3.11.2 Pipe Support Evaluation

Operation at uprated conditions causes a slight increase in the pipe support loadings due to increases in the temperature of the affected piping systems. The original support loads due to fluid transients remain unchanged since they were evaluated for uprate flow conditions. However, when considering the loading combination with other loads that are not affected by power uprate, such as seismic and deadweight, the overall combined support load increase is insignificant.

The supports of the systems with the most loading increases (main steam and feedwater systems) were reviewed to determine if there is sufficient margin to accommodate the increased loadings. This review shows that there is adequate difference between the original design stresses and code limits of the supports to accommodate the load increase within the appropriate code criteria. The existing conservatisms are introduced by the use of the lowest code allowables for various plant loading conditions; the use of generic enveloping design loads instead of actual loads; and the conservative load application on base plates, anchor bolts, and lugs.

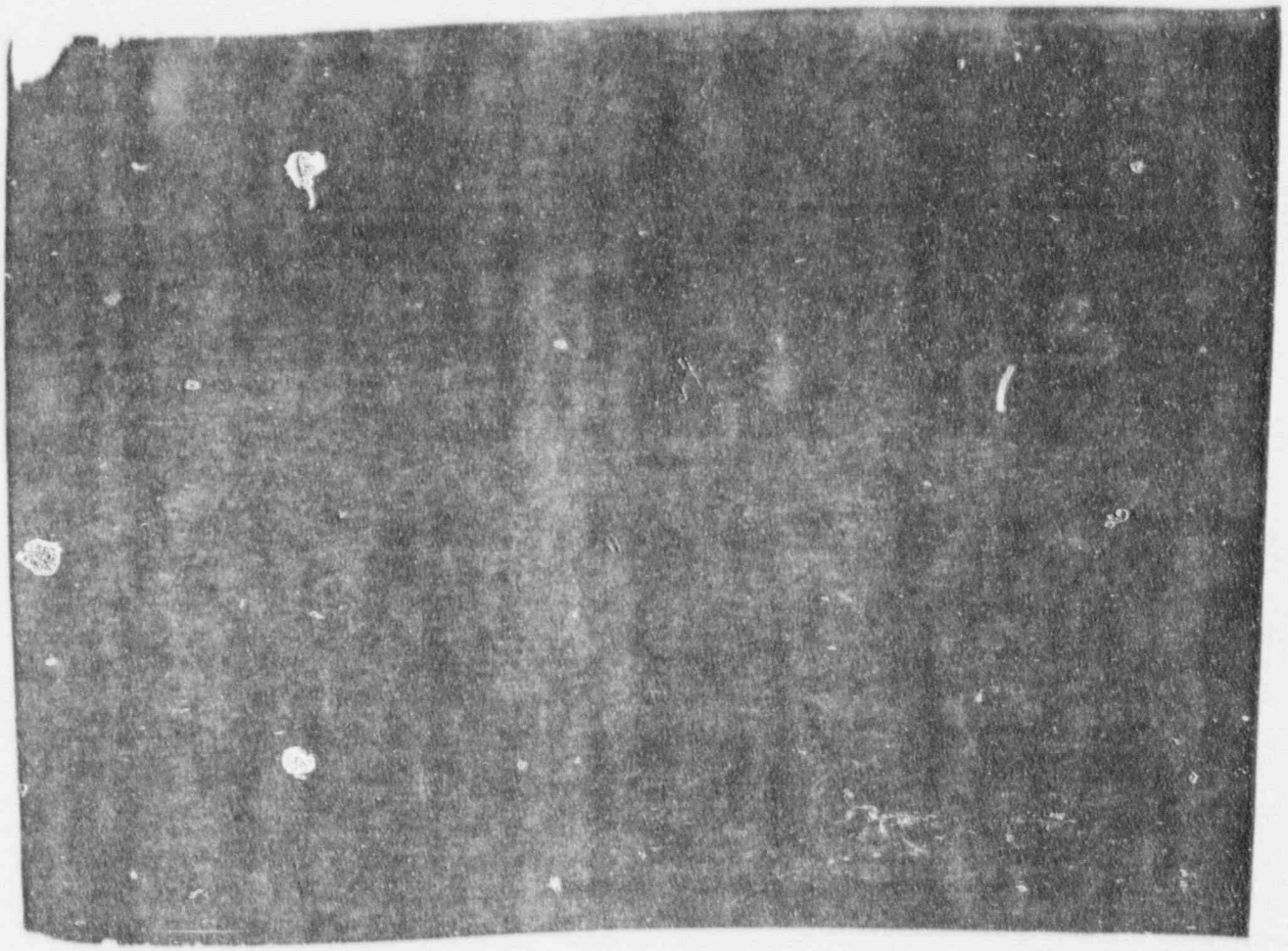
3.12 REFERENCES

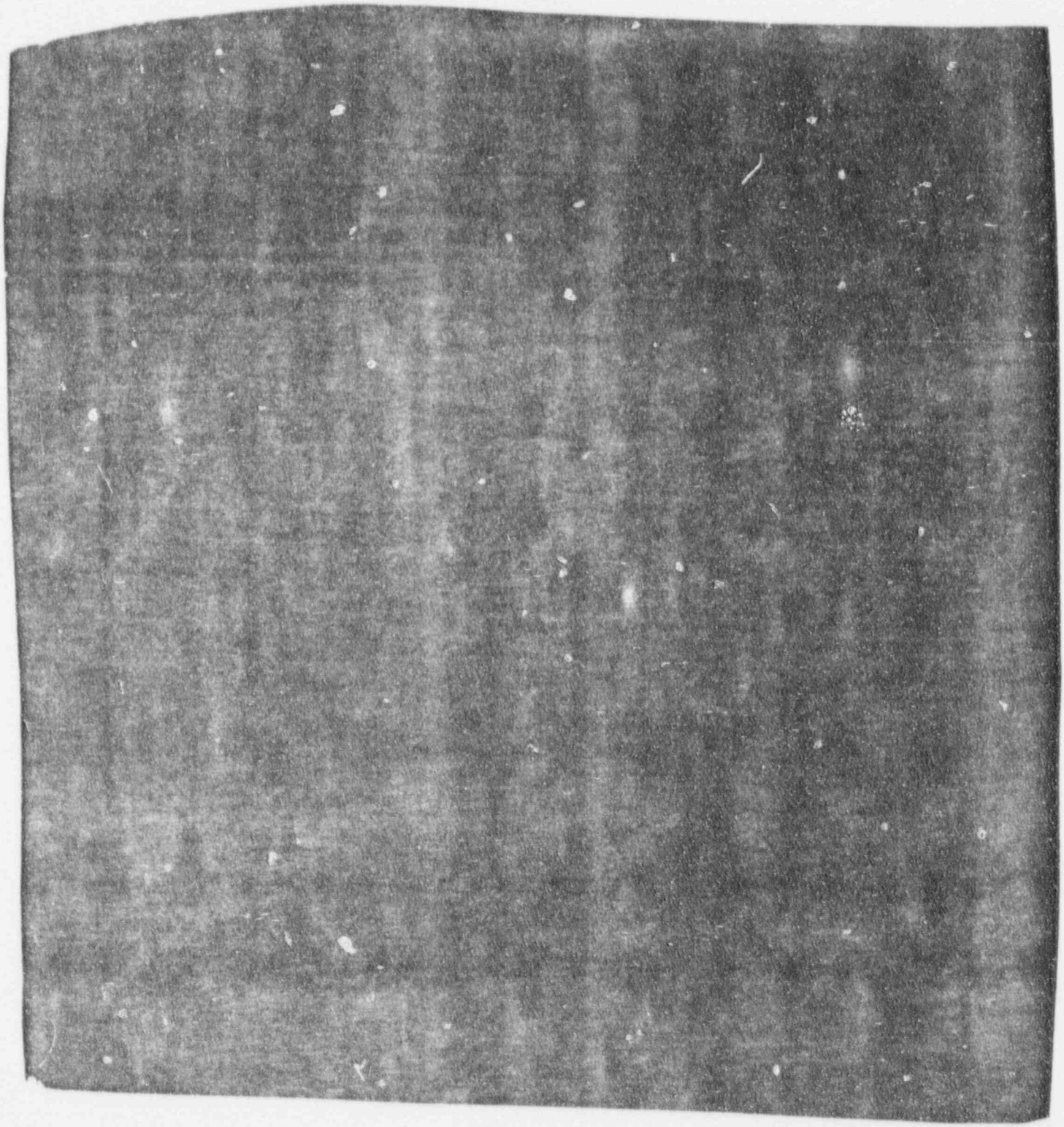
1. GE Nuclear Energy, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31984P, Class III (Proprietary), July 1991.

2. GE Nuclear Energy, "General Electric Standard Application for Reactor Fuel - GESTAR II, United States Supplement," NEDE-24011-P-A-10-US, March 1991.

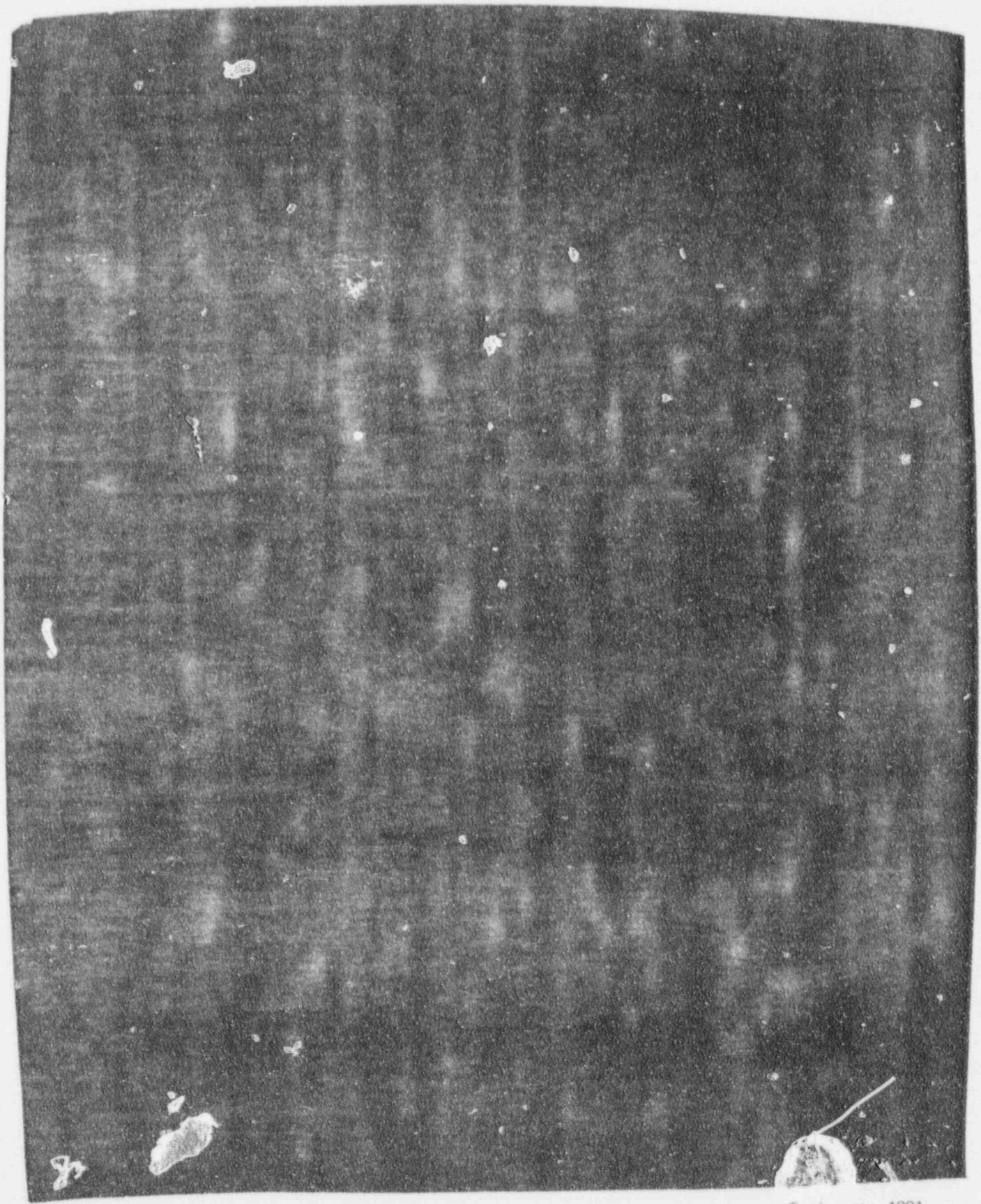
3. GE Nuclear Energy, "Maximum Extended Operating Domain Analysis for Detroit Edison Company Enrico Fermi Energy Center Unit 2," NEDC-31843P, Class III, (Proprietary), July 1990.

4. GE Nuclear Energy, "Maximum Extended Load Line Limit and Feedwater Heater Out-of-Service Analyses for Enrico Fermi Atomic Power Plant, Unit 2," NEDC-31515, Revision 1, (Proprietary), August 1989.

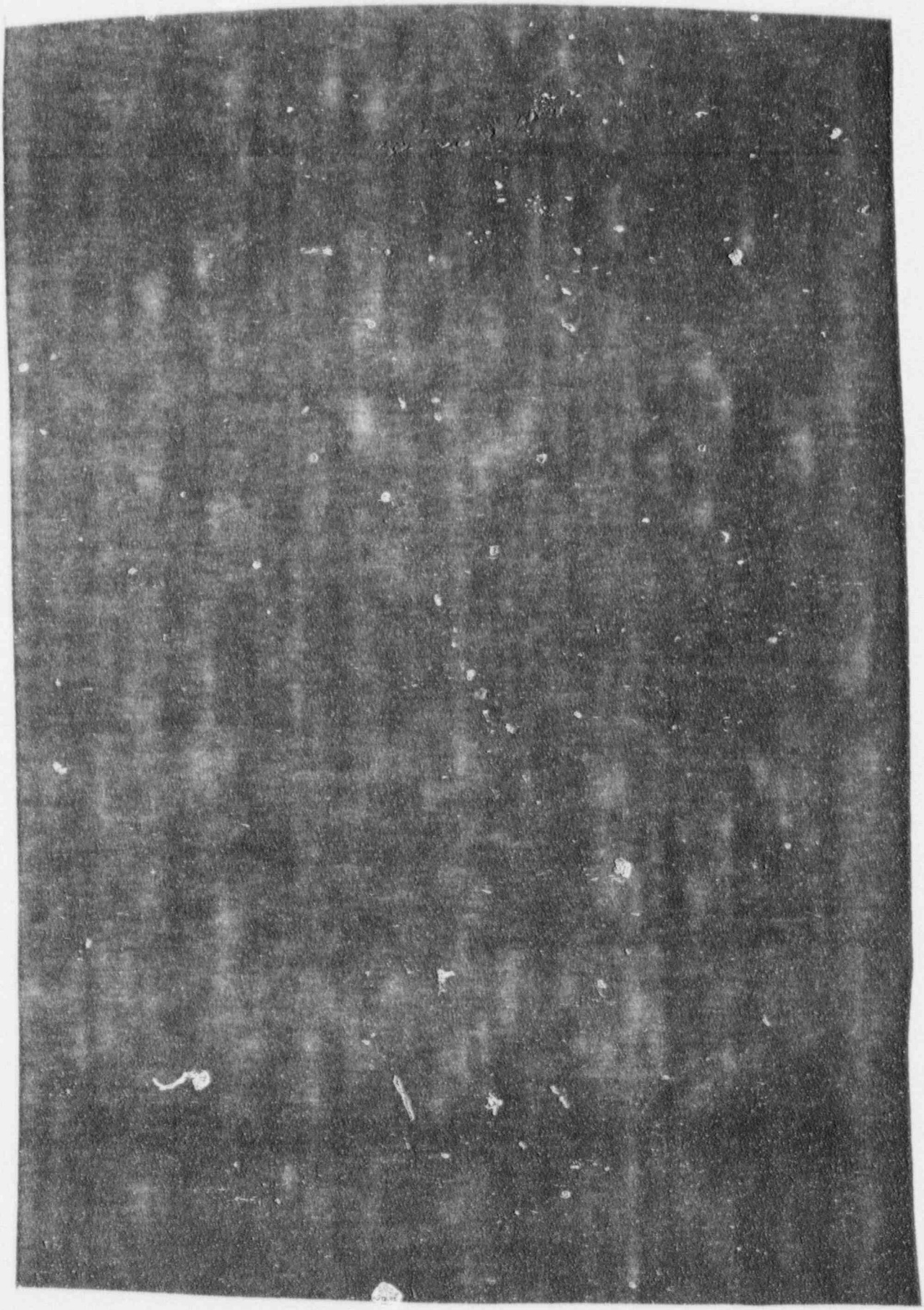




FERMI 2 - 91-150



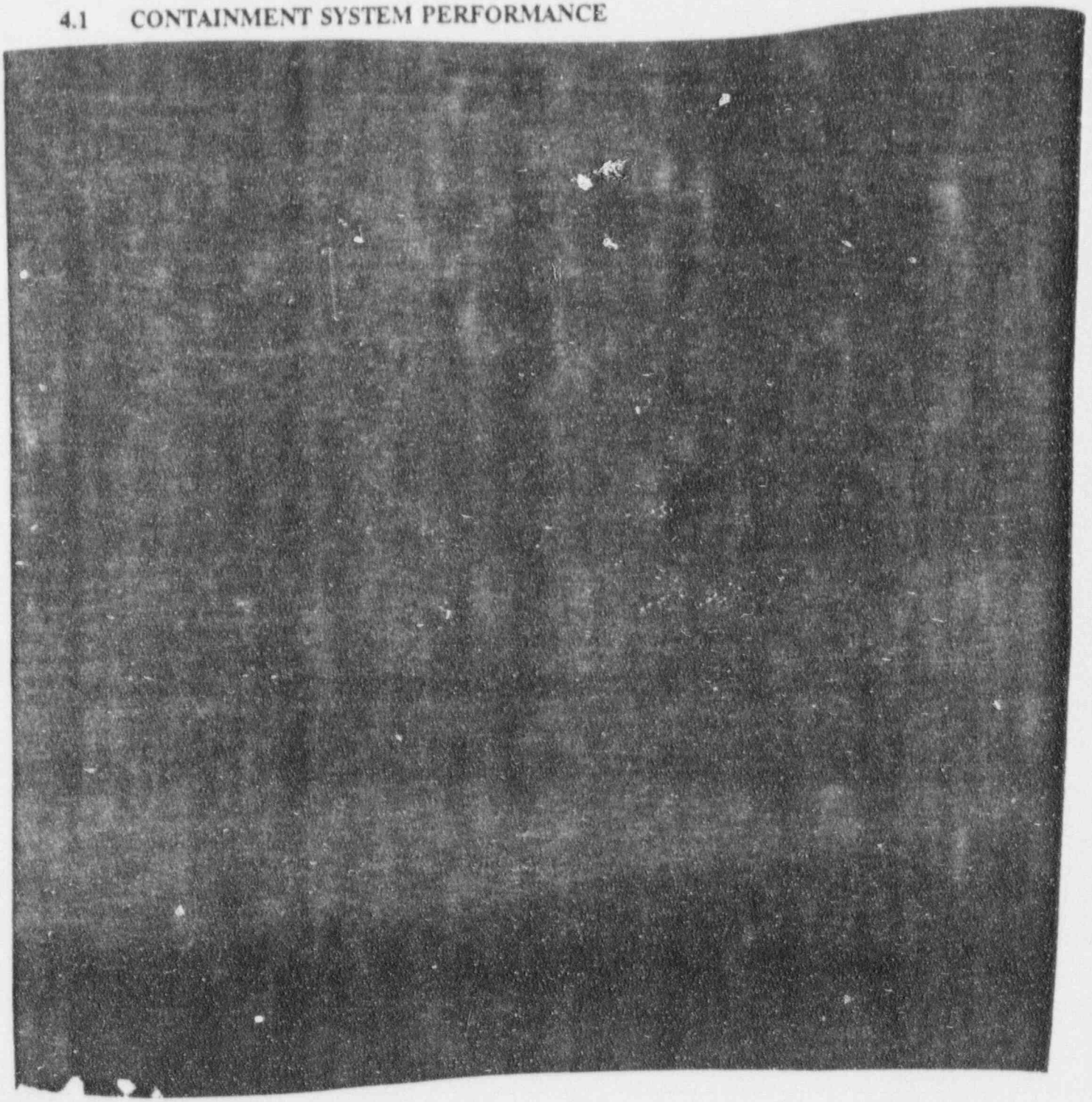
Sept ember 1991

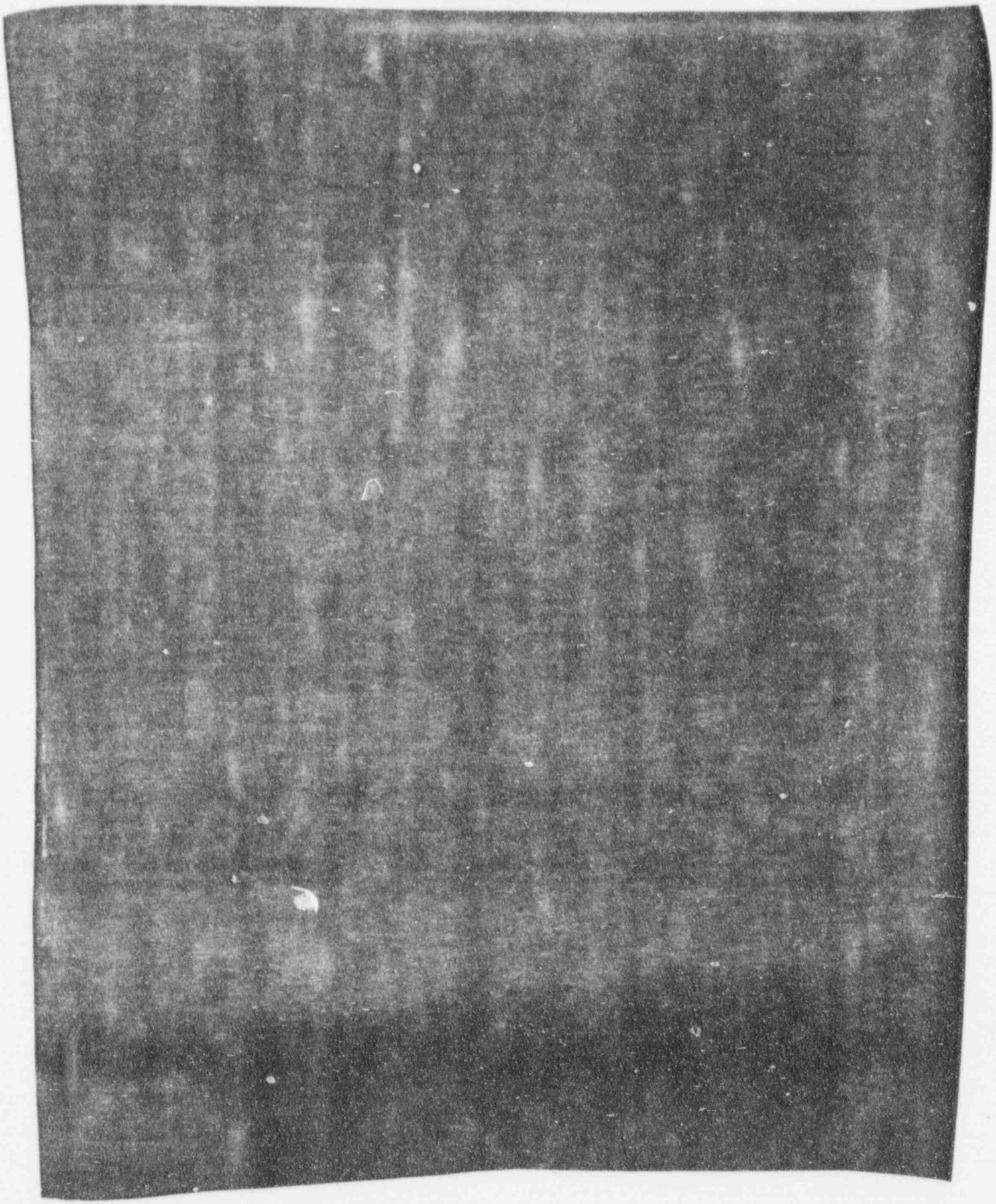


4.0 ENGINEERED SAFETY FEATURES

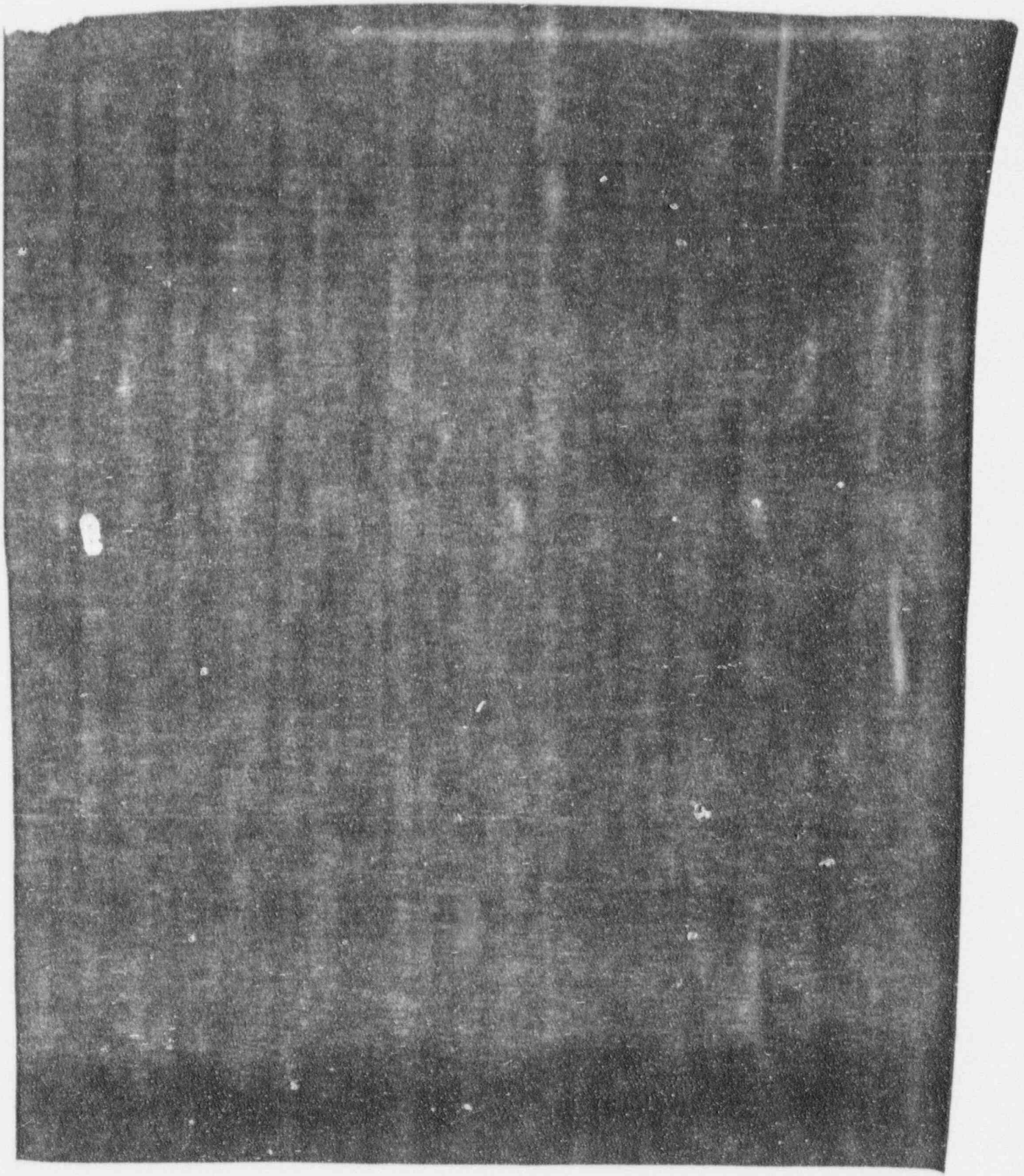
This section primarily focuses on the information requested in Regulatory Guide 1.70, Chapter 6, that applies to power uprate.

4.1 CONTAINMENT SYSTEM PERFORMANCE





[redacted]



[redacted]

4.2 EMERGENCY CORE COOLING SYSTEMS

[redacted]

[redacted]

[redacted]

4.3 EMERGENCY CORE COOLING SYSTEM PERFORMANCE EVALUATION

[redacted]

4.4 STANDBY GAS TREATMENT SYSTEM

The standby gas treatment system (SGTS) is designed to minimize offsite dose rates during venting and purging of both the primary and secondary containment atmosphere under accident or abnormal conditions, while containing airborne particulates and halogens that might be present. The capacity of the SGTS was selected to provide one secondary containment air volume change per day and maintain the Reactor Building at a slight negative pressure. This capability is not impacted by the uprate.

The design of the charcoal filter beds is unaffected by uprate. The total post-LOCA iodine loading increases slightly at the uprated conditions, but is well below the original design capability of the filter.

4.5 OTHER ESF SYSTEMS

Several systems/features are addressed in this section to be consistent with the NRC Safety Evaluation Report (Reference 13) and the Fermi 2 USAR.

4.5.1 Main Steam Isolation Valve Leakage Control System

The containment analysis in Section 4.1 demonstrates that the peak post-LOCA pressures do not increase beyond the original safety basis. Therefore, the operation of the MSIVLCS will not be impacted by power uprate, as noted in Section 6.8.

4.5.2 Post-LOCA Combustible Gas Control

The hydrogen recombiner operation is required after a LOCA to maintain the containment atmosphere as a non-combustible mixture. The combustibility of the post-LOCA containment atmosphere is controlled by the concentration of oxygen. As discussed in Reference 1, a result of power uprate is that the post-LOCA production of oxygen by radiolysis will increase proportionally with the power level. Sufficient capacity exists in the combustible gas control system to accommodate the increased oxygen production. Also, recombiner operation may be initiated earlier than for the original power level. However, initiation of the recombiners is controlled procedurally based on gas concentration, and not by time.

4.5.3 Emergency Cooling Water System

This system is addressed in Section 6.4.

4.5.4 Emergency Core Cooling Auxiliary Systems

These systems are addressed in Section 6.4.

4.5.5 Main Control Room Atmosphere Control System

This system is not affected by power uprate, as noted in Section 6.8.

4.5.6 Standby Power System

Standby power is addressed in Sections 6.1 and 6.2.

4.5.7 Main Steamline Flow Restrictors

The MSL flow restrictors are designed for power uprate, as discussed in Section 3.6.

4.5.8 CRD Velocity Limiters

The CRD velocity limiter design is not affected by power uprate.

4.6 REFERENCES

1. GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31897P-1, Class III (Proprietary), June 1991.

2. General Electric Co., "The GE Pressure Suppression Containment System Analytical Model," NEDO-10320, Supplement 1, May 1971.

3. General Electric Co., "The General Electric Mark III Pressure Suppression Containment System Analytical Model," NEDO-20533, June 1974.

4. General Electric Co., "The General Electric Mark III Pressure Suppression Containment System Analytical Model Supplement 1," NEDO-20533-1, September 1975.

5. General Electric Co., "Maximum Discharge of Liquid-Vapor Mixtures from Vessels," NEDO-21052, September 1975.

6. Nuclear Regulatory Commission, "Standard Review Plan, Pressure - Suppression Type BWR Containments," NUREG-0800, Setion 6.2.1.1.C, Revision 6, August 1984.

7. U. S. Nuclear Regulatory Commission, "Mark I Containment Long-Term Program Safety Evaluation Report," NUREG-0661, July 1980.

8. General Electric Co., "Mark I Containment Program Load Definition Report," NEDO-21888, Revision 0, December 1978, and Revision 2, November 1981.

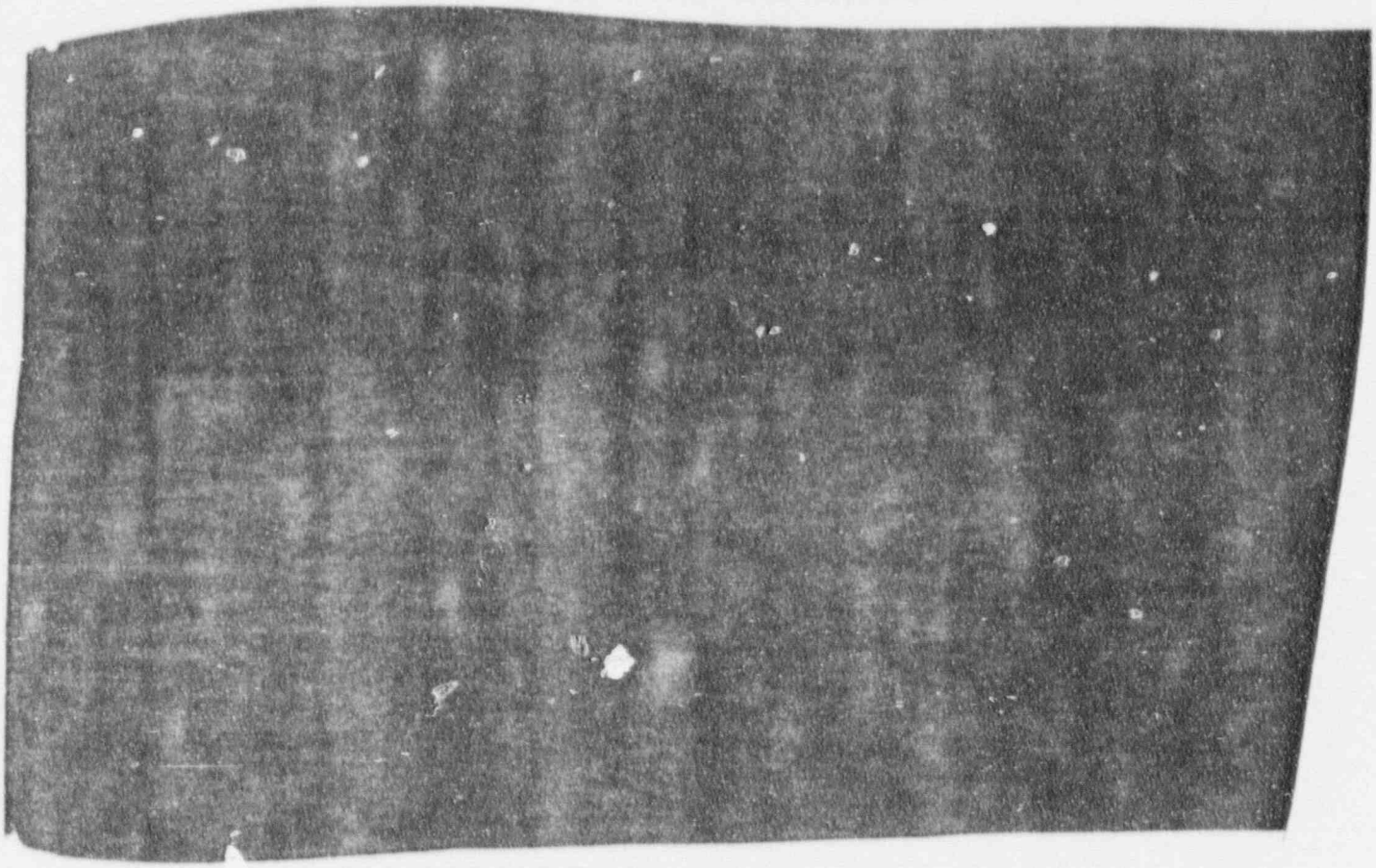
9. GE Nuclear Energy, "Elimination of Limit on Local Suppression Pool Temperature for SRV Discharge with Quenchers," NEDO-30832, Class I, December 1984.

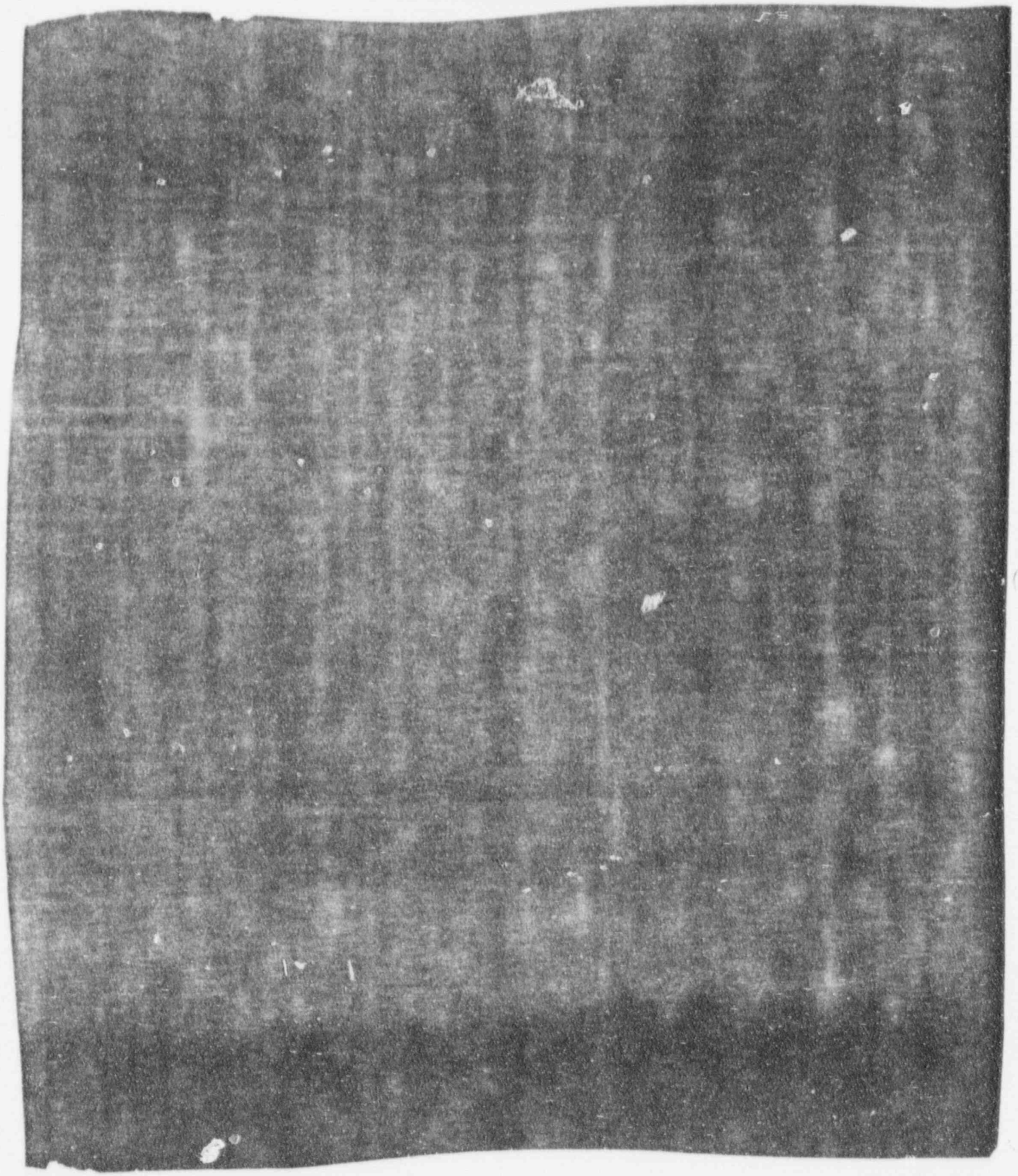
10. GE Nuclear Energy, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31984P, Class III (Proprietary), July 1991.

11. NUTECH Engineers, Inc., "Enrico Fermi Atomic Power Plant Unit 2, Plant Unique Analysis Report," Prepared for Detroit Edison Company, DET-04-028-1, Revision 0, April 1982.

12. GE Nuclear Energy," Fermi 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," NEDC-31982P, Class III (Proprietary), July 1991.

13. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Enrico Fermi Atomic Power Plant, Unit No. 2," NUREG-0798, July 1981.

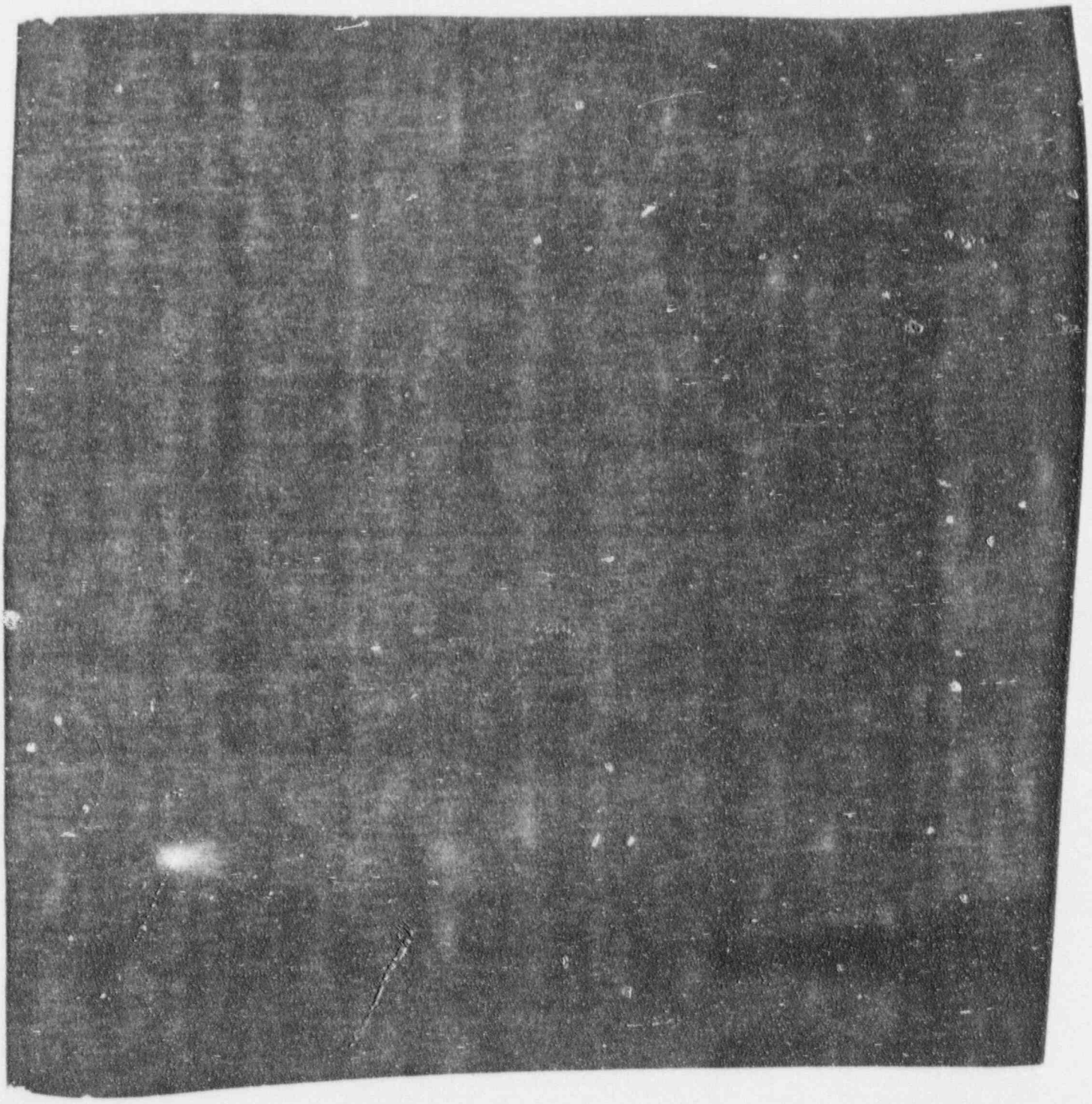




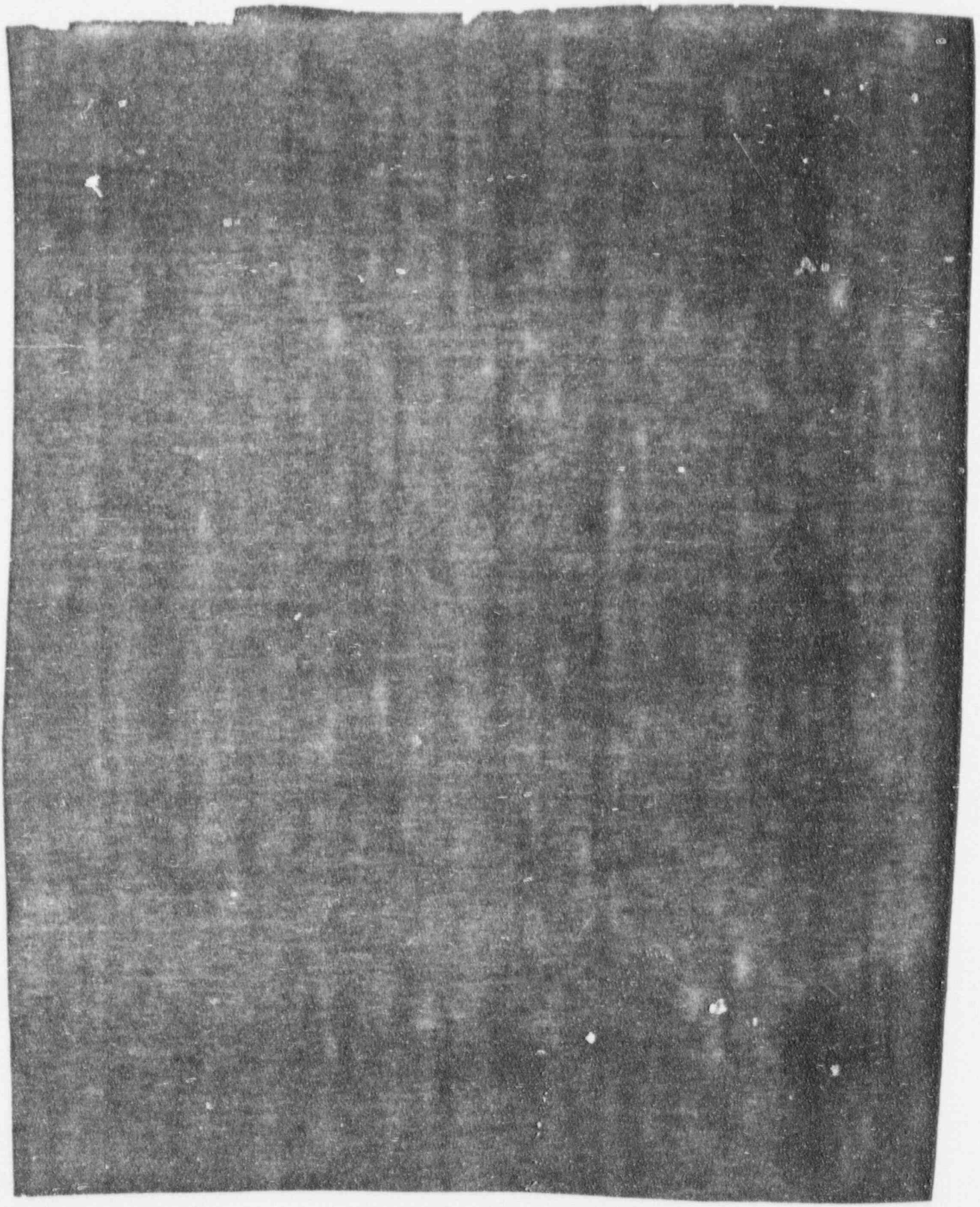
5.0 INSTRUMENTATION AND CONTROL

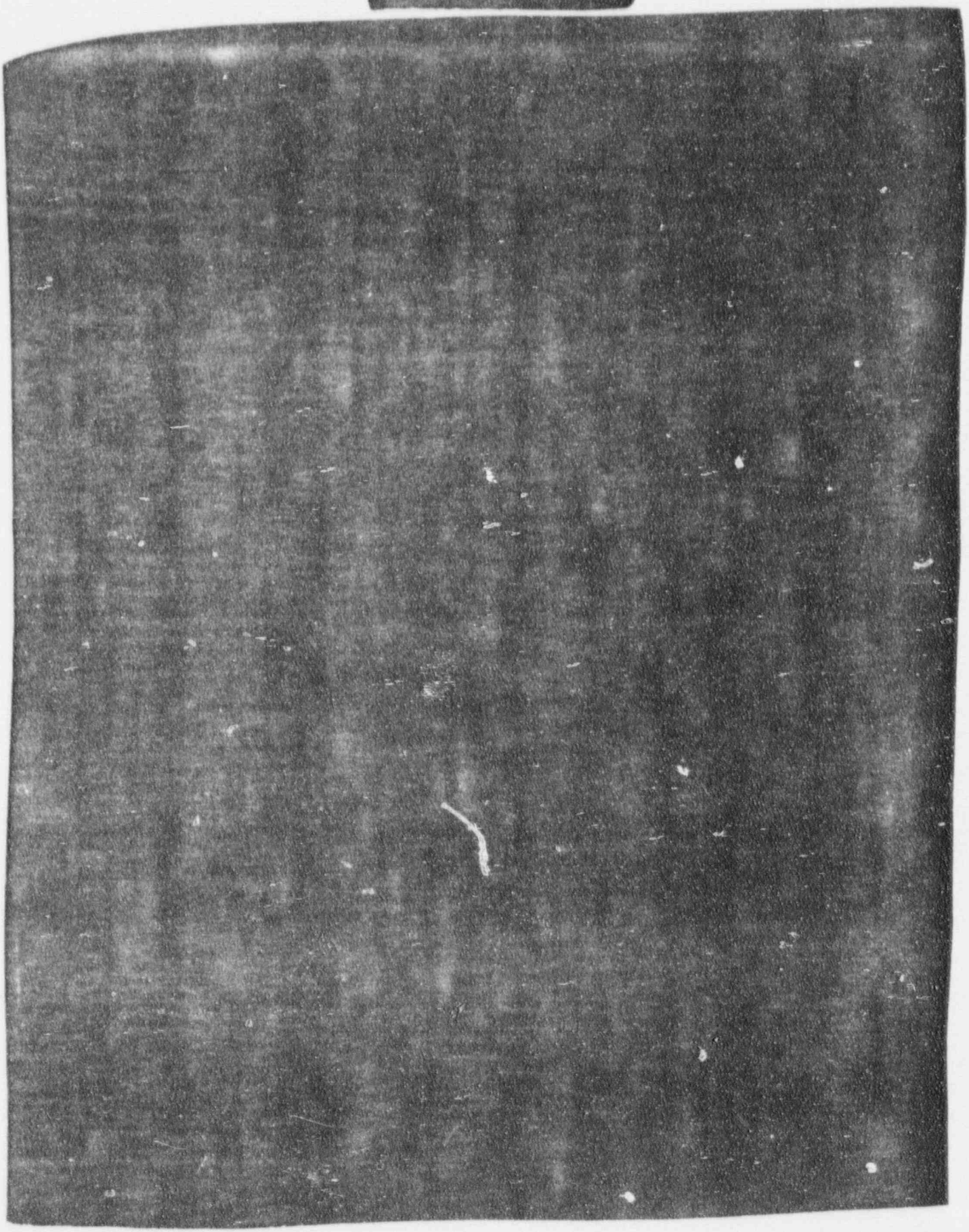
This section primarily focuses on the information requested in Regulatory Guide 1.70, Chapter 7, that applies to power uprate.

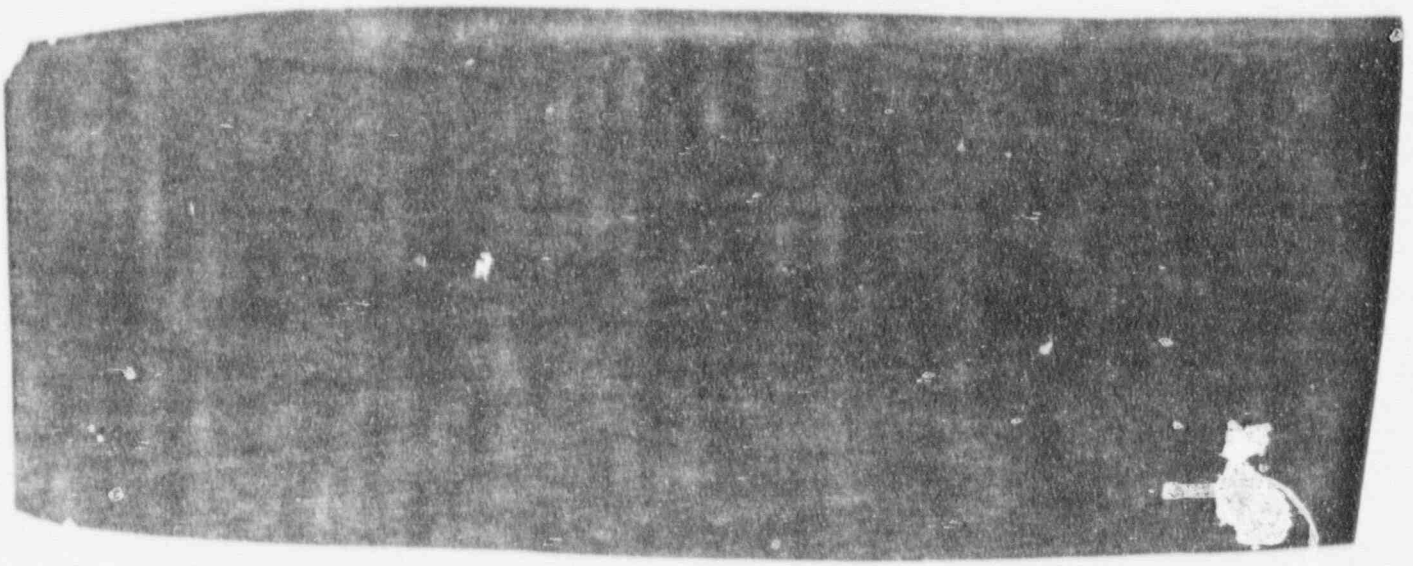
5.1 NUCLEAR STEAM SUPPLY SYSTEM



[redacted]







5.2 BALANCE-OF-PLANT - POWER CONVERSION AND AUXILIARY SYSTEMS

5.2.1 Control Systems Evaluation

Operation of the plant at the uprated power level has minimal impact on the BOP system instrumentation and control devices. Based on uprated operating conditions for the power conversion and auxiliary systems, the process control valves and instrumentation have sufficient range/adjustment capability for use at the expected uprated conditions.

5.2.1.1 Pressure Control System

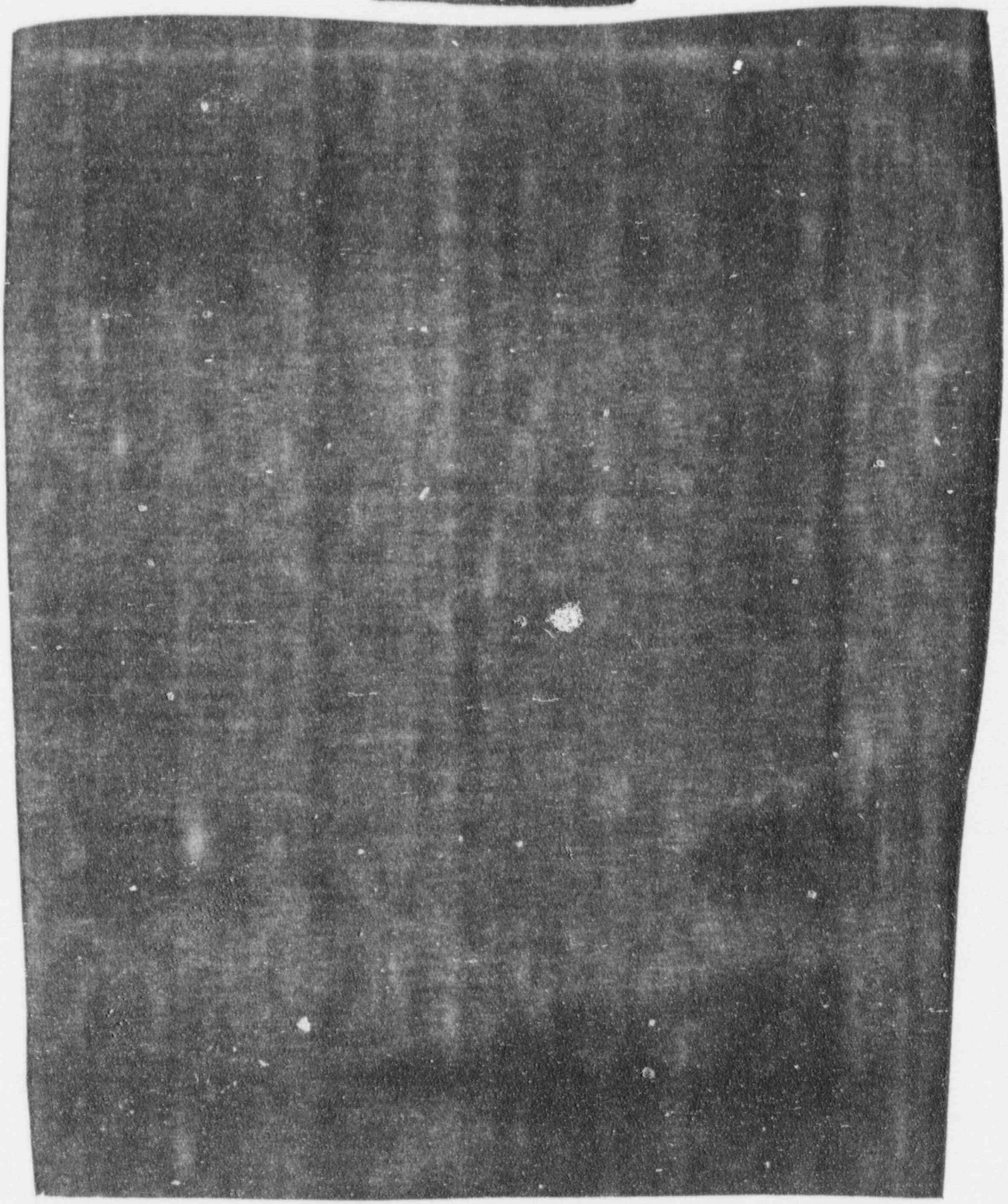
The objective of the pressure control system (PCS) is to provide a fast and stable response to pressure and steam flow disturbances in a manner that will control the reactor pressure within its allowed high and low limits. The PCS consists of the pressure regulation system, turbine control valve system, and steam bypass valve system. At uprated power conditions, the operating turbine inlet pressure becomes an important consideration in obtaining a desirable operating point on the turbine control valves. Adequate control valve range must be available to assure the ability of the pressure control system to respond to system disturbances demanding increased steam flow for the purpose of minimizing pressure excursions. Sufficient control pressure range during system disturbances is available with power uprate. Thus, the PCS is adequate for power uprate conditions.

5.2.1.2 Turbine Control System

No modification to the turbine control valves or the turbine bypass valves are expected for operation at the uprated throttle pressure conditions. No control problems associated with the increase in steam pressure and flow are expected for power uprate operation.

5.3 REFERENCES

1. GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31897P-1, Class III (Proprietary), June 1991.



6.0 ELECTRICAL POWER AND AUXILIARY SYSTEMS

This section primarily focuses on the information requested in Regulatory Guide 1.70, Chapters 8 and 9, that applies to power uprate.

6.1 AC POWER

6.1.1 Offsite Power System

Conformance to General Design Criteria 17 (10CFR50, Appendix A), which addresses onsite and offsite electrical supply and distribution systems for safety-related components, is not affected since there are no modifications associated with the power uprate which would increase electrical loads beyond those levels previously included or revise the logic of the distribution systems.

The isolated phase bus duct is adequate for rated current output at uprated power.

The main transformers and the associated switchyard components (rated for maximum transformer output) are adequate for the uprated transformer output.

6.1.2 Onsite Power Distribution System

The onsite power distribution system consists of transformers, numerous buses, and switchgear that support or are powered by the buses and switchgear.

Station loads under normal operation/distribution conditions are computed based on equipment nameplate data. Operation at the uprated level is achieved by utilizing existing equipment operating at or below the nameplate rating; therefore, under normal conditions, the electrical supply and distribution components (switchgear, MCCs, cables, etc.) are adequate.

Station loads under emergency operation/distribution conditions (emergency diesel generators - EDGs) are based on equipment nameplate data except for CS and RHR pumps where a conservatively high flow brake horsepower (BHP) is used. Operation at the uprated level is achieved by utilizing existing equipment operating at or below the nameplate rating and within the calculated BHP for the stated pumps; therefore, under emergency conditions the electrical supply and distribution components are adequate.

The amount of power required to perform safety-related functions (pump and valve loads) will not be increased with power uprate.

6.2 DC POWER

The DC loading requirements in the USAR were reviewed, and no power dependent loads were identified.

The DC power distribution system provides control and motive power for various systems/components within the plant. System loads are computed based on equipment nameplate data. Operation at the uprated level will not increase any loads beyond nameplate rating or revise any control logic; therefore, the DC power distribution system is adequate.

The RCIC modification discussed in Section 3.8 will add a small DC power valve to the system. This additional load will be evaluated during the design process; however, preliminary assessment indicates it will have very little impact on the DC power distribution system.

6.3 FUEL POOL COOLING

As a result of operation at the uprated power level, the spent fuel pool heat load will slightly increase.

The fuel pool cooling adequacy is determined by calculating the heat load generated by a full core discharge plus remaining spaces filled with used fuel discharged at regular intervals and calculating the bulk pool temperature (Table 6-1).

The power uprate analysis assumes an 18-month normal fuel cycle as the basis. Each reload will affect the decay heat generation in the spent fuel discharged from the reactor. The evaluation considered the expected heat load in the spent fuel storage pool at the uprated conditions, and confirms the capability of the fuel pool cooling system to maintain adequate fuel pool cooling.

This power uprate evaluation was performed on the fuel pool cooling system by reviewing the original design requirements and bases for the system and the current plant operating conditions as part of the power uprate program.

The maximum heat load in the fuel storage pool occurs immediately after a full core discharge. For this case, the heat load is greater than the total design capacity of the two fuel pool heat exchangers, even for current power conditions. The RHR system, however, is more than adequate to provide the needed additional cooling capacity until the heat load diminishes to within the capacity of the fuel pool heat exchangers alone. For normal refueling activity, the two fuel pool heat exchangers have capacity to maintain adequate temperature in the fuel storage pool for uprate conditions.

The loss of fuel pool cooling accident described in the USAR was evaluated for power uprate. The radiological consequences increase with power; however, they remain considerably smaller than 10CFR100 limits.

The normal radiation levels around the pool are expected to increase slightly primarily during fuel handling operation. This increase is acceptable and will not significantly increase the operational doses to personnel or equipment.

The fuel racks are designed for higher temperatures than are anticipated from power uprate. Therefore, the racks are acceptable for the higher local decay heat loads.

In summary, the changes are small and are within the design limits of the affected systems and components.

6.4 WATER SYSTEMS

6.4.1 Service Water Systems

6.4.1.1 Safety-Related Loads

The safety-related heat loads are rejected to one of the three safety-related service water systems. These systems are the emergency equipment service water system (EESW), the diesel generator service water system (DGSW) and the residual heat removal service water system (RHRSW), which are described below. All heat removed by these systems is rejected to the atmosphere via the ultimate heat sink (UHS), which includes the RHRSW cooling towers.

6.4.1.1.1 Emergency Equipment Service Water System

The EESW is designed to provide cooling water for the EECW during a loss of offsite power, high drywell pressure, or upon failure of the RBCCW. The EECW provides cooling to various pumps (such as the reactor recirculation, RHR, and CS pumps). The EECW heat loads also include the heat rejected by the drywell coolers and space coolers for such systems as RHR, SGTS, CS, RCIC, EECW, HPCI, and hydrogen recombiner. The EECW also removes heat from the Control Room A/C chiller condenser, the battery charger area, and the switchgear room. There is no increase in the design maximum heat load on the drywell coolers, since the original design loads were based on maximum equipment loads which were based on vessel and component temperature (575°F) greater than the anticipated operating temperatures after uprate (552°F). The design maximum heat loads of the other space coolers do not increase, due to the excess capacity in the original HVAC heat load design (Section 6.6).

6.4.1.1.2 Diesel Generator Service Water System

The diesel generator service water system is designed to provide cooling water for the emergency diesel generators during testing and emergency operation. No change in heat load is anticipated since no new or increased electrical loads are imposed on the EDGs.

6.4.1.1.3 Residual Heat Removal Service Water System

The following functions of the RHRSW system are affected to a minor degree by the uprate:

(a) Increased load from the RHR system when operating in the shutdown cooling mode

(b) Increased load from the RHR system when operating in the fuel pool cooling mode (backup system)

(c) Increased load from the RHR system when operating in the suppression pool cooling mode, following a postulated LOCA

Heat loads from all of these functions increase in direct proportion to the percentage increase in the power level. The RHRSW takes suction from the ultimate heat sink and returns water to the ultimate heat sink via the cooling towers.

The RHR cooling towers are designed to provide water with a temperature of 89°F (USAR Subsection 9.2.5) at design ambient conditions, with return water temperature as high as 116°F. The increased heat loads after uprate result in an RHRSW return temperature increase of less than 1°F. Since the uprated maximum return temperature is calculated to be 115°F no increase in RHRSW supply temperature results.

Use of the RHRSW for flooding purposes following a LOCA is not affected by the uprate.

6.4.1.2 Nonsafety-Related Loads

The service water discharge temperature results from the heat rejected to the service water system via the closed cooling water systems and other auxiliary heat loads. The major service water heat load increase from power uprate reflects an increase in main generator losses rejected to the stator water coolers, hydrogen coolers, and exciter coolers, in addition to increased bus cooler heat loads. The increase in service water heat loads from these sources due to uprated operation is projected to be approximately proportional to the uprate.

The general service water (GSW) system is capable of supplying sufficient water to remove the additional heat loads when operating at the uprated power level. GSW is drawn directly from Lake Erie; therefore, the uprate does not affect the inlet temperature to the supplied equipment. As designed, the system will experience a temperature rise of less than 1°F in the bulk outlet temperature which returns to the circulating water system and the cooling towers. This is insignificant to the design of the system.

6.4.2 Main Condenser/Circulating Water/Cooling Tower Performance

The main condenser, circulating water, and cooling tower systems are designed to remove the heat rejected to the condenser and thereby maintain adequately low condenser pressure as recommended by the turbine vendor. Maintaining adequately low condenser pressure assures the efficient operation of the turbine/generator and minimizes wear on the turbine last stage buckets. The cooling towers also remove heat rejected to the GSW system.

The performance of the main condenser was evaluated for the 1991 retubing. This evaluation is based on a design duty over the actual yearly range of circulating water inlet temperatures and confirms that the condenser, circulating water system, and cooling towers are adequate for uprated operation.

6.4.2.1 Discharge Limits

The state discharge limits shown in the National Pollution Discharge Elimination System permit were compared to the current discharges and bounding analysis discharges for power uprate. The comparison demonstrates that the plant will remain within the state discharge limit during operation at uprated power and no permit change will be required.

6.4.3 Reactor Building Closed Cooling Water System

The RBCCW system is designed to remove heat from the auxiliary equipment housed in the reactor building. The increase in this heat load due to the uprate is insignificant for all loads except fuel pool cooling. Fuel pool cooling requirements increase in proportion to the power level. However, the system was conservatively designed with design heat loads which bound those anticipated for operation at the uprated power level. Therefore, there is no impact to the system design.

6.4.4 Turbine Building Closed Cooling Water System

The TBCCW system provides cooling to both generator-related and nongenerator-related equipment in the turbine building. The generator-related equipment experiences an increase in load approximately proportional to the percentage of power uprate. However, the equipment heat loads provided by the generator manufacturer were

based on a generator rating of 1215 MWe, which is slightly higher than the uprated generator output; therefore, the design heat loads for the generator-related equipment bound the actual loads at the uprated conditions. The nongenerator related heat loads do not significantly increase with uprate. Therefore, the TBCCW system maximum cooling capacity, as shown in USAR Subsection 9.2.7, will not be exceeded during uprated operation.

6.4.5 Ultimate Heat Sink

As a result of operation at the uprated power level, the post-LOCA ultimate heat sink (UHS) water temperature increases, primarily due to higher reactor decay heat. This results in a higher evaporation rate and, therefore, a higher minimum water inventory requirement in the RHR reservoir.

A review was performed to evaluate the increased water inventory requirement and temperature. It was determined that the existing ultimate heat sink system will provide a sufficient quantity of water at a temperature less than 89° F (design temperature) following a design basis LOCA. The Technical Specification for RHR reservoir water level is adequate due to conservatism in the original water requirement calculations. Therefore, the UHS design is adequate for uprated operation.

6.5 STANDBY LIQUID CONTROL SYSTEM

6.6 POWER DEPENDENT HVAC

Operation of the plant at the uprated power level results in a maximum increase in process fluid temperatures of approximately 6°F in the steam cycle systems, and a maximum of 1°F in other auxiliary systems, with the exception of the fuel pool which increases approximately 2°F during its maximum loading. This results in higher heat gains from piping, piping supports, and mechanical equipment affected by the temperature increase.

The impact of the higher heat loads was evaluated in all affected areas: the Reactor, Auxiliary, and Turbine Buildings; drywell; areas with supplemental room coolers; areas with emergency room coolers; the Control Center; and the fuel pool area. This evaluation indicates that there is no real increase in calculated total heat gain for most areas of the plant due to high margins in calculated heat gain from pipe supports and high drywell piping temperatures used in the original design calculations. Therefore, there is no increase in maximum design temperatures for these areas.

Two areas in the Reactor and Auxiliary Buildings do have small increases in total heat gain. The RWCU heat exchanger room under normal conditions and the CRD pump room under LOCA conditions experience a total heat gain increase of 2 percent or less, which is negligible when considering inherent heat sinks and existing unit cooler margins. Expected operating temperatures for these areas remain within the maximum design temperatures of 125°F and 148°F degrees, respectively.

This evaluation concludes that the design heat loads in the original design calculations are adequate for operation at the uprated condition. Therefore, the design basis for the existing HVAC equipment and systems, the normal environmental temperatures used to establish qualified equipment life, and the area design temperature for all plant operating modes are not impacted by the uprate.

6.7 FIRE PROTECTION

Operation of the plant at the uprate power level does not affect the fire suppression or detection systems. There are no physical plant configuration or combustible load changes resulting from the uprate. The safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change, and are adequate for the uprated conditions. The operator actions required to mitigate the consequences of a fire are not affected. Therefore, the fire protection systems and analyses are not affected by plant uprate.

6.8 SYSTEMS NOT IMPACTED BY POWER UPRATE

6.8.1 Systems With No Impact

In addition to the systems listed in Table J-1 of Reference 1 and not addressed elsewhere, the following systems are not affected by operation of the plant at the uprated power level:

Torus Water Management System

Turbine Low Pressure Exhaust Spray Cooling System

Mechanical Vacuum Pumps Lube Oil and Seal Water System

Station Air Systems

Interruptible and Non-interruptible Control Air Systems

Emergency Breathing Air System

Auxiliary Boiler System

Turbine Oil Purification Transfer and Waste System

Diesel Generator Auxiliary Systems - Divisions I and II

Diesel Generator Fuel Oil Systems - Divisions I and II

Control Center Air Conditioning (Air and Water Side) Systems

Primary Containment Pneumatic Supply System

Nitrogen Inerting System

Main Steam Isolation Valve Leakage Control System

Seismic Monitoring

6.8.2 Systems With Insignificant Impact

In addition to the systems listed in Table J-2 in Reference 1, some systems are affected in a very minor way by operation of the plant at the uprated power level. For these systems, the effects are insignificant to the design or operation of the system and equipment:

Service Water Makeup, Decant and Overflow System-RHR Complex-Divisions I and II

RCIC Turbine Lube Oil System

HPCI Turbine Lube Oil System

Heater Feed Pump Lube Oil System

Reactor Feed Pump Turbine Lube Oil System

Standby Feedwater Lube Oil System

Turbine Gland Steam Sealing System

Generator Seal Oil System

Turbine Lubricating Oil System

Turbine Flange Heater System

Generator Gas System

Condensate Storage and Transfer System

Condensate Makeup Demineralizer System

Process Sampling Systems

Equipment Drains - All floors of Auxiliary and Reactor Buildings.

Post Accident Sampling System

Loose Parts Monitoring

6.9 REFERENCES

1. GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31897P-1, Class III (Proprietary), June 1991.

Table 6-1

FUEL POOL COOLING ANALYSIS

Parameters	Power Uprate
Maximum Bulk Pool Temperature Uprate (°F)	127
Structural Design Temperature (°F)	150
Bulk Pool Temper[illegible] ith Emergency Full Core Offloa[illegible] Capacity, with Su[illegible] RHR Cooling (°F)	125

[redacted]

7.0 POWER CONVERSION SYSTEMS

This section primarily focuses on the information requested in Regulatory Guide 1.70, Chapter 10, that applies to power uprate.

The power conversion systems were designed to utilize the energy available from the nuclear steam supply system and to accept the system and equipment flows resulting from continuous operation at 105% of rated steam flow.

The system operating pressures and temperatures anticipated for uprated conditions are shown in USAR Figure 10.1-2, the plant heat balance for operation at the original 105% warranted steam flow.

7.1 TURBINE/GENERATOR

The turbine/generator was purchased with a design capability for operating continuously at 105% of rated steam flow conditions. Both the turbine and generator were sized to allow operation at this higher flow with a degree of margin to allow control of important variables such as steam inlet pressure.

An engineering review of the turbine/generator was performed to provide additional assurance that adequate margin exists in the current design to allow successful operation at the higher power level without compromising reliability or equipment life.

7.2 CONDENSER AND STEAM JET AIR EJECTORS

The design of the mechanical vacuum pumps are not adversely affected by power uprate and no modification to the system is required. The physical size of the primary condenser and evacuation time are the main factors in establishing the capabilities of the vacuum pumps. These parameters do not change. Since flow rates will not change, there will be no change to the two-minute holdup time in the pump discharge line routed to the reactor building vent stack. The design capacity of the steam jet air ejectors will not be affected by the uprate since they originally were designed for operation at significantly greater than warranted flows.

7.3 TURBINE STEAM BYPASS

The turbine bypass system was originally designed for a flow equal to 25% of the original 105% warranted steam flow. The small pressure increase due to power uprate is within the original design capability of the turbine bypass system. Therefore this system is not affected by power uprate.

7.4 FEEDWATER AND CONDENSATE SYSTEMS

The feedwater and condensate systems are designed to provide a reliable supply of feedwater at the temperature, pressure, quality, and flow rate required by the reactor. These systems are not safety-related, however, their performance has a major effect on plant availability and capability to operate at the uprated condition.

For power uprate, an evaluation was performed to determine the capability of the condensate and feedwater system equipment to remain within the design limitations of the parameters listed below. The results showed that the design of the system equipment would not be affected by power uprate.

Pump NPSH
Ability to avoid suction pressure trip
Flow capacity
Bearing cooling capability
Rated motor horsepower
Full load motor amps
Vibration

Normal Operation

The condensate and feedwater systems were originally designed for 105% warranted steam flows. USAR Figures 10.1-1 and 10.1-2 show the heat balance at 100% and 105% warranted steam flow. Operation at the uprated power level does not significantly affect operating conditions of these systems. Sufficient margins exist between the calculated minimum pump suction pressures experienced and the low trip setpoint to provide adequate trip margins during steady-state conditions. The existing feedwater design pressure and temperature requirement are adequate.

The feedwater heaters and associated regulating valves were originally designed for 105% of warranted flow conditions and, therefore, are adequate for the uprated conditions.

Transient Operation

The plant response to a trip of a single reactor feedwater pump (RFP) is the same for power uprate as for current power conditions since the reactor will runback along the same load line to the same final operating conditions. The trip of a RFP is not expected to result in a reactor scram provided the heater drain system remains in operation.

7.4.1 Condensate Demineralizers

The impact of power uprate on the condensate polishing demineralizers (CPD) was reviewed. In summary, the system is adequate for uprate operation, but will experience a slight reduction in effectiveness.

The reduction in effectiveness will result in slightly reduced CPD run times. However, the reduced run times will be acceptable (refer to Section 8 for impact on radwaste systems).

7.4.2 Standby Feedwater System

The SBFW system was designed with two SBFW pumps with a combined nominal capacity of 1300 gpm and 1247 psig. This is not a safety-related system and is not required to support safe shutdown except for use as an alternate shutdown system during a fire (NRC SRP 9.5.1). The existing system can meet all of its design objectives. No change to this system is required.

8.0 RADWASTE SYSTEMS AND RADIATION SOURCES

This section primarily focuses on the information requested in Regulatory Guide 1.70, Chapters 11 and 12, that applies to power uprate.

8.1 LIQUID WASTE MANAGEMENT

The liquid radwaste system collects, monitors, processes, stores, and returns processed radioactive waste to the plant for reuse or for discharge.

The single largest source of liquid waste is from the backwash of the condensate demineralizers. With power uprate, the average time between backwash/precoat will be reduced slightly. This reduction does not affect plant safety.

The floor drain collector subsystem and the waste collector subsystem both receive periodic inputs from a variety of sources. Neither subsystem is expected to experience a significant increase in the total volume of liquid waste due to operation at the uprated condition.

The activated corrosion products in liquid wastes are expected to increase proportionally to the power uprate. However, the total volume of processed waste is not expected to increase appreciably since the only significant increase in processed waste is due to the more frequent backwashes of the condensate demineralizers. Based on a review of plant operating effluent reports and the slight increase expected from power uprate, it is concluded that the requirements of 10CFR20 and 10CFR50, Appendix I, will be met. Therefore, power uprate will not have an adverse effect on the processing of liquid radwaste.

8.2 GASEOUS WASTE MANAGEMENT

The gaseous waste systems collect, control, process, store, and dispose of gaseous radioactive waste generated during normal operation and abnormal operational occurrences. The gaseous waste management systems include the offgas system, SGTS (Section 4.4), and various building ventilation systems. The systems are designed to meet the requirements of 10CFR20 and 10CFR50, Appendix I.

Noncondensible radioactive gas from the main condenser systems, along with air inleakage, normally contains activation gases (principally ^{16}N, ^{19}O, and ^{13}N) and radioactive noble gas parents. The noncondensible gas is continuously removed from the main condenser by the steam jet air ejectors which discharge into the offgas system.

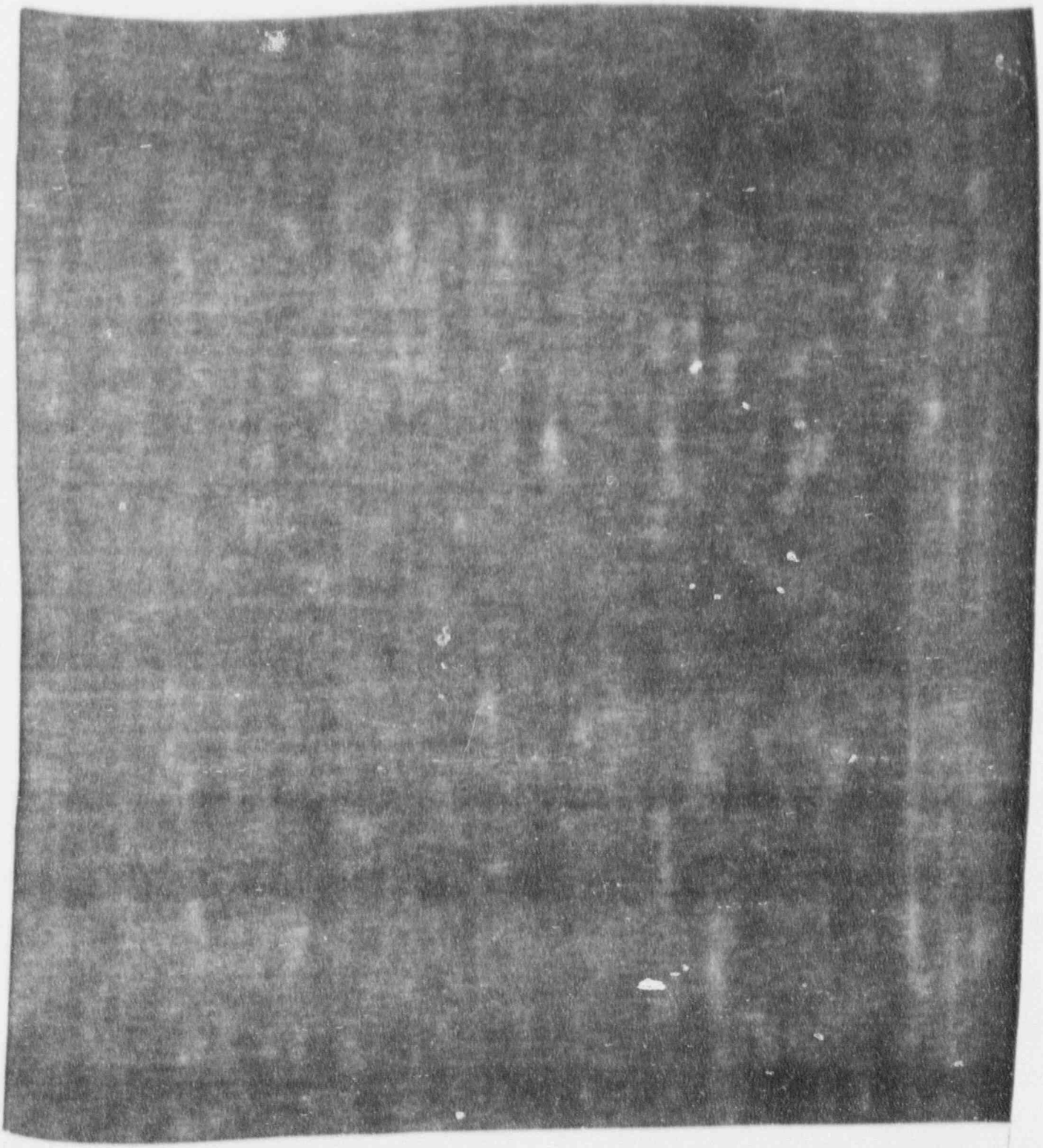
[redacted]

[redacted]

8.3 RADIATION SOURCES IN THE REACTOR CORE

[redacted]

8.4 RADIATION SOURCES IN THE COOLANT



8.5 RADIATION LEVELS

8.5.1 Normal Operation

The effect of power uprate on normal operation radiation levels was evaluated. The actual radiation levels in the plant are expected to increase slightly, primarily due to increased activation products. For conservatism, many aspects of the plant were originally evaluated based on higher design radiation sources and found to be acceptable. The effect of power uprate on plant radiation levels was reevaluated with the conservative assumption that the design radioactive source increase is proportional to the increase in power. The slight increase in calculated radiation levels as a result of the power uprate will not effect the radiation zoning or shielding in the various areas of the plant, since it is offset by conservatism in the original design, source terms used, and analytical techniques. This is demonstrated by the differences between the originally calculated and the actually experienced radiation levels.

8.5.2 Post-Operation/Accident

This section evaluates the effect of the power uprate on the post-operation (post-shutdown) and post-accident radiation levels. Due to the change in core inventory (Section 8.3), post-operation/accident radiation levels change slightly. The slight increase in the post-operation/accident radiation levels will have no impact on the design of the plant, the Technical Support Center (TSC), or the Emergency Operations Facility (EOF). Procedural control of the time spent in any radiation area compensates for increased radiation levels.

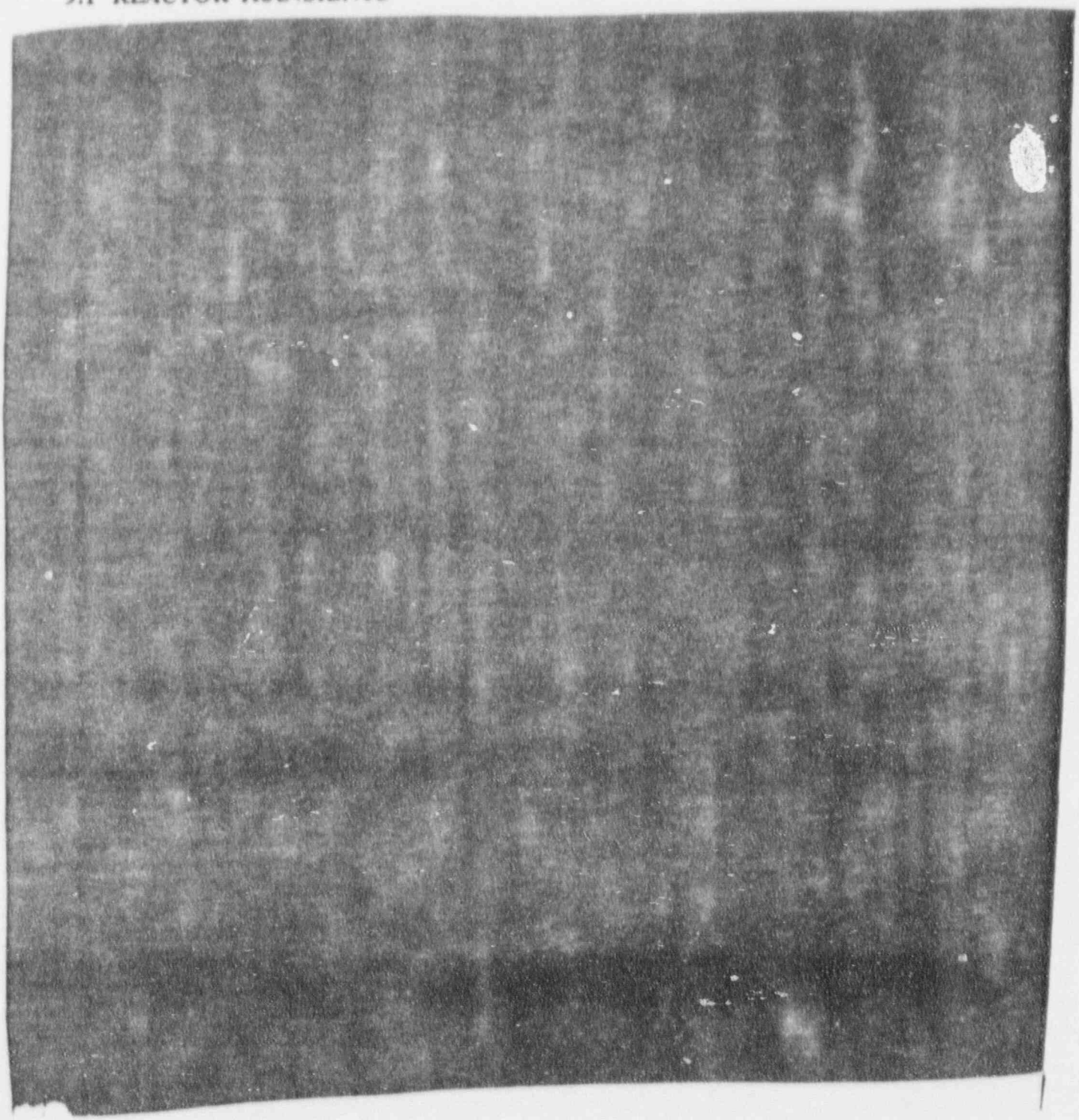
8.5.3 Offsite Doses (Normal Operation)

The normal offsite doses are not expected to be significantly affected by operation at the uprated power level. A review was performed to evaluate these effects based on a small general increase in releases. The calculated increase in the offsite radiological consequences, using conservative methods and assumptions, is well below any regulatory limits of 10CFR20 and 10CFR50, Appendix I.

9.0 REACTOR SAFETY PERFORMANCE EVALUATIONS

This section primarily focuses on the information requested in Regulatory Guide 1.70, Chapter 15, that applies to power uprate.

9.1 REACTOR TRANSIENTS



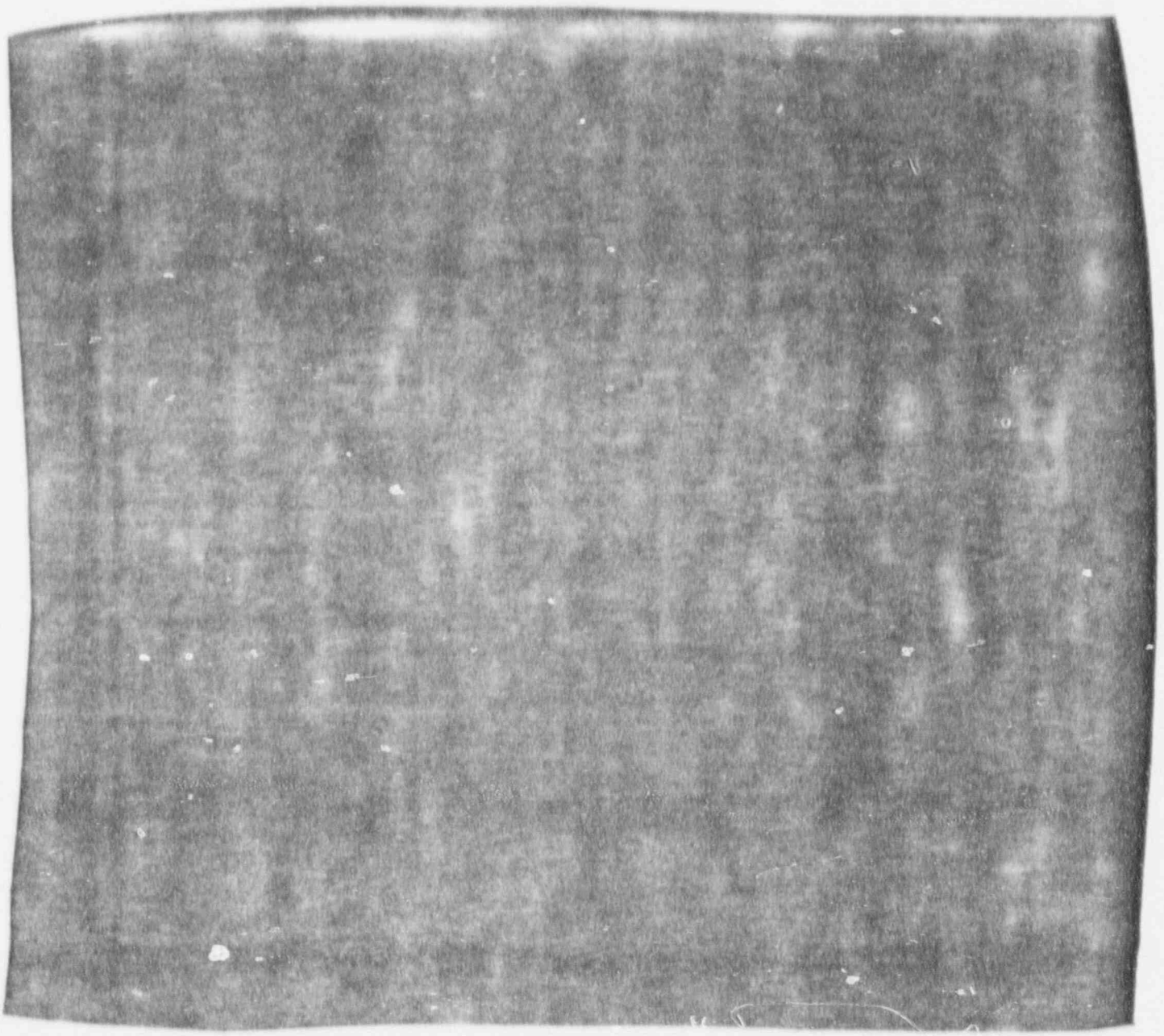
[redacted]

9.2 DESIGN BASIS ACCIDENTS

[redacted]

9.3 SPECIAL EVENTS

[redacted]

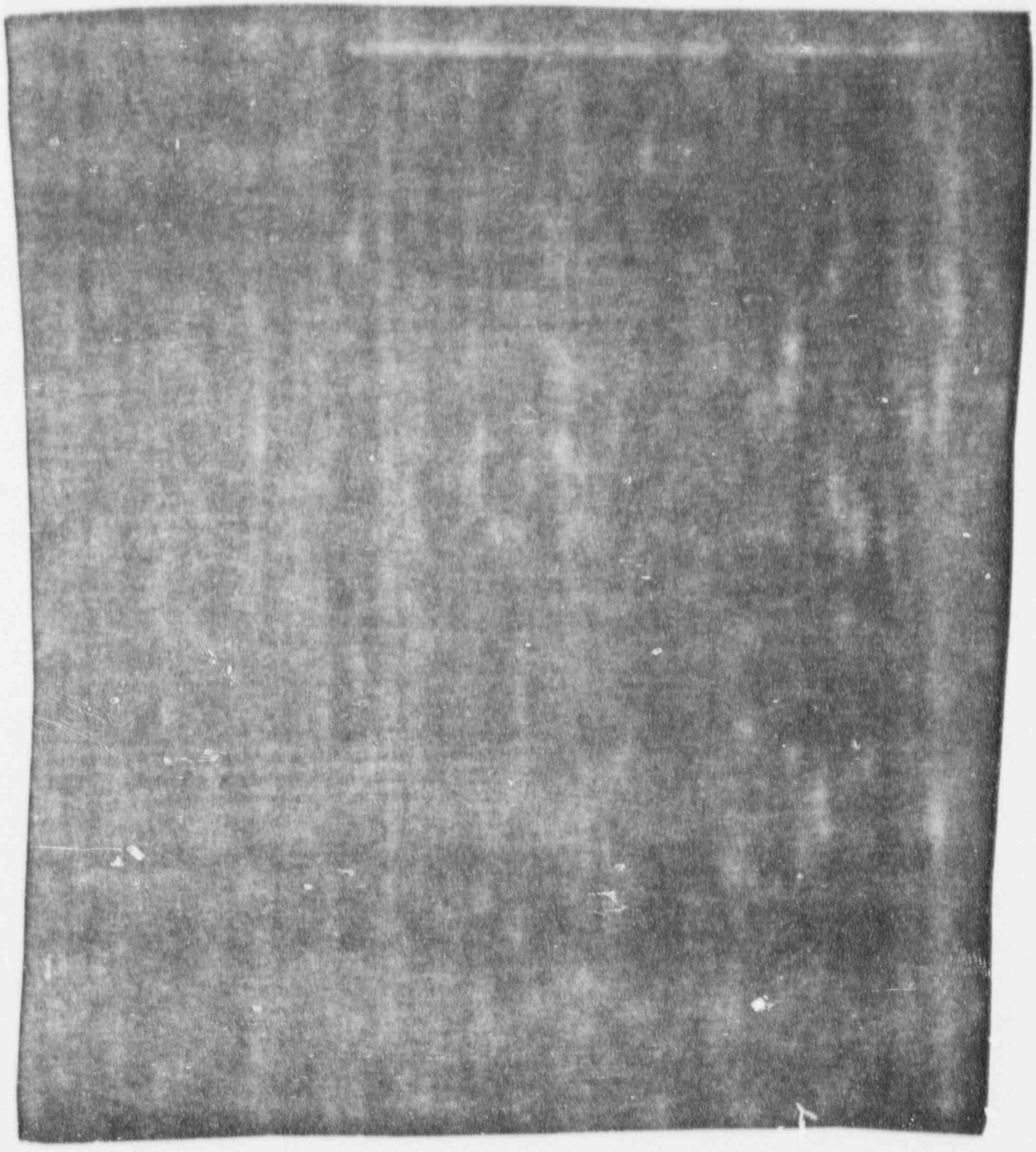


9.4 REFERENCES

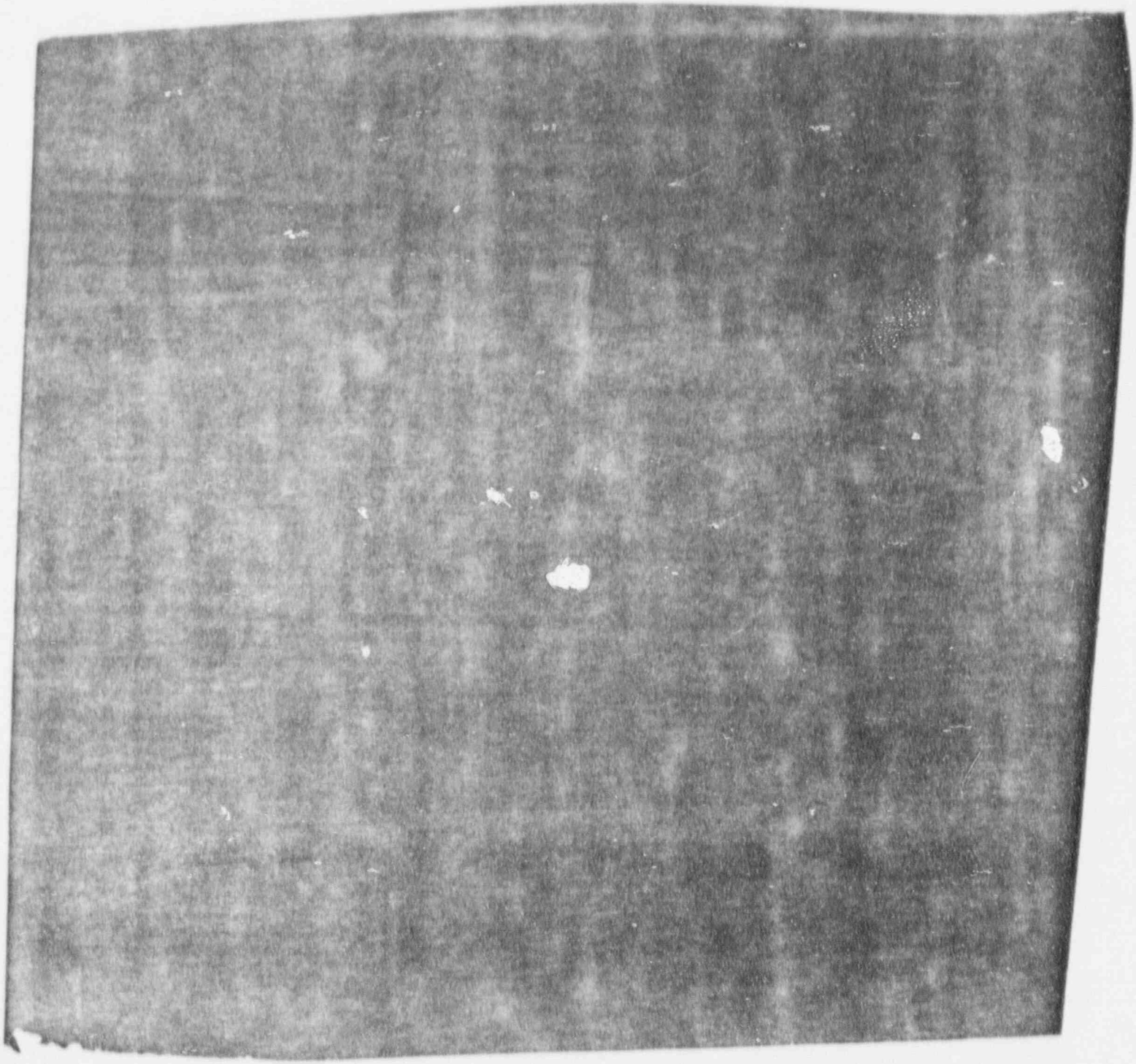
1. GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31897P-1, Class III (Proprietary), June 1991.

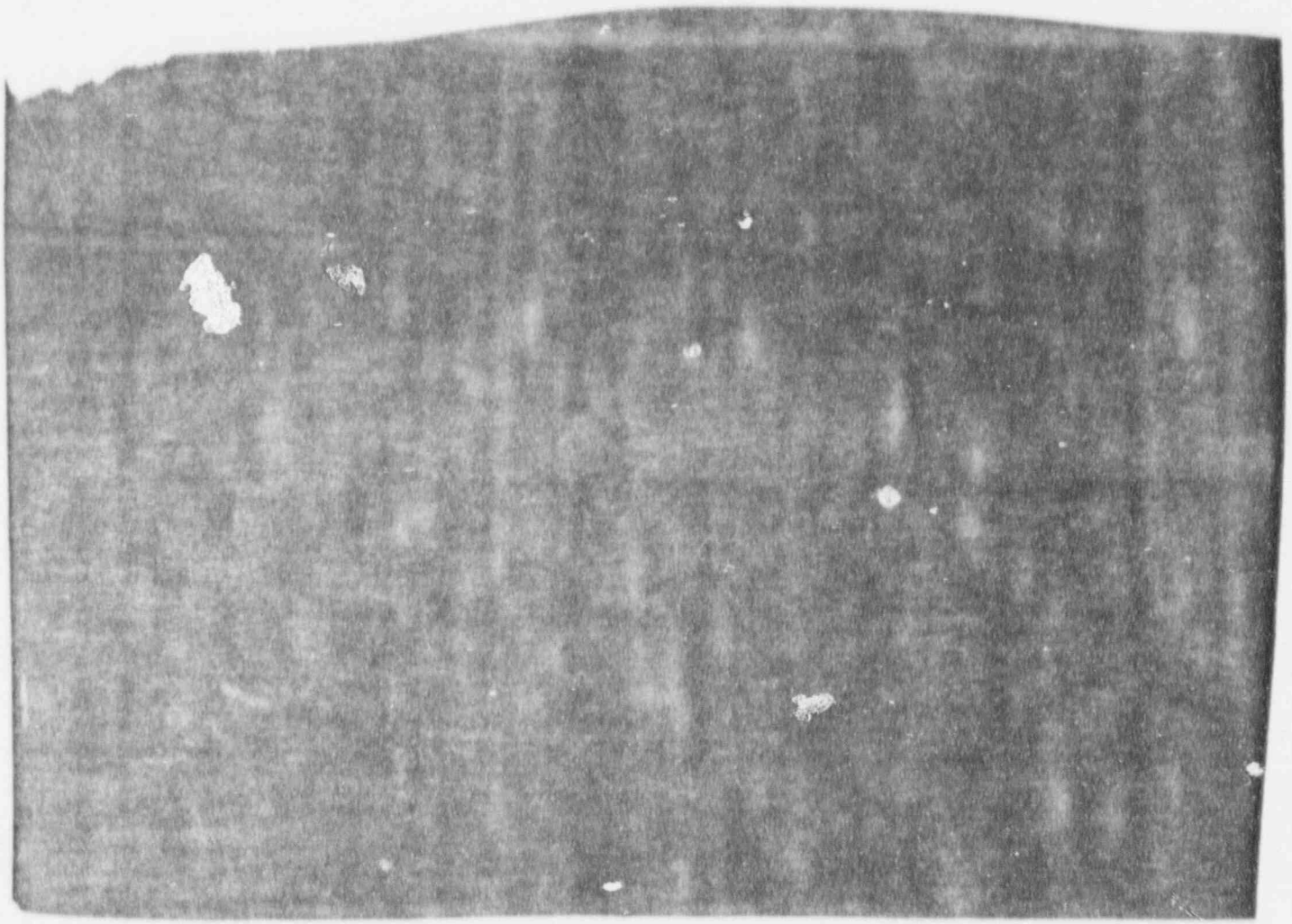
2. GE Nuclear Energy, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31984P, Class III (Proprietary), July 1991.

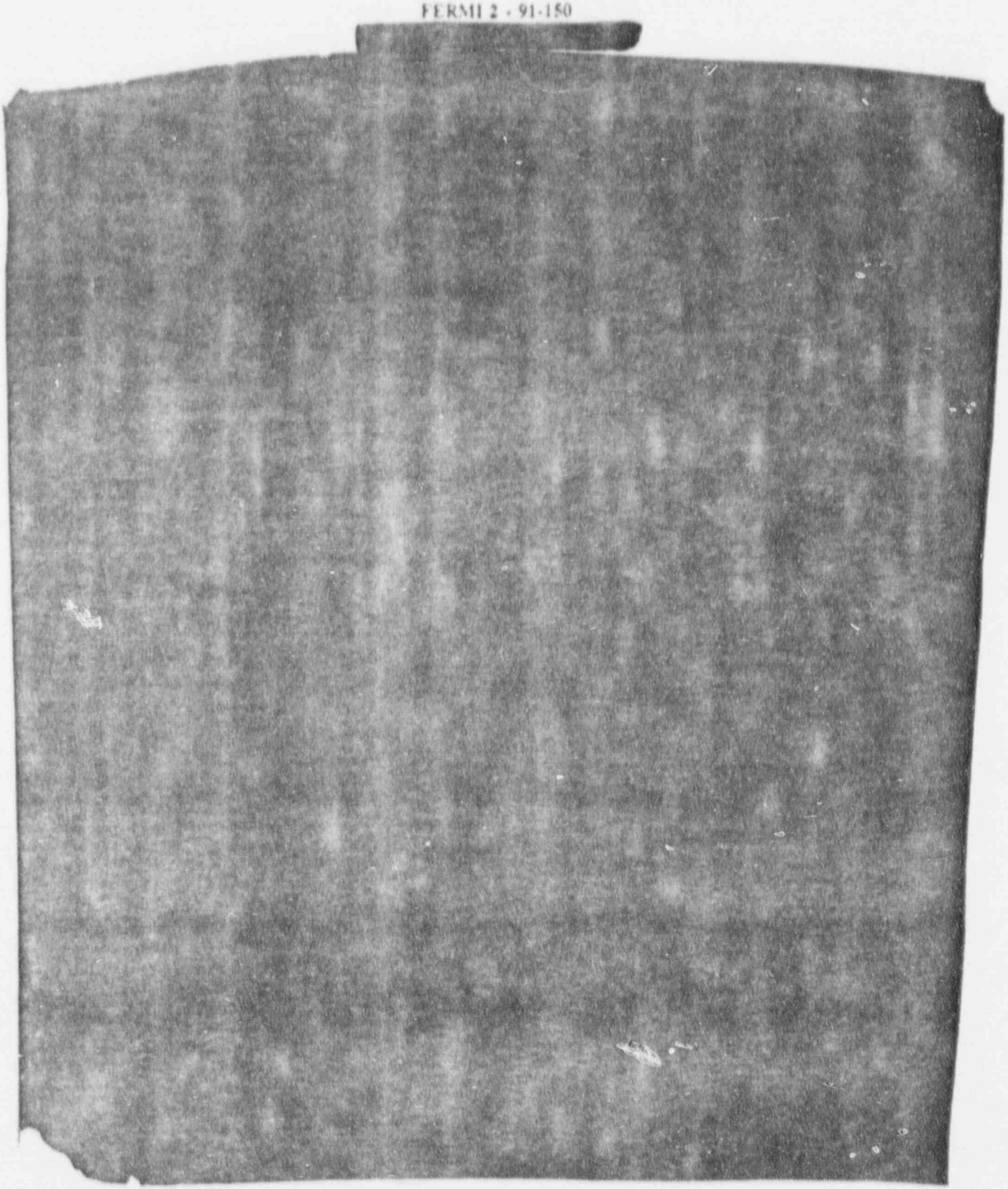
3. GE Nuclear Energy, "Supplemental Reload Licensing Submittal for Reload 2 Cycle 3," 23A6522, September 1990.

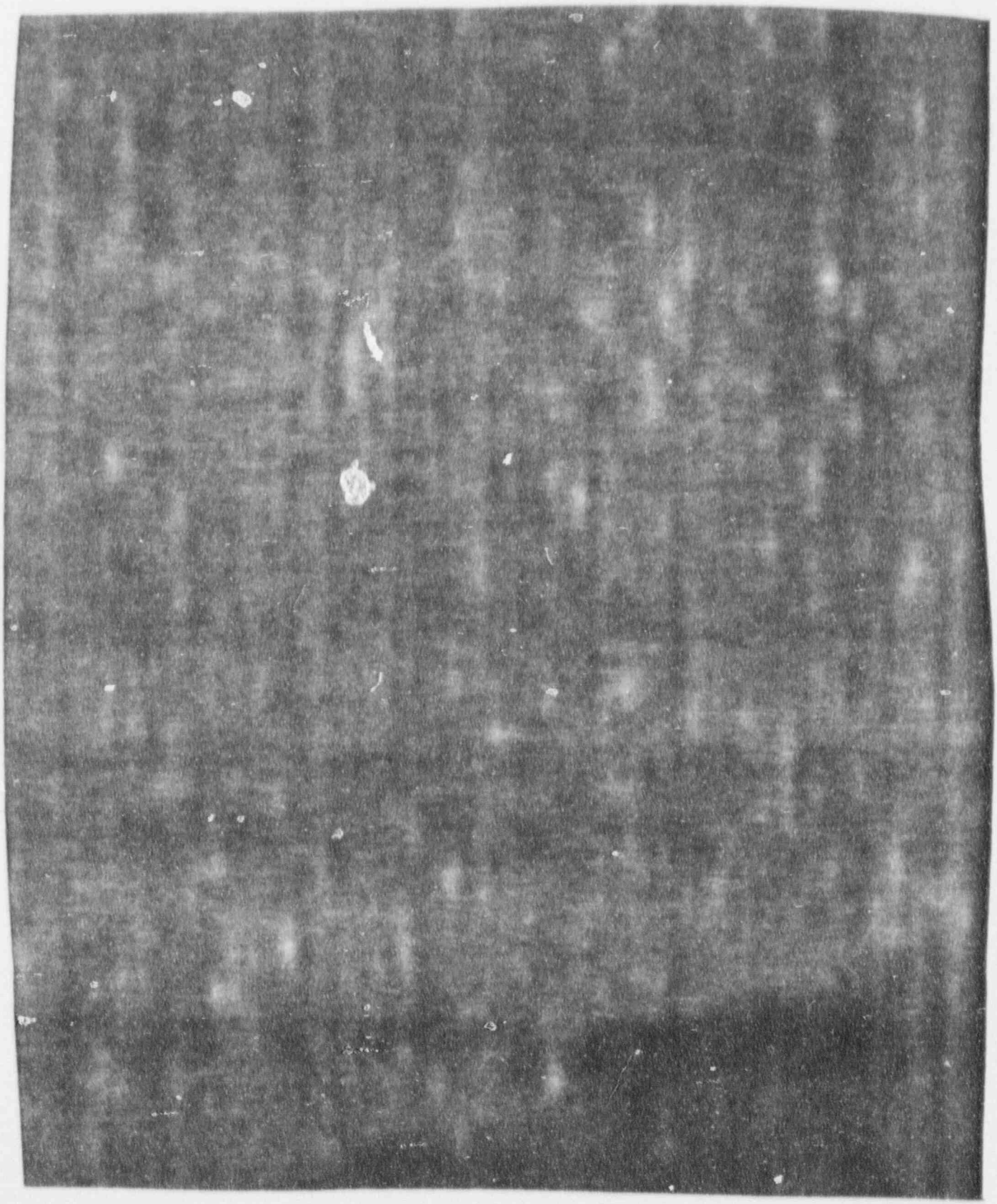


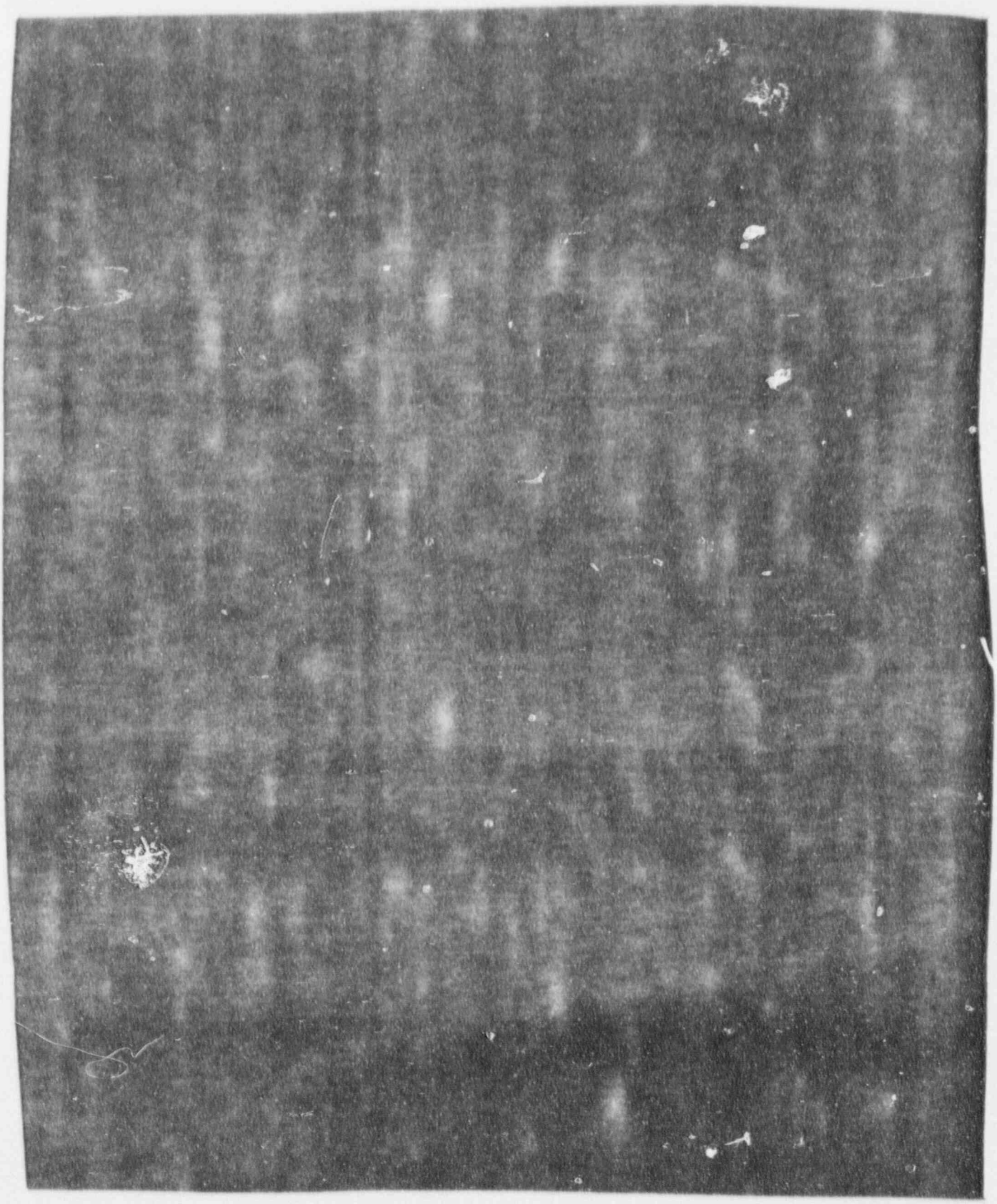
[redacted]

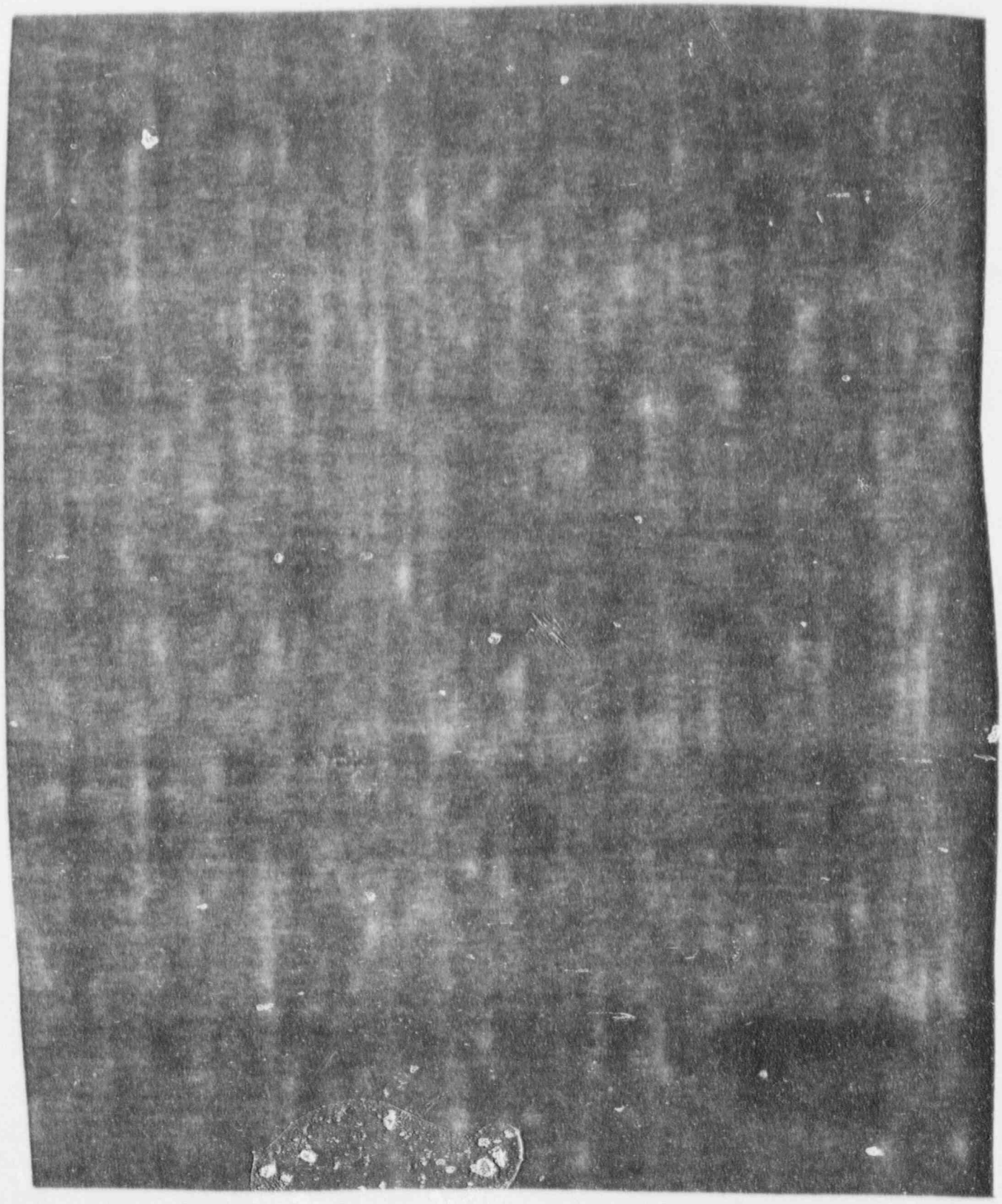


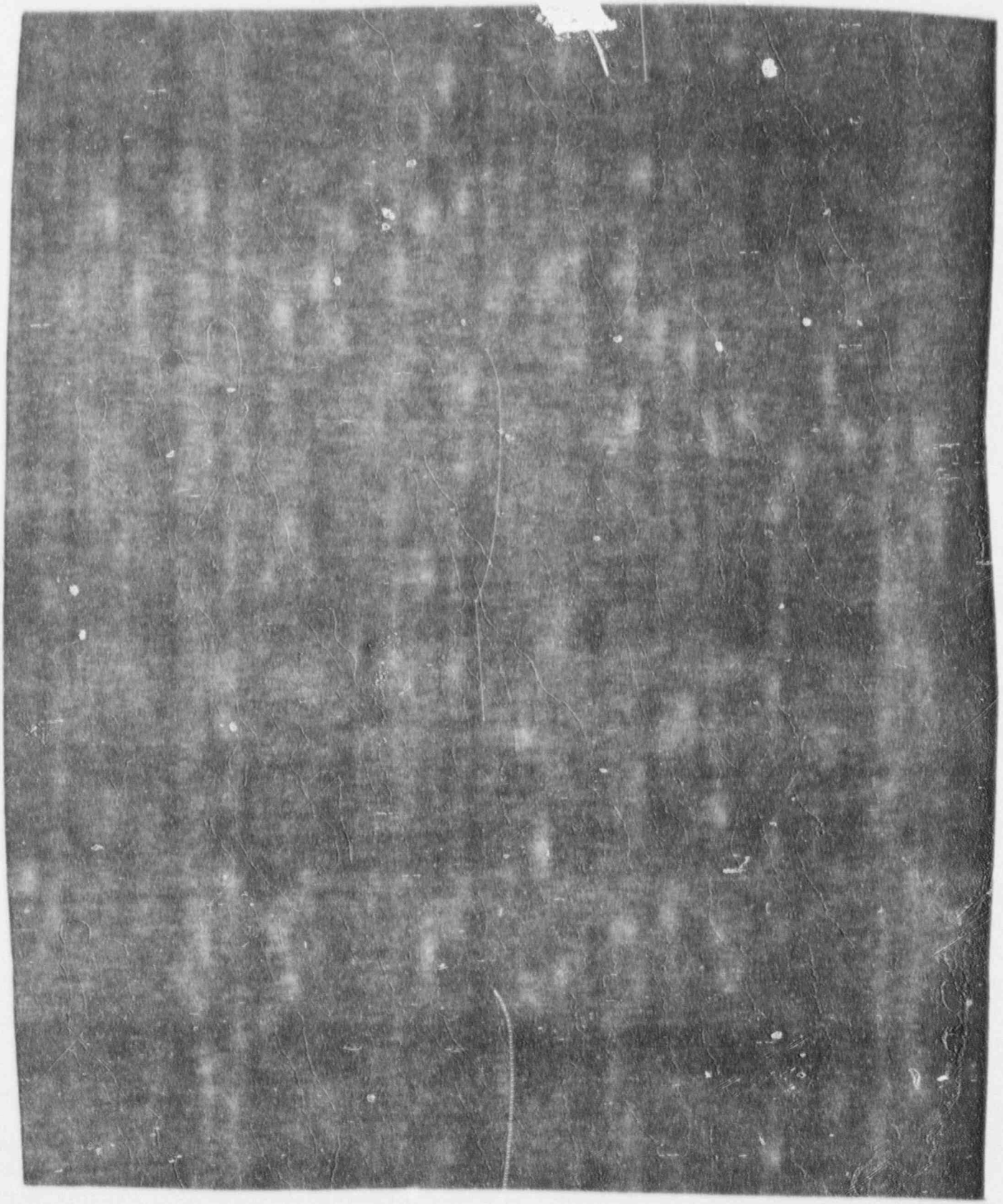




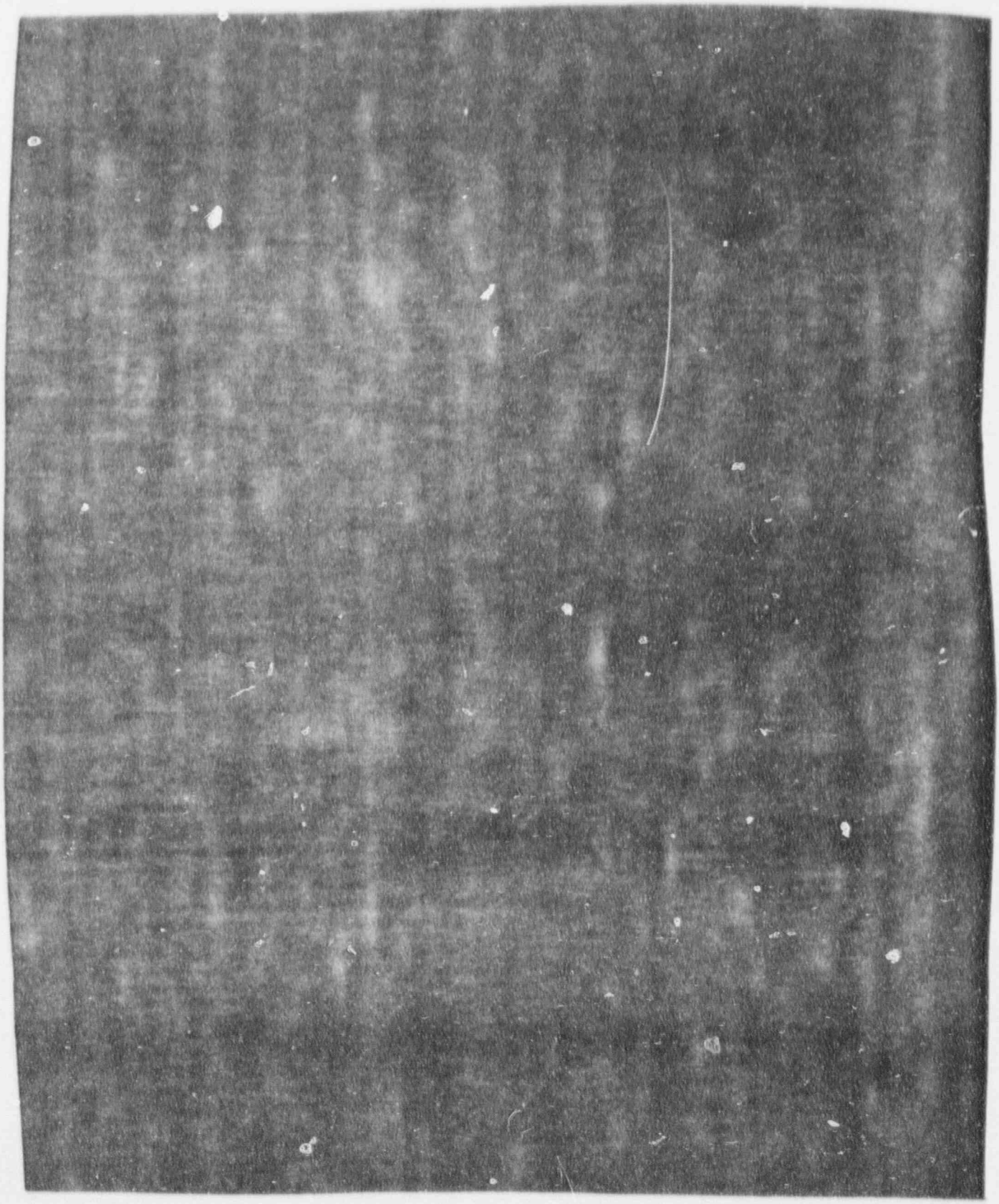








[redacted]



10.0 ADDITIONAL ASPECTS OF POWER UPRATE

10.1 HIGH-ENERGY LINE BREAK

Operation at an uprated power requires a small increase in the RPV dome operating pressure to supply more steam to the turbine. The slight increase in the vessel pressure and temperature results in a small increase in the mass and energy release rates following high-energy line breaks (HELB). Evaluation of these piping systems determined that there is no change in postulated break locations.

10.1.1 Temperature, Pressure, and Humidity Profiles

At the uprated power level, high-energy line breaks outside the primary containment cause the subcompartment pressure and temperature profiles to increase as shown in Table 10-1. The relative humidity change is negligible. In most cases, the increase in the blowdown rate is small and the resulting profiles are bounded by the existing profiles due to the conservatism in the original high-energy line break analyses, as discussed for each break.

The high-energy line break analysis evaluation was made for all systems evaluated in the USAR. The evaluation shows that the affected building and cubicles that support the safety-related function are designed to withstand the resulting pressure and thermal loading following a high-energy line break. The equipment and systems that support the safety-related function are also qualified for the environmental conditions imposed.

10.1.1.1 Main Steam System Line Break

The critical parameter affecting the high-energy line break analysis relative to the power uprate is the small increase in reactor vessel dome pressure. As a result of the power uprate, the blowdown rate increases by a small amount, and the differential pressure between subcompartments and peak temperature in the steam tunnel increases a very small amount. Profiles for surrounding areas are not significantly affected and are acceptable.

10.1.1.2 Feedwater System Line Break

A feedwater system line break is the critical case for flooding considerations. The flooding rate is dependent upon the feedwater system hardware (piping, pump, etc.), rather than the reactor vessel conditions. Since the feedwater system hardware is not changing, the existing feedwater break flooding analysis is valid for the uprated power condition.

10.1.1.3 High Pressure Coolant Injection System Line Break

The steamline break in the HPCI pump/turbine room is the limiting break for structural design and equipment qualification in the HPCI room. The peak pressure differential between rooms increases by an amount which is structurally acceptable. The temperature and relative humidity profiles in the original design are bounding for operations at the uprated power level.

10.1.1.4 Reactor Core Isolation Cooling System Line Break

The steamline break in the RCIC pump/turbine and torus rooms are the limiting breaks for structural design and equipment qualification in the RCIC room. The mass flow rate increases due to the increase in the reactor dome pressure relative to the original analysis. However, the original analysis was performed with conservative model assumptions. Inclusion of more realistic assumptions regarding pipe friction reduces the blowdown rate by more than enough to compensate for the uprate increase. Therefore the previous HELB analysis is bounding for the uprated core condition.

10.1.1.5 Reactor Water Cleanup System Line Break

The pump suction line break in RWCU pump rooms, pump holdup room, and the discharge line break in the torus room are the limiting breaks for structural design and equipment qualification. As a result of the slightly increased vessel pressure, the blowdown rate increases slightly. However, the original analysis was performed with conservative model assumptions. Conservatism regarding sub-cooled flow would more than offset the effect of increase in the pressure thus the original HELB analysis is bounding for the uprated core condition.

10.1.2 Pipe Whip and Jet Impingement

Calculations supporting the disposition of potential targets of pipe whip and jet impingement from the postulated HELBs have been evaluated and determined to be adequate for the safe shutdown effects in the uprated power conditions. Existing pipe whip restraints and jet impingement shields and their supporting structures are also adequate for the uprated conditions.

10.1.3 Moderate Energy Line Crack

Operation at an uprated level requires a small increase in the reactor vessel pressure during full power operation. The HPCI, RCIC, RWCU, and CRD systems have moderate energy piping. Of these, the CRD system is the only one that has the limiting crack in any plant area. The moderate energy line crack (MELC) evaluation for the CRD piping was originally done at the system design pressure, which is unaffected by the power uprate. Therefore the MELC evaluation for the plant is not impacted.

10.2 EQUIPMENT QUALIFICATION

10.2.1 Equipment Qualification of Electrical Equipment

The safety-related electrical equipment is evaluated in accordance with IEEE 323 to assure qualification for the normal and accident conditions expected in the area where the devices are located.

10.2.1.1 Inside Containment

Qualification for safety-related electrical equipment located inside the containment is based on environmental conditions resulting from a main steamline break and/or a DBA/LOCA, as well as those conditions expected during normal operation. The current accident and normal design conditions for temperature, pressure, and humidity are unchanged for power uprate. The radiation levels were increased in relationship to the increase in power.

Reevaluation of equipment qualification (EQ) has not identified any equipment as unqualified for the uprate environmental conditions. However, the qualified life of certain equipment will be reduced based on increased environments. The appropriate documentation to direct early replacement prior to exceeding its qualified life will be based upon aging analysis.

10.2.1.2 Outside Containment

Temperature, pressure, and humidity environments used for qualification of equipment outside containment result from a main steamline break in the pipe tunnel, DBA, or other high-energy line breaks, whichever is limiting for each plant area. The accident temperature, pressure, and humidity conditions resulting from a DBA/LOCA do not change with the power level, but some of the HELB profiles do increase by a small amount (Section 10.1). Maximum accident radiation levels used for qualification of equipment outside containment are from a DBA/LOCA.

The net result is the accident temperature, pressure, and radiation levels increase due to operation at the uprated power level. The accident temperatures increase less than 5° F, the pressures increase less than 1 psi. The radiation levels used in the evaluation were increased in relationship to power. The normal temperature, pressure, and humidity conditions do not change as a result of power uprate.

Reevaluation of EQ equipment has not identified any item as unqualified for the uprate environmental conditions. However, the qualified life of certain equipment will be reduced based on increased environments. The appropriate documentation to direct early replacement prior to exceeding its qualified life will be based upon aging analysis.

10.2.2 EQ of Mechanical Equipment with Non-Metallic Components

Operation at the uprated power levels increases the normal process temperature up to 6°F. The accident radiation level and the normal radiation level also increase slightly due to uprate and were evaluated as discussed in Section 10.2.1.

Reevaluation of EQ equipment has not identified any item as unqualified for the uprate environmental conditions. However, the qualified life of certain equipment will be reduced based on increased environments. The appropriate documentation to direct early replacement prior to exceeding its qualified life will be based upon aging analysis.

10.2.3 Mechanical Component Design Qualification

The mechanical design of equipment/components (pumps, heat exchangers, etc.) in certain BOP systems is affected by operation at the uprated power level due to slightly increased temperatures, pressure, and some cases, flow.

Increases to component nozzle loads and component support loads due to the revised operating conditions are insignificant, and become negligible when combined with the governing seismic loads. Therefore, the components and component supports are considered to be adequately designed for the uprated conditions.

10.3 REQUIRED TESTING

The testing for power uprate will be conducted based on Reference 1. Specifically the following testing will be performed as a result of power uprate at the time of implementation:

1. Normal design and construction acceptance testing will be performed on the RCIC one-inch steam bypass modification which is made to meet SIL 377 recommendations. Testing will be performed at power using the appropriate plant surveillance tests.

2. Surveillance testing will be performed on any instrumentation which requires recalibration as a result of power uprate.

3. Steady-state data will be taken at points from 90% up to the previous rated thermal power so that operating performance parameters can be projected for uprated power before the previous power rating is exceeded.

4. Power increases beyond the previous rating will be made along an established flow control/rod line in increments of 3% (or less). Steady-state operating data will be taken and evaluated at each step.

5. Control system tests will be performed for the feedwater/water level controls and the pressure controls. These operational checks will be performed at the previous rated power condition and at each 3% power increment to show acceptable adjustment and operational capability. The same performance criteria shall be used as in the original power ascension tests.

As discussed in Reference 1, large transient tests will not be performed since this uprate is less than 5%. Initial plant testing and experience during plant operation is considered to be sufficient.

10.4 REFERENCES

1. GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31897P-1, Class III (Proprietary), June 1991.

Table 10-1

HIGH-ENERGY LINE BREAK

Break Location	Increase Due to Uprate: Blowdown Rate	Increase Due to Uprate: Area Δ Pressure	Increase Due to Uprate: Area Δ Temperature
Main steamline break in steam tunnel	1%	0.15psi	≤ 1°F
HPCI ECCS break in pump room	4%	< 0.10psi	NC
RCIC break in pump room or torus room	NC	NC	NC
RWCU break in pump room, holdup room, or torus room	NC	NC	NC

NC = No Change

11.0 LICENSING EVALUATIONS

11.1 EVALUATION OF OTHER APPLICABLE LICENSING REQUIREMENTS

11.1.1 Generic Communications

The analysis, design, and implementation of power uprate was reviewed for compliance with the original licensing basis and with new regulatory requirements and operating experience in the nuclear industry. A generic review of the BWR power uprate program for compliance with regulatory requirements from the sources and industry recommendations listed below is provided in Reference 1. Reference 1 also summarizes the issues resulting from the generic review that require plant - specific review. The sources listed below were reviewed and the review documented for Fermi 2 to ensure that the issues involved were incorporated, as applicable, in the power uprate evaluation.

Regulatory Guides

Generic Letters

TMI Action Items

Action Items (Formerly Unresolved Safety Issues)

New Generic Issues

Information and Enforcement Circulars (IEC)

Inspection and Enforcement Notices (IENs)

Inspection and Enforcement Bulletins (IEBs)

GE Service Information Letters (SILs)

GE RAPID Information Communication Service Information Letters (RICSIL)

INPO Significant Operating Experience Reports (SOER)

11.1.2 Plant Unique Items

Four types of items whose previous evaluation could be impacted by operation at the uprated power level were also identified. These are Deviation Event Reports (DER); Regulatory Action Commitment Tracking System (RACTS) items; Safety Evaluations (SE)

for work in progress and not yet integrated into the plant design basis; and Temporary Modifications (TM) which are not permanent changes, but might be in effect prior to power uprate and still exist after uprate implementation. The DERs, RACTS items, and SEs have been reviewed for possible impact by power uprate and were found to be acceptable for uprate.

11.1.2.1 Deviation Event Reports

Deviation Event Reports are initiated to document plant problems that require evaluation and resolution. The subject of each open DER was reviewed to determine whether it could potentially be impacted by power uprate. Those items that could have been impacted were further evaluated by review of the actual DER. The following criteria formed the basis for the conclusion that the DERs were not impacted by power uprate:

(a) The subject did not impact the power generation capability of the plant.

(b) The subject did not impact systems or components important to safety whose operating performance requirements would be impacted by power uprate.

(c) The subject did not impact pipe stress, transient, or accident analyses affected by the power uprate.

11.1.2.2 Regulatory Action Commitment Tracking

Regulatory Action Commitment Tracking System (RACTS) commitments are those made to the NRC, usually by letter. Documents from the NRC that require evaluation and response or corrective action are assigned RACTS item numbers. All RACTS commitments were reviewed using the same criteria as that used in the DER review.

11.1.2.3 Safety Evaluations

Safety Evaluations that are approved, but not included in the USAR, Revision 4, March 1991, or in the 1990 annual Safety Evaluation Summary Report, prepared at the same time, were reviewed. The subject of each SE was evaluated to determine whether it could potentially be impacted by power uprate. The SEs were evaluated against the criteria used for DERs in section 11.1.2.1.

11.1.2.4 Temporary Modifications

Temporary Modifications that are in effect at the of time of power uprate implementation will be reviewed and revised, if necessary, to include uprated conditions.

11.1.3 Emergency Operating Procedures

The plant Emergency Operating Procedures (EOP) will be reviewed for any effects of power uprate and will be updated, as necessary. This review will be based on Reference 1, Section 2.3.

11.2 IMPACT ON TECHNICAL SPECIFICATIONS

Implementation of power uprate will require revision of a number of the Technical Specifications. Table 11-1 contains a list of the sections that have been identified for the effects of power uprate and a brief description of the nature of the effect.

11.3 ENVIRONMENTAL ASSESSMENT

Environmental evaluations were performed to determine that power uprate would not involve an unreviewed environmental issue and would meet the criteria established for a categorical exclusion not requiring an environmental review.

The non-radiological environmental impacts of the proposed power uprate were reviewed based on the information submitted in the Environmental Report, Operating License Stage (ER/OL); the NRC Final Environmental Statement (FES); and the requirements of the Environmental Protection Plan (EPP), Section 3.0 (Appendix B to the Operating License). Based on this review, it was concluded that the proposed uprate will have insignificant impacts on the non-radiological elements of concern and the plant will be operated in an environmentally acceptable manner as established by the FES. Existing Federal, State, and local regulatory permits presently in effect will accommodate power uprate without modification. Impacts to air, water, and land resources will be essentially non-existent.

The effect of power uprate on radiological effluents or offsite doses, as evaluated in the ER/OL and the FES, is insignificant since the original analyses were based on 104.2% (3430 MWt) of licensed power (3293 MWt). The analyses for power uprate were performed at 102% of uprated power, resulting in a calculated increase of approximately 2% in effluents and doses, still well within 10CFR20 and 10CFR50, Appendix I, limits.

A slight increase in occupational radiation exposures may occur due to the slight increase in radiation levels in some areas of the plant, primarily due to increased activation products. Conservative assumptions were used; the design radiation source increase is proportional to the increase in power. Even with this assumption, neither individual nor cumulative occupational radiation exposure will be significantly increased.

The proposed power uprate does not require a change to the Environmental Protection Plan or constitute an unreviewed environmental question since it does not involve:

(a) A significant increase in any adverse environmental impact previously evaluated in the final environmental statement, environmental impact appraisals, or in any decisions of the Atomic Safety and Licensing Board; or

(b) A significant change in effluents or power level; or

(c) A matter not previously reviewed and evaluated in the documents specified in Items (a) which may have a significant adverse environmental impact.

The evaluations also establish that power uprate qualifies for a categorical exclusion not requiring an environmental review in accordance with 10CFR51.22(c)(9) because it does not:

(a) Involve a significant hazards consideration, or

(b) Result in a significant increase in the amounts of any effluents that may be released offsite, or

(c) Result in a significant increase in individual or cumulative occupational radiation exposure.

11.4 SIGNIFICANT HAZARDS CONSIDERATION ASSESSMENT

11.4.1 Introduction

Uprating the power level of nuclear power plants can be done safely within certain plant specific limits up to as much as 40% power. Many light water reactors have already been uprated world wide; many MWe have already been added by uprate in the United States among which are several BWRs.

Modification Summary

An increase in electrical output of a BWR is accomplished primarily by supplying higher steam flow to the turbine/generator. Most BWRs, as originally licensed, have an as-designed equipment and system capability to accommodate steam flow rates at least 5% above the original rating. In addition, continuing improvements in the analytical techniques (computer codes) based on several decades of BWR safety technology, plant performance feedback, and improved fuel and core designs have resulted in a significant increase in the difference between calculated safety analysis results and the licensing limits. This available

safety analysis difference, combined with the excess as-designed equipment, system, and component capability, provide GE BWRs with the potential for an increase in their thermal power rating of between 5% and 10% without major hardware modifications and with no significant increase in the hazards presented by the plant as approved by the NRC at the original license stage.

The strategy for achieving higher power is to expand the power/flow map by increasing core flow along existing flow control lines. However, there will not be an increase in the maximum recirculation flow limit over the pre-uprate value. Uprated operation will also involve slightly higher reactor vessel dome pressure to provide adequate inlet pressure conditions at the turbine, accounting for the larger pressure drop through the steamlines at higher flow, and providing sufficient pressure control and turbine flow capability.

Specifically, the following modifications have been identified as necessary to accommodate uprated power operation at Fermi 2:

(a) The safety/relief valve setpoints are increased by 25 psig.

(b) Power, pressure, and steam-feedwater-flow related instrument recalibrations and/or related setpoint changes are required.

(c) A one-inch bypass valve will be installed on the RCIC system inlet steam, as recommended by General Electric SIL 377, to reduce the tendency of the RCIC turbine to overspeed on startup due to the higher operating pressure.

(d) The standby liquid control system sodium pentaborate storage tank high and low level alarms and limits are modified to ensure that an equivalent boron concentration of 900 ppm, including a 25% safety margin, can be achieved. This increases the core reload design flexibility while providing adequate shutdown capability.

11.4.2 Discussion of Issues Being Evaluated

Background

This assessment evaluates the responses for a power uprate license amendment. It is the intention of this safety assessment to summarize the safety significant plant reactions to licensing events and thereby conclude that no significant hazards consideration will be involved.

[redacted]

Uprate Analysis Basis

Fermi 2 is currently licensed at 3293 MWt (100% power). The original safety analysis was based on the reactor operating continuously at a power level at least 1.02 times the licensed power level. The uprated power level included in this evaluation is 3430 MWt, approximately 104.2% with 105% steam flow. Several of the original analyses have already been performed at 104.2% power, including the overpressurization analysis, ECCS/LOCA, and DBA/LOCA. A 2% power uncertainty factor has been included in the initial conditions used for the analyses at uprated power level. Therefore, as described herein, the plant has been reanalyzed at a power level of at least 1.02 times the uprated power rating.

Margin

The above uprate analysis basis assures that the margins prescribed by the Code of Federal Regulations are maintained by meeting the appropriate regulatory criteria. NRC accepted computer codes and calculational techniques are used to make the calculations that demonstrate meeting the stipulated criteria. Similarly, margin specified by application of the American Society of Mechanical Engineers (ASME) design rules has been maintained, as have other margin-assuring criteria used to judge the acceptability of the plant.

Fuel Thermal Limits

No change is required in the basic fuel design to achieve the uprated power levels or to meet the plant licensing limits [redacted] The fuel operating limits such as maximum average planar linear heat generation rate and operating limit minimum critical power ratio will still be met at the uprated power level. Reload analyses will continue to meet the criteria accepted by the NRC as specified in Reference 2. New fuel designs will meet acceptance criteria approved by the NRC. As an example, GE fuel will meet the criteria accepted by the NRC as specified in Reference 3.

LOCA Evaluations

The BWR design concept includes various means of supplying water to the reactor. The basic method to deal with all Loss-Of-Coolant Accidents is to maintain core integrity by providing adequate water for core cooling. Consequently, there are high and low pressure, high and low volume, safety and nonsafety grade means of delivering water to the vessel. These range from the large feedwater/condensate pumps down through low pressure emergency core cooling system pumps, high pressure coolant injection pump, reactor core isolation cooling pump, standby liquid control pumps, and control rod drive pumps. Many of these water supplies are redundant in equipment and systems (e.g. there are several LPCI/CS pumps and completely redundant piping systems).

Power uprate does not result in an increase or decrease in the available water sources nor does it change the selection of those assumed to be functional in the safety analyses. The NRC-approved SAFER/GESTR - LOCA methodology was used for the uprated power analyses which is the first time this code has been applied at Fermi 2.

Design Basis Accidents

For BWR licensing evaluations, capability is demonstrated for coping with the full spectrum of hypothetical pipe break sizes including the recirculation, steam, feedwater, ECCS, and instrument lines. This break spectrum concept analytically investigates the full spectrum of large and small, high and low energy line breaks and the success of the plant systems in dealing with them while accommodating a single active equipment failure in addition to the postulated LOCA. Several of the most significant licensing assessments are made using these LOCA ground rules. These assessments are:

(a) Challenges to fuel or ECCS performance analyses (Regulatory Guide 1.70, Section 6.3) in accordance with the rules and criteria of 10CFR50.46 and 10CFR50, Appendix K, wherein the predominant figure of merit is the fuel peak clad temperature.

(b) Challenges to the containment (Regulatory Guide 1.70, Section 6.2) wherein the primary figures of merit are the maximum containment pressure calculated during the course of the LOCA and maximum suppression pool temperature for long term cooling in accordance with 10CFR50, Appendix A, Criterion 38.

(c) Calculation of the design basis accident radiological consequences (Regulatory Guide 1.70, Section 15.6.5) and comparison to the criteria of 10CFR100.

(a) Challenges to Fuel

Emergency core cooling systems are described in the plant USAR, Section 6.3. The ECCS performance evaluation was conducted through application of the 10CFR50, Appendix K, evaluation models and then showing conformance to the acceptance criteria of 10CFR50.46. As mentioned above, a complete spectrum of pipe breaks was investigated from the largest recirculation line to instrument lines. As shown schematically in Figure 11-2 and numerically in Table 4-2, the licensing safety margin is not impacted by power uprate. The increased PCT consequences for power uprate are insignificant compared to the large amount by which the results are below the regulatory criteria.

(b) Challenges to the Containment

Table 4-1 provides the worst case results of analyses of the plant containment response to various large and small LOCAs. The impact of power uprate on the peak values for containment pressure and temperature confirms the suitability of the plant for operation at uprated power. Likewise, the impact of power uprate on the conditions which affect the containment dynamic loads has been determined and the plant is judged satisfactory for uprated power operation.

(c) Design Basis Accident Radiological Consequences

Radiological consequences of the design basis accidents that were reevaluated for power uprate are discussed in Section 9.2 of this report. These accidents were originally evaluated at 3430 MWt, the uprated power level. The recalculated radiological consequences remain well within the regulatory acceptance criteria limits of 10CFR100; therefore, the radiological safety margins established by those limits are maintained.

Transient Analyses

The effects of plant transients are evaluated by investigating a number of disturbances of process variables and malfunctions or failures of equipment according to a scheme of postulating initiating events. These events are primarily evaluated against the safety limit minimum critical power ratio. The SLMCPR is determined using NRC-approved methods (Figure 11-3).

The OLMCPR is increased appropriately to insure that the SLMCPR is not infringed upon when the transient is initiated from the uprated power level. Licensing safety margin is not impacted. Table 9-2 gives results for the most limiting transients for Fermi 2.

Combined Effects

Uprate analyses use fuel designed to present NRC-approved criteria and operated within present NRC-approved limits to produce more heat in the reactor which slightly increases reactor pressure and increases steam flow to the turbine. NRC-approved design duty cycle criteria are used to assure mechanical performance at uprate. Design basis accidents are hypothesized to assess challenges to the fuel, containment, and offsite dose limits. These challenges are evaluated separately in accordance with conservative regulatory procedures such that the separate effects are more severe than any combined effects. The

offsite dose evaluation specified by Regulatory Guide 1.3 and Standard Review Plan 15.6.5 provides a more severe DBA radiological consequences scenario than the combined effects of the hypothetical LOCA which produces the greatest challenge to the fuel and/or containment. That is, the DBA which produces the highest PCT and/or containment pressure does not fail large amounts of fuel and thus the source term and doses are much smaller than those calculated in the Regulatory Guide 1.3 evaluation.

Individual Plant Examination

The proposed power uprate has been assessed to have minimum impact on the Individual Plant Examination (IPE) currently underway for Fermi 2. This is not unexpected since the IPE is utilizing traditional Level 1 and Level 2 probabilistic risk assessment (PRA) techniques, which are not highly sensitive to small variations in reactor power level. Some reasons for this insensitivity are discussed below.

(a) System success criteria, an integral part of the PRA, are treated as discrete, e.g., typical success criteria would establish whether one or two pumps are needed to perform an individual function with typical margins in these determinations on the order of 20% to 50%. A 4% change in power level or corresponding decay heat level would have little effect. It is possible that the number of SRVs required to open to prevent excedence of a pressure limit may increase by one during an overpressure transient due to the higher operating pressure, but the resulting change in potential for failure of the pressure relief function would be trivial due to the number of excess relief valves.

(b) Component failure rates are also an integral component of the PRA, but these would not be expected to change since the operating environment of the components would either remain unchanged or change by very slightly.

(c) Operator actions, some equipment performance, and other phenomena considered in the PRA are sensitive to timing, but the times are generally on the order of hours, whereas the effects of power uprate would be on the order of minutes.

(d) Given radionuclide releases would increase fractionally in proportion to the power increase, i.e., about 4%, but this is a very small change compared to the order of magnitude radionuclide release bins used to characterize releases in the Level 2 PRA.

For these reasons, it was determined that power uprate would have minimal impact on the IPE.

Non-LOCA Radiological Release Accidents

All other radiological releases discussed in Regulatory Guide 1.70, Chapters 11 and 15 are either unchanged because they are not power dependent, or increase at most by the amount of the uprate. Most of the radiological assessments presented in the USAR for the original power level were run at 3430 MWt. Thus the assessment for these events at uprate, had the calculations been done using the same methodology and assumptions, would be only a 2% increase in the calculated dose. The dose consequences for all of the non-LOCA radiological release accident events are bounded by the design basis radiological consequences events.

Equipment Qualification

Plant equipment and instrumentation have been evaluated against the criteria appropriate for uprate. Significant groups/types of equipment have been justified for uprate by generic evaluations. Some of the qualification testing/justification at the current power level was done at more severe conditions than the minimum required. In some cases the qualification envelope did not change significantly due to uprate. Section 10.2 of this report discusses the plant specific EQ evaluations. Those evaluations confirm that the plant safety-related equipment will still be able to perform its intended functions.

Balance-of-Plant

BOP systems/equipment used to perform safety-related and normal operating functions have been reviewed for uprate in a manner comparable to that for safety-related NSSS systems/equipment. This includes, but was not necessarily limited to all or portions of the main steam, feedwater, turbine, condenser, condensate, essential and non-essential service water, emergency diesel generator, BOP piping, and support systems. Significant groups/types of BOP equipment/systems are justified for uprate by generic evaluations. Plant unique evaluations justify power uprate operation for BOP systems/equipment that are not generically justified.

Technical Specification Changes

The Technical Specifications ensure that plant and system performance parameters are maintained at the values assumed in the safety analysis. That is, the Technical Specifications (setpoints, trip settings, etc.) are selected such that the actual equipment is maintained equal to or conservative with respect to the assumptions used in the safety analyses. The Technical Specification changes justified by the safety analyses summarized in this report are listed in Table 11-1 and summarized below. Proper account is taken of inaccuracies introduced by instrument drift, instrument accuracy, and calibration accuracy. For example, the reactor pressure trip is set lower in the Technical Specifications than that used in the safety analysis. This assures that the actual plant behavior will be less severe

than that represented by the safety analysis. Similarly, the Technical Specifications address equipment availability and put limits on equipment out-of-service such that the actual plant can be expected to have at least the complement of equipment available to deal with plant transients as that assumed in the safety analyses. Since the safety analyses show that the results are acceptable within regulatory limits, the plant can be expected to behave even more mildly to actual transients than portrayed by the safety analyses thereby assuring the public health and safety.

Technical Specification setpoint changes associated with the uprated power level are made in accordance with setpoint methodology consistent with that presented in Reference 4 and continue to provide the same level of protection as Technical Specifications previously issued by the NRC.

The Technical Specification changes for Fermi 2 uprated power operation are summarized as follows:

(a) Power increase related changes including:

1. Redefining of rated thermal power in Technical Specification Definition 1.32 to reflect the approximately 4.2% increase to 3430 MWt.

2. Rescaling of Technical Specification limits for flow-biased APRM rod blocks and scrams to account for the redefined rated thermal power while maintaining the same limits in terms of megawatts thermal.

3. Rescaling of single recirculation loop operation power limits and associated bases to maintain the same limits in terms of megawatts thermal.

4. Rescaling of the idle loop startup restrictions and associated bases to reflect the redefinition of rated thermal power to maintain the current rod lines and absolute power restrictions.

5. Rescaiing of the core thermal-hydraulic stability operating restrictions and associated bases to reflect the redefinition of rated thermal power such that the same restrictions are conservatively maintained in terms of absolute power level.

(b) Pressure increase related changes including:

1. Increasing the dome pressure operating limit by 5 psi to 1045 psig.

2. Increasing the high pressure scram trip setpoints and allowable values by 25 psi to $\leq$ 1093 psig and $<$ 1113 psig, respectively.

3. Increasing the safety/relief valve setpoints by 25 psi such that five valves each are set at 1135, 1145, and 1155 psig ± 1%.

4. Redefining the turbine first stage pressure scram bypass at 30% power to reflect the uprated 30% power pressure of 161.9 psig.

5. Increasing the HPCI and RCIC surveillance testing pressure by 25 psig to allow testing at the new normal operating pressure.

6. Increasing the standby liquid control system surveillance test pressure by 25 psig to allow the appropriate demonstration of system capability at uprated operating conditions.

7. Increasing the reactor coolant leakage pressure specification to 1045 psig to limit maximum allowable isolation valve leakage to 1 gpm at the uprated maximum allowable operating dome pressure.

(c) Rated steam flow increase related changes including:

1. The main steamline flow increase that accompanies uprated power operation results in increased steamline isolation differential pressure setpoints and allowable values at 137.9% and 139.5% rated steam flow which correspond to 118.9 psid and 121.9 psid, respectively.

2. The turbine bypass valve/system flow design capabilities in the Technical Specifications bases are changed to approximately 12½% and 25% rated steam flow, respectively.

(d) The pressure/temperature curves are modified to indicate an estimated 5°F increase in the 32 effective-full-power-year core beltline nil-ductility shift to 114°F due to increased neutron fluence. The dotted end-of-life core beltline curves shift slightly, but remain non-limiting.

(e) The standby liquid control system high and low level alarms/limits are increased to 3040 gallons and 2560 gallons, respectively to achieve an equivalent boron-10 concentration increase from 825 to 900 parts per million (ppm). This includes a 25% margin over the 720 ppm calculated as necessary to shutdown the reactor. The associated bases are revised accordingly.

(f) The LOCA analysis methodology identified in Technical Specification Bases 3/4.2 Reference 2, is revised to reflect the NRC-approved SAFER/GESTR methodology in "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident - SAFER/GESTR Application Methodology", NEDE-23785-1-PA.

Similarly, Technical Specification 6.9.3 is revised to reference NEDE-23785-1-PA. The application of the SAFER/GESTR methodology eliminates the need for the single recirculation loop operation MAPLHGR multiplication factor which has been removed from the Technical Specifications and associated bases.

(g) The containment system Technical Specification bases Section 3/4.6 is revised in several places to indicate that the updated analysis results for peak containment pressure is "less than" the 56.5 psig value which was retained as the bases for these Technical Specifications.

11.4.3 Assessment Against 10CFR50.92 Criteria

For this significant hazards consideration assessment, the criteria of 10CFR50.92 were applied to power uprate. The conclusions are based on the evaluations provided in this report, and are summarized as appropriate for the following safety considerations in accordance with 10CFR50.92.

(1) **Will the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The increase in power level discussed herein and associated Technical Specification changes described above do not significantly increase the probability or consequences of an accident previously evaluated.

The probability (frequency of occurrence) of design basis accidents occurring is not affected by the increased power level because the design or regulatory criteria established for plant equipment (ASME code, IEEE standards, NEMA standards, etc.) are still imposed for the uprated power level. Scram setpoints are established such that there is no significant increase in scram frequency due to uprate.

The changes in consequences of hypothetical accidents that occur from 102% of the uprated power, as opposed to those previously evaluated from ≥102% of the current power, are in all cases insignificant since the accident evaluations for power uprate do not result in exceeding the NRC-approved acceptance limits. The spectrum of hypothetical accidents and transients has been investigated and shown to meet the same regulatory criteria after uprate as before uprate. In the area of core design, for example, the fuel operating limits such as maximum average planar linear heat generation rate and minimum critical power ratio are still met at the uprated power level. Reload analyses will confirm that plant transients meet the criteria accepted by the NRC as specified in GESTAR II (Reference 2). Challenges to fuel or ECCS performance are evaluated and shown to still meet the criteria of 10CFR50.46 and 10CFR50, Appendix K (Section 9.2 in this report and Regulatory Guide 1.70, Section 6.3). Challenges to the containment have been evaluated and still meet

10CFR50, Appendix A, Criterion 38, long term cooling, and Criterion 50, containment. Radiological release events have been evaluated and shown to meet the criteria of 10CFR100 (Regulatory Guide 1.70, Chapter 15).

The plant was originally designed anticipating a 5% steam flow increase and many of the analyses were originally performed using 3430 MWt, including overpressurization and ECCS/LOCA. The radiological consequences of the DBA/LOCA and MSLB accidents were initially calculated at the uprate power value. The only increase in power for the power uprate analyses is the addition of the 2% uncertainty factor, which is not a significant power increase.

The increased licensed power level to 3430 MWt and the associated Technical Specification redefinition of rated thermal power have been factored into the aforementioned analyses, as applicable. The results of these analyses, as described in this report, meet all applicable regulatory criteria and, as discussed above, do not significantly contribute any increase to the probability or consequences of any accident previously evaluated. The power-related Technical Specification changes made only to reflect the redefinition of rated thermal power, while maintaining current limits in terms of megawatts thermal, result in no actual change in limits and have no effect on previously evaluated accidents.

The pressure-increase-related Technical Specification changes described above have been factored into the safety analyses or have been determined to be non-limiting (i.e. turbine first stage pressure - Reference 5, Appendix F.4.2.(c)). The HPCI and RCIC surveillance test pressure specification increase to allow testing at normal operating pressures fulfills the original intent of that specification. The standby liquid control system test pressure increase verifies the ability of the system to pump sodium pentaborate at uprated operating pressure conditions, and is well within the system design pressure. Increasing the leakage rate pressure is consistent with the new maximum allowable operating dome pressure and is conservative. The increase of the high pressure scram setpoint and of the SRV setpoints by the same amount as the planned operating pressure increase maintains the same level of trip avoidance for the scram and the same simmer margin for the SRVs as originally provided. Thus, the increase in pressure will not result in an increase in the number of unnecessary scrams or SRV actuations. Since the associated safety analyses have been reperformed and meet the applicable regulatory criteria/ASME code limits and the trip/actuation avoidance margins are maintained, the pressure-increase-related Technical Specification changes do not increase the probability or consequences of a previously evaluated accident.

The main steamline flow related Technical Specifications are changed to reflect the redefinition of rated main steamline flow that accompanies uprated power operation. The main steamline flow differential pressure trip setpoints are maintained at 137.9%

and 139.5% of uprated steam flow. These limits were retained to ensure that an adequate trip avoidance margin is maintained for the normal plant testing of MSIVs and turbine stop valves. A high assurance of isolation protection is still provided for a main steamline break accident which meets the original design intent. These setpoints have no effect on the probability of the occurrence of a main steamline break. Also, since a high level of assurance of break isolation is maintained, these setpoint changes do not significantly increase the consequences of the main steamline break accident. Similarly, the turbine bypass system capabilities are redefined in the Technical Specification bases to reflect the redefinition of rated steam flow. This bases change is informational and has no effect on the safety analyses which conservatively reflect the system capability. Thus, the bases change does not significantly affect the probability or consequences of an accident.

The pressure/temperature Technical Specification curve changes reflecting the estimated effect of increased neutron fluence are still well within the 200° F nil-ductility shift regulatory limit. Also, the upper shelf energy remains greater than 50 ft.-lb. for the design life of the vessel. The assumed end-of-life core beltline shift represented by the dotted curves remains non-limiting. The actual effect of neutron fluence is determined based on the results of in-vessel surveillance sample flux wire evaluation. Actual fluences are expected to be less than those used in this evaluation. Since the vessel is still in compliance with regulatory requirements, the increase in fluence does not significantly affect reactor vessel fracture toughness. Therefore, there is no significant increase in the probability or consequences of an accident previously evaluated.

The standby liquid control system sodium pentaborate solution inventory increase proposed Technical Specification change provides the core reload design flexibility to maintain 18-month fuel reload cycles. Cycle length is maintained through higher reload batch sizes or higher enrichment. The proposed change does not change the permissible solution concentrations in the SLC storage tank which could impact the minimum temperature limit. The amount of solution in the tank is increased to achieve the desired 900 ppm equivalent boron-10 concentration. This increase in equivalent boron-10 concentration will provide the required design flexibility while maintaining an adequate shutdown capability. Thus, the probability or consequences of an accident previously evaluated are not significantly increased.

Technical Specification changes are made to reflect the use of the NRC-approved SAFER/GESTR - LOCA methodology and to eliminate the single-loop operation MAPLHGR multiplication factor. An evaluation has been performed for SLO with no reduction in MAPLHGR. The calculated PCTs are higher than for the corresponding two-loop operation condition, but remained well below the PCT limit of 2200° F. Furthermore, the required MCPR and MAPLHGR limits at the SLO low power/low flow operating condition ensure that the calculated SLO/PCT is below

that calculated for normal two-loop operation. Therefore, these Technical Specification changes will not result in a significant increase in the probability or consequences of a previously evaluated accident.

The containment system Technical Specification bases revisions are informational only. As indicated in Table 4-1 of this report, peak drywell pressure was calculated to be 49.9 psig, less than the 56.5 psig cited in the bases. The 56.5 psig is retained as a conservative bounding value. Containment leakage rate testing is performed at the bounding 56.5 psig value which produces larger indicated leakage rates than if the lower value were used. Since these changes are merely informational and have no effect on plant design or operation, they will not result in a significant increase in the probability or consequences of a previously evaluated accident.

(2) **Will the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

This change will not create the possibility of a new or different kind of accident from any accident previously evaluated. The full spectrum of accident considerations defined in Regulatory Guide 1.70 has been reviewed and no new or different kind of accident has been identified. Power uprate uses already developed technology and applies it within the capabilities of already existing plant equipment in accordance with presently existing regulatory criteria to include NRC-approved codes, standards, and methods. GE has designed BWRs of higher power levels than the uprated power of any of the currently operating BWRs, and no new power dependent accidents have been identified. In addition, Fermi 2 was originally designed to the proposed uprated steam flow (105%) and many of the accident analyses were performed at that condition.

The plant systems have been assessed and have been verified to be adequately designed and capable of performing their design intent at uprated operating conditions. The only modification planned is implementation of the SIL 377 recommended one-inch steam inlet bypass valve modification which reduces the RCIC turbine tendency to overspeed before adequate governor control valve hydraulic oil pressure is achieved from the turbine driven oil pump. This modification will be designed to the same quality standards as the RCIC system and will be assessed by a 10CFR50.59 safety evaluation prior to implementation, since the modification does not require a change to the Technical Specifications.

The changes to the Technical Specifications described above do not change the plant configuration. The Technical Specification operating parameters of power, pressure, flow, neutron flux, and shutdown boron concentration are increased to accommodate power uprate, but since there is little change to plant hardware configuration, system hardware lineups, or operating methods, no new accident scenarios have been

identified. The change in accident evaluation methodology to the NRC-approved SAFER/GESTR - LOCA code and resulting elimination of the MAPLHGR multiplier under single recirculation loop operation, similarly has no effect on plant configuration and does not affect operating methods in such a way as to create a new or different type of accident from those previously evaluated.

The addition of the clarification that analyzed peak pressures are less than the 56.5 psig peak accident pressure specified is informational and has no effect on the types of accidents that have been established.

(3) **Will the change involve a significant reduction in a margin of safety?**

As discussed in Section 5 of Reference 1, the safety margins prescribed by the Code of Federal Regulations have been maintained by meeting the appropriate regulatory criteria. Similarly, the margins provided by the application of the American Society of Mechanical Engineers (ASME) design acceptance criteria have been maintained where applicable, as well as other margin-assuring acceptance criteria used to judge the acceptability of the plant.

Several accident and transient analyses have been reperformed at uprated plant operating conditions consistent with the requested Technical Specification changes. The NRC-approved SAFER/GESTR - LOCA methodology was used in the LOCA analysis.

Additionally, Reference 1 addresses the BWR generic acceptability of analytical evaluations for the loss of feedwater transient, stability, core spray distribution, safety limit minimum critical power ratio, containment atmosphere combustibility, materials and coolant chemistry, and anticipated transients without scram (ATWS).

The radiological doses have been recalculated for the events discussed in Section 9.2 of this report. As discussed in Section 5.2.3 of Reference 1, actual offsite doses for the DBA/LOCA will increase proportionately to reactor power when compared on a consistent basis. The doses presented in Tables 9.3 and 9.4 of this report are not comparable on a consistent basis to previous calculations since the NRC-approved methodology and assumptions used have undergone extensive revision. However, as can be determined from these tables, the required acceptance criteria of 10CFR100 are still met.

The radiological doses resulting from the DBA/LOCA and MSLB accidents were initially analyzed at the uprated power level (3430 MWt). For the current power uprate program, the increase is only the 2% uncertainty factor added for conservatism.

As discussed throughout this report, in each case the relevant acceptance criteria is met which preserves the margin of safety provided by these criteria. It is therefore concluded that the requested changes do not involve a significant reduction in any margin of safety.

Conclusions

This report describes investigations made for a power uprate to 3430 MWt, equivalent to 105% of original steam flow. A 5% increase in steam flow was factored into the original design basis for Fermi 2. Several major analyses, including overpressurization, ECCS/LOCA, and the radiological consequences of the DBA/LOCA were originally performed at 3430 MWt.

The strategy for achieving higher power is to expand the power/flow map by increasing core flow along existing flow control lines. The predominant plant licensing challenges have been reviewed to demonstrate that this uprate can be implemented without a significant increase in the probability or consequences of an accident previously evaluated, without creating the possibility of a new or different kind of accident from any accident previously evaluated, and without exceeding any presently existing regulatory limits applicable to the plant which might cause a significant reduction in a margin of safety. Having arrived at negative declarations with respect to these considerations of 10CFR50.92, this assessment concludes that uprate of the amount described herein does not involve a significant hazards consideration.

11.5 REFERENCES

1. GE Nuclear Energy, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31984P, Class III (Proprietary), July 1991.

2. GE Nuclear Energy, "General Electric Standard Application for Reactor Fuel - GESTAR II, United States Supplement," NEDE-24011-P-A-10-US, March 1991.

3. GE Nuclear Energy, "Licensing Criteria for Fuel Designs," NEDO-31908, Amendment 22 to NEDE-24011-P-A, February 1991.

4. General Electric Co., "Instrument Setpoint Methodology," NEDC-31336, Class III (Proprietary), October 1986.

5. GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31897P-1, Class III (Proprietary), June 1991.

Table 11-1

TECHNICAL SPECIFICATIONS AFFECTED BY POWER UPRATE

Location	Effect
Definition	
1.32	Revise value of rated thermal power definition to uprated power level
Safety Limits and Limiting Safety System Settings	
Table 2.2.1-1	Revise the APRM scram and reactor vessel steam high dome pressure of the reactor protection system setpoints
Limiting Conditions for Operation and Surveillance Requirements	
Figure 3.1.5-1	Revise to implement 900 ppm minimum boron-10 concentration provided by SLC
4.1.5.c	Increase SLC test pressure by 25 psig to 1275 psig
3.2.1	Eliminate the MAPLHGR limit for single-loop operation
Table 3.3.1-1	Revise value of turbine first stage pressure to be equivalent to that at 30% of rated thermal power in Notation j and Action 6
Table 3.3.2-2	Revise the main steamline high flow differential pressure setpoints
Table 3.3.6-2	Revise the APRM control rod block instrument setpoints
3.4.1.1	Revise the thermal power value and eliminate the MAPLHGR limit for single-loop operation
4.4.1.1.3	Revise the thermal power value for single-loop operation
Figure 3.4.1.4-1	Revise the thermal power value for idle loop startup
3.4.2.1	Revise values of SRV setpoints
3.4.3.2.d	Revise leakage rate pressure applicability

Table 11-1 (continued)

TECHNICAL SPECIFICATIONS AFFECTED BY POWER UPRATE

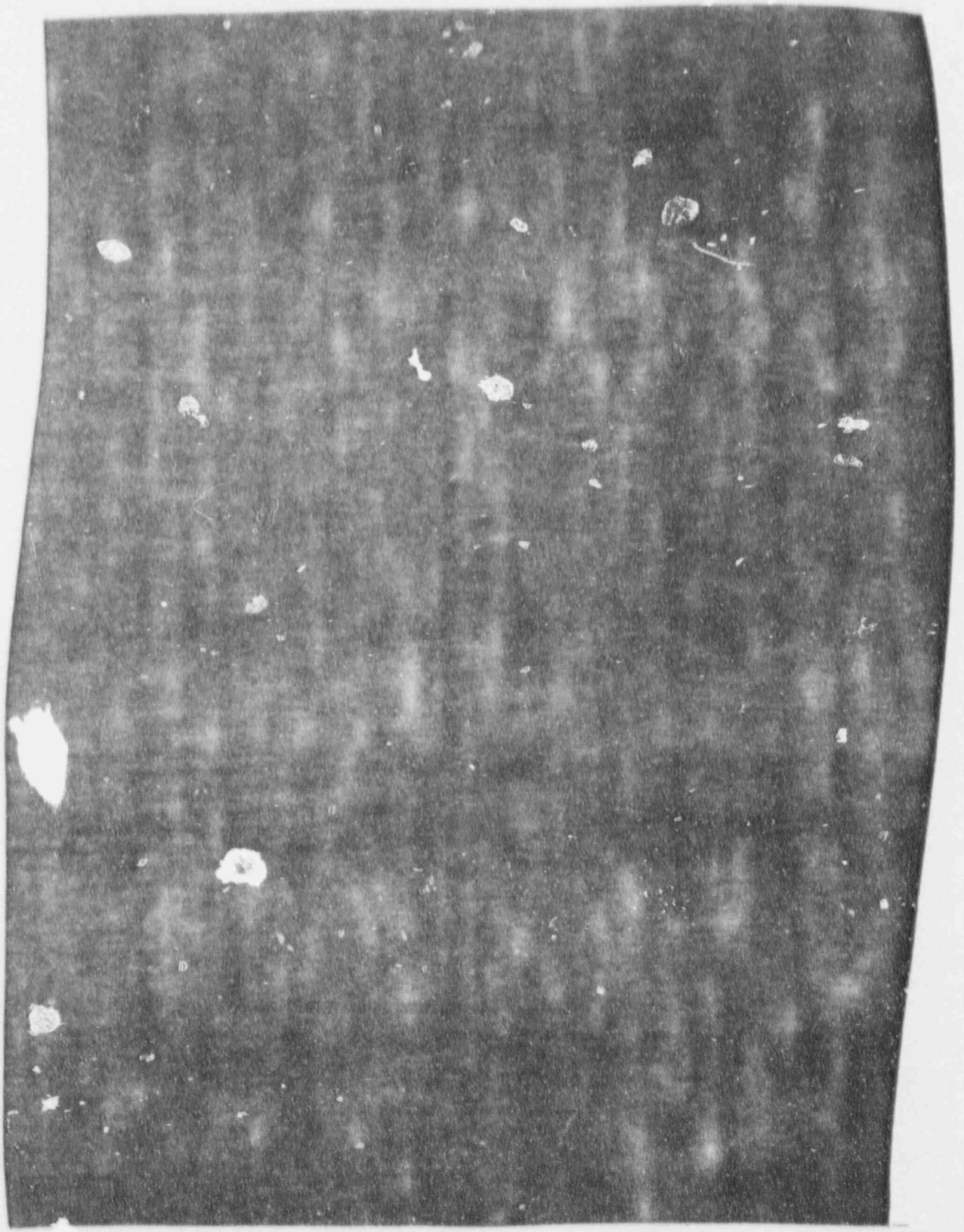
Location	Effect
Limiting Conditions for Operation and Surveillance Requirements (cont'd)	
Figure 3.4.6.1-1	Revise vessel pressure/temperature curves
3.4.6.2/4.4.6.2	Revise the value of the reactor steam dome pressure
Figure 3.4.10-1	Revise the thermal power value for core thermal-hydraulic stability
4.5.1.b.3	Revise HPCI test pressure
4.7.4.b	Revise RCIC test pressure
BASES for Sections 3 and 4	
3/4.1.5	Revise to implement 720 ppm minimum (900 ppm minimum with 25% margin) boron concentration for SLC
3/4.2	Revise reference to reflect the NRC-approved generic GE SAFER/GESTR-LOCA analysis basis
3/4.2.1	Eliminate discussion of MAPLHGR reduction factor for SLO
3/4.4.1	Revise to reflect single-loop operation thermal power and eliminate the MAPLHGR reduction factor
3/4.4.10	Revise to reflect thermal power for thermal-hydraulic stability
3/4.6.1.2 & 3/4.6.1.6	Revise to indicate that uprated analysis demonstrates maximum expected pressure is less than 56.5 psig
3/4.6.2	Revise the reactor steam dome pressure, Revise the containment accident pressure from "approximately" to "less than" 56.5 psig
3/4.7.9	Revise the bypass valve capabilities to reflect the redefinition of rated steam flow.

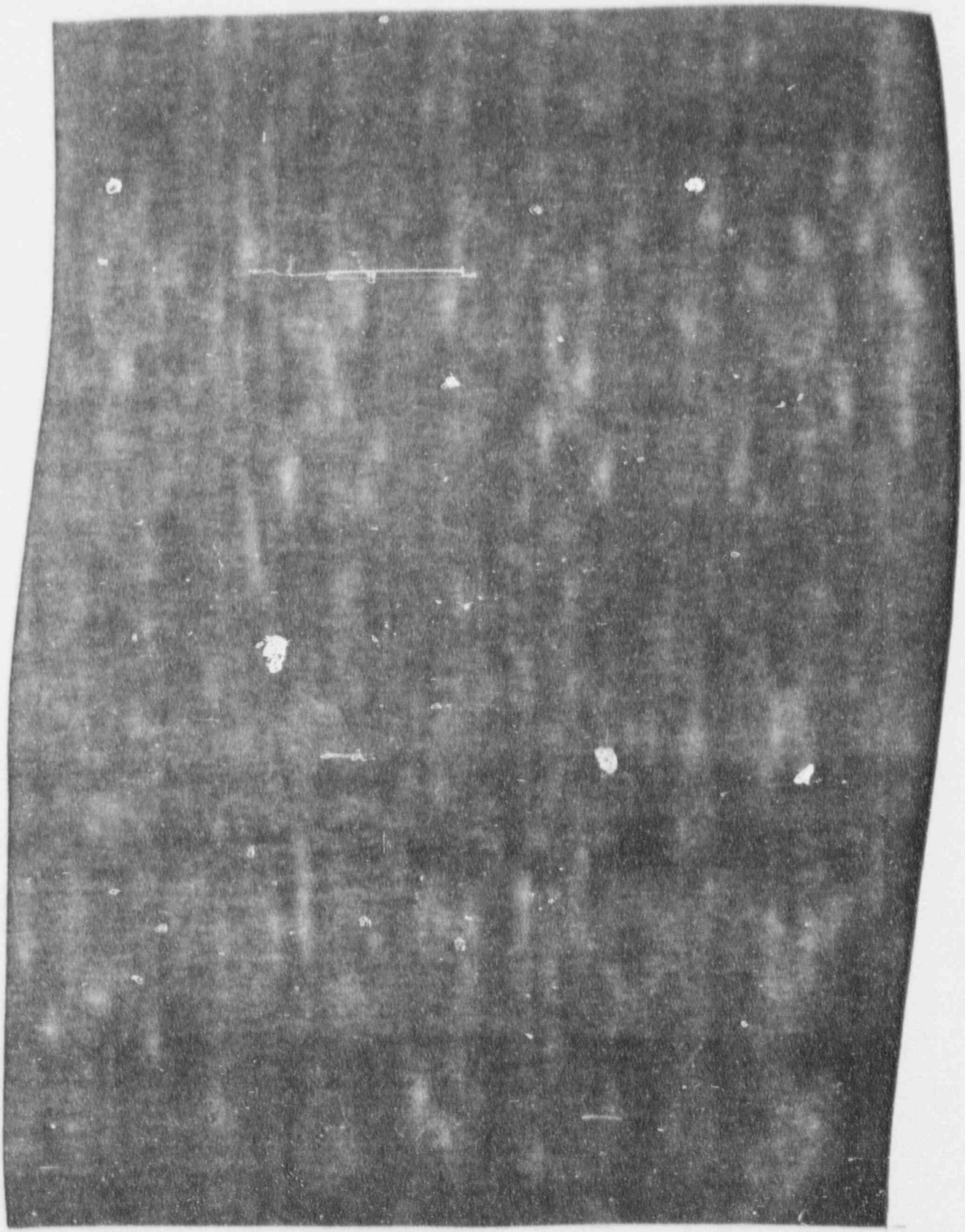
Table 11-1 (continued)

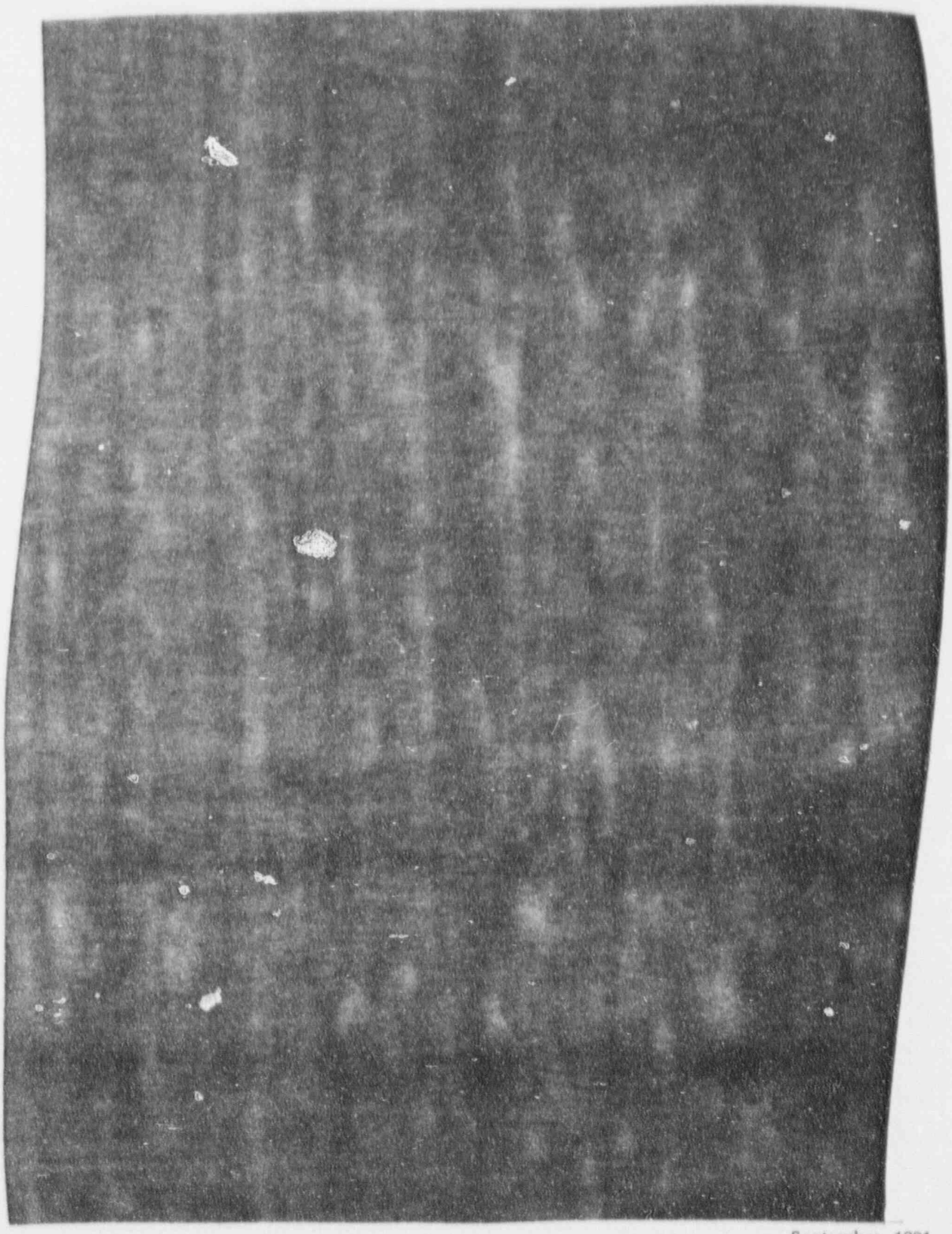
TECHNICAL SPECIFICATIONS AFFECTED BY POWER UPRATE

Location	Effect
Administrative Controls	
6.9.3	Revise to add the generic GE SAFER/GESTR - LOCA reference

Note: As used in this table, the term "setpoints" include the nominal trip setpoints and allowable values (where applicable).







GENERAL ELECTRIC COMPANY

AFFIDAVIT

I, DAVID J. ROBARE, being duly sworn, depose and state as follows:

1. I am Manager, Plant Licensing Services, General Electric Company, and have been delegated the function of reviewing the information described in paragraph 2 which is sought to be withheld and have been authorized to apply for its withholding.

2. The information sought to be withheld is contained in Detroit Edison Report Fermi-2-91-150 "Power Uprate Safety Analysis" September 1991. The GE Proprietary portions of this report are identifiable by the "GE Proprietary Information" designation at the top of the page.

3. In designating material as proprietary, General Electric utilizes the definition of proprietary information and trade secrets set forth in the American Law Institute's Restatement of Torts, Section 757. This definition provides:

> "A trade secret may consist of any formula, pattern, device or compilation of information which is used in one's business and which gives him an opportunity to obtain an advantage over competitors who do not know or use it...A substantial element of secrecy must exist, so that, except by the use of improper means, there would be difficulty in acquiring information...Some factors to be considered in determining whether given information is one's trade secret are (1) the extent to which the information is known outside of his business; (2) the extent to which it is known by employees and others involved in his business; (3) the extent of measures taken by him to guard the secrecy of the information; (4) the value of the information to him and to his competitors; (5) the amount of effort or money expanded by him developing the information; (6) the ease or difficulty with which the information could be properly acquired or duplicated by others."

4. Some examples of categories of information which fit into the definition of Proprietary Information are:

   a. Information that discloses a process, method or apparatus where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

   b. Information consisting of supporting data and analyses, including test data, relative to a process, method or apparatus, the application of which provide a competitive economic advantage, e.g., by optimization or improved marketability;

Information in this record was deleted
in accordance with the Freedom of Information
Act, exemptions 4
FOIA- 91-579

GENERAL ELECTRIC COMPANY

AFFIDAVIT

c. Information which if used by a competitor, would reduce his expenditures of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality or licensing of a similar product;

d. Information which reveals cost or price information, production capacities, budget levels or commercial strategies of General Electric, its customers or suppliers;

e. Information which reveals aspects of past, present or future General Electric customer-funded development plans and programs of potential commercial value to General Electric;

f. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection;

g. Information which General Electric must treat as proprietary according to agreements with other parties.

5. Initial approval of proprietary treatment of a document is typically made by the Subsection Manager of the originating component, the person who is most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within the Company is limited on a "need to know" basis and such documents are clearly identified as proprietary.

6. The procedure for approval of external release of such a document typically requires review by the Subsection Manager, Project Manager, Principal Scientist or other equivalent authority, by the Subsection Manager of the cognizant Marketing function (or delegate) and by the Legal Operation for technical content, competitively effect and determination of the accuracy of the proprietary designation in accordance with the standards enumerated above. Disclosures outside General Electric are generally limited to regulatory bodies, customers and potential customers and their agents, suppliers and licensees then only with appropriate protection by applicable regulatory provisions or proprietary agreements.

7. The document mentioned in paragraph 2 above has been evaluated in accordance with the above criteria and procedures and has been found to contain information which is proprietary and which is customarily held in confidence by General Electric.

8. The information to the best of my knowledge and belief has consistently been held in confidence by General Electric Company, no public disclosure has been made, and it is not available in public sources.

GENERAL ELECTRIC COMPANY

AFFIDAVIT

8. All disclosures to third parties have been made pursuant to regulatory provisions of proprietary agreements which provide for maintenance of the information in confidence.

9. Public disclosure of the information sought to be withheld is likely to cause substantial harm to the competitive position of the General Electric Company and deprive or reduce the availability of profit making opportunities. A substantial effort has been expended by General Electric to develop this information.

GENERAL ELECTRIC COMPANY

AFFIDAVIT

STATE OF CALIFORNIA)
) ss:
COUNTY OF SANTA CLARA)

David J. Robare, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are truly and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 11TH day of SEPTEMBER 19 91.

David J. Robare
General Electric Company

Subscribed and sworn before me this 11th day of September 19 91.

OFFICIAL SEAL
PAULA F. HUSSEY
NOTARY PUBLIC - CALIFORNIA
SANTA CLARA COUNTY
My comm. expires APR 5, 1994

Notary Public, State of California