June 20, 1988

Docket No. 50-249

DISTRIBUTION

Docket File	JPartlow
NRC/Local PDRs	TBarnhart (4)
PDIII-2 Rdg.	Wanda Jones
GHolahan	EButcher
LLuther	LKopp
BSiegel	ACRS (10)
OGC-Rockville	GPA/PA
DHagan	ARM/LFMB
EJordan	PDIII-2 Plant File

Mr. Henry Bliss
Nuclear Licensing Manager
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Bliss:

SUBJECT: CYCLE 11 RELOAD (TAC NO. 67467)

Re: Dresden Nuclear Power Station, Unit No. 3

The Commission has issued the enclosed Amendment No. 94 to Facility Operating License No. DPR-25 for Dresden Unit 3. This amendment is in response to your application dated March 9, 1988 as supplemented by your June 17, 1988 submittal.

The amendment changes the Technical Specifications to support Cycle 11 operation of Dresden Unit 3 with Advanced Nuclear Fuels Company 9x9 reload fuel, and Section 3.E Restrictions, of the license.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be published in the Federal Register .

Sincerely,

/S/ LNorrholm

Leif J. Norrholm, Acting Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projecs

Enclosures:
1. Amendment No.94 to License No. DPR-25
2. Safety Evaluation

cc w/enclosures:
See next page

*See previous concurrence

OFFICE:	PDIII-2:PM	PDIII-2:LA	*OGC-Rockville	PDIII-2:PD
SURNAME:	*BSiegel:bj	LLuther		LNorrholm
DATE:	06/9/88	06/20/88	06/10/88	06/20/88

DFol
/1

8807140068 880620
PDR ADOCK 05000249
P PDC

Docket No. 50-249

DISTRIBUTION	
Docket File	JPartlow
NRC/Local PDRs	TBarnhart (4)
PDIII-2 Rdg.	Wanda Jones
GHolahan	EButcher
LLuther	LKopp
BSiegel	ACRS (10)
OGC-Rockville	GPA/PA
DHagan	ARM/LFMB
EJordan	PDIII-2 Plant File

Mr. Henry Bliss
Nuclear Licensing Manager
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Bliss:

SUBJECT: CYCLE 11 RELOAD (TAC NO. 67467)

Re: Dresden Nuclear Power Station, Unit No. 3

The Commission has issued the enclosed Amendment No.94 to Facility Operating License No. DPR-25 for Dresden Unit 3. This amendment is in response to your application dated March 9, 1988.

The amendment changes the Technical Specifications to support Cycle 11 operation of Dresden Unit 3 with Advanced Nuclear Fuels Company 9x9 reload fuel, and Section 3.E Restriction, of the license.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be published in the Federal Register .

Sincerely,

Leif J. Norrholm, Acting Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projecs

Enclosures:
1. Amendment No. 94 to License No. DPR-25
2. Safety Evaluation

cc w/enclosures:
See next page

w/ noted revisions to SE + admit ck STATE before issuance

OFFICE:	PDIII-2:PM BJ	PDIII-2:LA	OGC-Rockville	PDIII-2:PD
SURNAME:	BSiegel:bj	LLuther	MYoung	LNorrholm
DATE:	06/ 9 /88	06/ /88	06/10 /88	06/ /88

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

June 20, 1988

Docket No. 50-249

Mr. Henry Bliss
Nuclear Licensing Manager
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Bliss:

SUBJECT: CYCLE 11 RELOAD (TAC NO. 67467)

Re: Dresden Nuclear Power Station, Unit No. 3

The Commission has issued the enclosed Amendment No. 94 to Facility Operating License No. DPR-25 for Dresden Unit 3. This amendment is in response to your application dated March 9, 1988 as supplemented by your June 17, 1988 submittal.

The amendment changes the Technical Specifications to support Cycle 11 operation of Dresden Unit 3 with Advanced Nuclear Fuels Company 9x9 reload fuel, and Section 3.E Restrictions, of the license.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be published in the Federal Register .

Sincerely,

Leif J. Norrholm, Acting Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projecs

Enclosures:
1. Amendment No.94 to
   License No. DPR-25
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Henry Bliss
Commonwealth Edison Company

Dresden Nuclear Power Station
Units 2 and 3

cc:
Michael I. Miller, Esq.
Sidley and Austin
One First National Plaza
Chicago, Illinois 60602

Mr. J. Eenigenburg
Plant Superintendent
Dresden Nuclear Power Station
Rural Route #1
Morris, Illinois 60450

U. S. Nuclear Regulatory Commission
Resident Inspectors Office
Dresden Station
Rural Route #1
Morris, Illinois 60450

Chairman
Board of Supervisors of
Grundy County
Grundy County Courthouse
Morris, Illinois 60450

Regional Administrator
Nuclear Regulatory Commission, Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Mr. Michael E. Parker, Chief
Division of Engineering
Illinois Department of Nuclear Safety
1035 Outer Park Drive, 5th Floor
Springfield, Illinois 62704

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 94
License No. DPR-25

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated March 9, 1988 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B. of Facility Operating License No. DPR-25 is hereby amended to read as follows:

   B. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No.94, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Section 3.E of the license is hereby changed to read:

   "3.E <u>Restrictions</u>
   Operation in the coastdown mode is permitted to 40% power."

4. This license amendment is effective as of the date of its issuance to be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Leif J. Norrholm, Acting Director
Project Directorate III-2
Division of Reactor Projects - III, IV, V and Special Projects

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 20, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 94

FACILITY OPERATING LICENSE DPR-25

DOCKET NO. 50-249

1. A new page 4 for your license is included for your convenience to reflect the change in Section 3E Restrictions.

2. Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT	REMOVE	INSERT
iii	iii	B 3/4 5-30	B 3/4 5-30
v	v	B 3/4 5-31	B 3/4 5-31
viii	viii	B 3/4 5-32	B 3/4 5-32
1.0-2	1.0-2	B 3/4 5-33	B 3/4 5-33
1.0-5	1.0-5	B 3/4 5-34	B 3/4 5-34
1.0-6	1.0-6	B 3/4 5-35	B 3/4 5-35
1/2 1-1	1/2 1-1	B 3/4 5-36	B 3/4 5-36
1/2 1-2	1/2 1-1	B 3/4 5-37	B 3/4 5-37
1/2 1-3	1/2 1-3	B 3/4 5-38	B 3/4 5-38
1/2 1-4	1/2 1-4	B 3/4 5-39	B 3/4 5-39
B 1/2 1-7	B 1/2 1-7	B 3/4 5-40	B 3/4 5-40
B 1/2 1-8	B 1/2 1-8	B 3/4 5-41	B 3/4 5-41
B 1/2 1-12	B 1/2 1-12	B 3/4 5-42	B 3/4 5-42
B 1/2 1-14	B 1/2 1-14	B 3/4 5-43	B 3/4 5-43
3/4 1-1	3/4 1-1	3/4 6-15	3/4 6-15
3/4 1-2	3/4 1-2	3/4 6-16	3/4 6-16
B 3/4 1-20	B 3/4 1-20	B 3/4 6-36	B 3/4 6-36
3/4 2-12	3/4 2-12	B 3/4 6-37	B 3/4 6-37
3/4 2-13	3/4 2-13	B 3/4 6-38	B 3/4 6-38
3/4 5-8	3/4 5-8	3/4 10-8	3/4 10-8
3/4 5-9	3/4 5-9		
3/4 5-15	3/4 5-15		
3/4 5-16	3/4 5-16		
3/4 5-17	3/4 5-17		
3/4 5-18	3/4 5-18		
3/4 5-19	3/4 5-19		
3/4 5-20	3/4 5-20		
3/4 5-21	3/4 5-21		
3/4 5-22	3/4 5-22		
3/4 5-23	3/4 5-23		
3/4 5-24	3/4 5-24		
3/4 5-25	3/4 5-25		
3/4 5-26	3/4 5-26		
3/4 5-27	3/4 5-27		
3/4 5-28	3/4 5-28		
3/4 5-29	3/4 5-29		

3. E. Restrictions

Operation in the coastdown mode is permitted to 40% power.

Am.

F. Deleted

Am. 33 3/22/78

G. The licensee may proceed with and is required to complete the modification identified in Paragraphs 3.1.1 through 3.1.23 of the NRC's Fire Protection Safety Evaluation (SE) dated March 1978 on the facility. All modifications are to be completed by startup following the 1979 Unit 3 refueling outage. In addition, the licensee shall submit the additional information identified in Table 3.1 of the SE in accordance with the schedule contained therein. In the event these dates for submittal cannot be met, the licensee shall submit a report, explaining the circumstances, together with a revised schedule.

H. Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the following Commission approved documents, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p). These approved documents consist of information withheld from public disclosure pursuant to 10 CFR 2.790(d).

Am. 49 2/11/81

(1) "Security Plan for the Dresden Nuclear Power Station", dated November 19, 1977, as revised May 19, 1978, May 27, 1978, July 28, 1978 and February 19, 1979.

(2) "Dresden Nuclear Power Station Safeguards Contingency Plan", dated March 1980, as revised June 27, 1980, submitted pursuant to 10 CFR 73.40. The Contingency Plan shall be fully implemented, in accordance with 10 CFR 73.40(b), within 30 days of this approval by the Commission.

(Table of Contents, Cont'd.)

		Page
3.5.C	HPCI Subsystem	3/4.5-6
3.5.D	Automatic Pressure Relief Subsystems	3/4.5-8
3.5.E	Isolation Condenser System	3/4.5-9
3.5.F	Minimum Core and Containment Cooling System Availability	3/4.5-11
3.5.G	Deleted	
3.5.H	Maintenance of Filled Discharge Pipe	3/4.5-13
3.5.I	Average Planar LHGR	3/4.5-15
3.5.J	Local Steady State LHGR	3/4.5-16
3.5.K	Local Transient LHGR	3/4.5-17
3.5.L	Minimum Critical Power Ratio (MCPR)	3/4.5-22
3.5.M	Condensate Pump Room Flood Protection	3/4.5-24
	Limiting Conditions for Operation Bases (3.5)	B 3/4.5-30
	Surveillance Requirement Bases (4.5)	B 3/4.5-38
3.6	Primary System Boundary	3/4.6-1
3.6.A	Thermal Limitations	3/4.6-1
3.6.B	Pressurization Temperature	3/4.6-2
3.6.C	Coolant Chemistry	3/4.6-3
3.6.D	Coolant Leakage	3/4.6-5
3.6.E	Safety and Relief Valves	3/4.6-6
3.6.F	Structural Integrity	3/4.6-7
3.6.G	Jet Pumps	3/4.6-10
3.6.H	Recirculation Pump Flow Limitations	3/4.6-12
3.6.I	Snubbers (Shock Suppressors)	3/4.6-16
	Limiting Conditions for Operation Bases (3.6)	B 3/4.6-25
	Surveillance Requirement Bases (4.6)	B 3/4.6-39
3.7	Containment Systems	3/4.7-1
3.7.A	Primary Containment	3/4.7-1
3.7.B	Standby Gas Treatment System	3/4.7-19
3.7.C	Secondary Containment	3/4.7-25
3.7.D	Primary Containment Isolation Valves	3/4.7-27
	Limiting Conditions for Operation Bases (3.7)	B 3/4.7-33
	Surveillance Requirement Bases (4.7)	B 3/4.7-40
3.8	Radioactive Effluents	3/4.8-1
3.8.A	Gaseous Effluents	3/4.8-1
3.8.B	Liquid Effluents	3/4.8-9
3.8.C	Mechanical Vacuum Pump	3/4.8-14
3.8.D	Radioactive Waste Storage	3/4.8-15
3.8.E	Radiological Environmental Monitoring Program	3/4.8-15
3.8.F	Solid Radioactive Waste	3/4.8-19
3.8.G	Miscellaneous Radioactive Material Sources	3/4.8-20
3.8.H	Miscellaneous LCO	3/4.8-21
	Limiting Conditions for Operation Bases (3.8)	B 3/4.8-32
	Surveillance Requirement Bases (4.8)	B 3/4.8-37
3.9	Auxiliary Electrical Systems	3/4.9-1
3.9.A	Requirements	3/4.9-1
3.9.B	Availability of Electric Power	3/4.9-2

(Table of Contents, Cont'd.)

		Page
4.3.D	Control Rod Accumulators	3/4.3-11
4.3.E	Reactivity Anomalies	3/4.3-12
4.3.F	(N/A)	
4.3.G	Automatic Generation Control System	3/4.3-13
4.4	Standby Liquid Control System	3/4.1-1
4.4.A	Normal Operation	3/4.4-1
4.4.B	Surveillance With Inoperable Components	3/4.4-2
4.4.C	Boron Solution	3/4.4-3
4.5	Core and Containment Cooling Systems	3/4.5-1
4.5.A	Core Spray and LPCI Subsystems	3/4.5-1
4.5.B	Containment Cooling Subsystem	3/4.5-5
4.5.C	HPCI Subsystem	3/4.5-6
4.5.D	Automatic Pressure Relief Subsystems	3/4.5-8
4.5.E	Isolation Condenser System	3/4.5-9
4.5.F	Core and Containment Cooling System	3/4.5-11
4.5.G	(Deleted)	
4.5.H	Maintenance of Filled Discharge Pipe	3/4.5-13
4.5.I	Average Planar Linear Heat Generation Rate	3/4.5-15
4.5.J	Linear Heat Generation Rate (LHGR)	3/4.5-16
4.5.K	Transient Linear Heat Generation Rate (LHGR)	3/4.5-17
4.5.L	Minimum Critical Power Ratio (MCPR)	3/4.5-22
4.5.M	Condensate Pump Room Flood Protection	3/4.5-24
4.6	Primary System Boundary	3/4.6-1
4.6.A	Thermal Limitations	3/4.6-1
4.6.B	Pressurization Temperature	3/4.6-2
4.6.C	Coolant Chemistry	3/4.6-3
4.6.D	Coolant Leakage	3/4.6-5
4.6.E	Safety and Relief Valves	3/4.6-6
4.6.F	Structural Integrity	3/4.6-7
4.6.G	Jet Pumps	3/4.6-10
4.6.H	Recirculation Pump Flow Limitations	3/4.6-12
4.6.I	Snubbers (Shock Suppressors)	3/4.6-16
4.7	Containment System	3/4.7-1
4.7.A	Primary Containment	3/4.7-1
4.7.B	Standby Gas Treatment System	3/4.7-19
4.7.C	Secondary Containment	3/4.7-25
4.7.D	Primary Containment Isolation Valves	3/4.7-27
4.8	Radioactive Effluents	3/4.8-1
4.8.A	Gaseous Effluents	3/4.8-1
4.8.B	Liquid Effluents	3/4.8-9
4.8.C	Mechanical Vacuum Pump	3/4.8-14
4.8.D	Radioactive Waste Storage	3/4.8-15
4.8.E	Radiological Environmental Monitoring Program	3/4.8-15
4.8.F	Solid Radioactive Waste	3/4.8-19
4.8.G	Miscellaneous Radioactive Materials Sources	3/4.8-20

List of Figures

		Page
Figure 2.1-3	APRM Bias Scram Relationship to Normal Operating Conditions	B 1/2.1-17
Figure 4.1.1	Graphical Aid in the Selection of an Adequate Interval Between Tests	B 3/4.1-18
Figure 4.2.2	Test Interval vs. System Unavailability	B 3/4.2-38
Figure 3.4.1	Standby Liquid Control Solution Requirements	3/4.4-4
Figure 3.4.2	Sodium Pentaborate Solution Temperature Requirements	3/4.4-5
Figure 3.5-1	MAPLHGR Limit vs Bundle Average Exposure ANF 8x8 Fuel (Sheet 1 of 2)	3/4.5-18
Figure 3.5-1	MAPLHGR Limit vs Bundle Average Exposure ANF 9x9 Fuel (Sheet 2 of 2)	3/4.5-19
Figure 3.5-1A	Steady State Linear Heat Generation Rate vs Nodal Exposure	3/4.5-20
Figure 3.5-1B	Transient Linear Heat Generation Rate vs Nodal Exposure	3/4.5-21
Figure 3.5-2	MCPR LIMIT for Reduced Total Core Flow (Sheet 1 of 3)	3/4.5-25
Figure 3.5-2	8x8 MCPR Operating Limit for Automatic Flow Control (Sheet 2 of 3)	3/4.5-26
Figure 3.5-2	9x9 MCPR Operating Limit for Automatic Flow Control (Sheet 3 of 3)	3/4.5-27
Figure 3.6.1	Minimum Temperature Requirements per Appendix G of 10 CFR 50	3/4.6-23
Figure 3.6.2	Thermal Power vs Core Flow Limits for Thermal Hydraulic Stability Surveillance In Single Loop Operation	3/4.6-24
Figure 4.6.1	Minimum Reactor Pressurization Temperature	B 3/4.6-29
Figure 4.6.2	Chloride Stress Corrosion Test Results at 500°F	B 3/4.6-31
Figure 4.8-1	Owner Controlled/Unrestricted Area Boundary	B 3/4.8-38
Figure 4.8-2	Detail of Central Complex	B 3/4.8-39
Figure 6.1-1	Offsite Organization	6-3
Figure 6.1-2	Station Organization	6-4

1.0 DEFINITIONS (Cont'd.)

plant can be operated safely and abnormal situations can be safely controlled.

J. Limiting Safety System Setting (LSSS) - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represents margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.

K. Steady State Linear Heat Generation Rate (SLHGR) - The steady state linear heat generation rate limit protects against exceeding the fuel end-of-life steady state design criteria developed by Advanced Nuclear Fuels.

L. Logic System Function Test - A logic system functional test means a test of all relays and contacts of a logic circuit from sensor to activated device to insure all components are operable per design intent. Where possible, action will go to completion, i.e., pumps will be started and valves opened.

M. Minimum Critical Power Ratio (MCPR) - The minimum in-core critical power ratio corresponding to the most limiting fuel assembly in the core.

N. Mode - The reactor mode is that which is established by the mode-selector-switch.

O. Operable - A system, subsystem, train, component, or device shall be operable when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

P. Operating - Operating means that a system, subsystem, train, component or device is performing its intended functions in its required manner.

Q. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

1.0 DEFINITIONS (Cont'd.)

AA. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alternations are being performed. When the mode switch is placed in the shutdown position a reactor scram is initiated, power to the control rod drives is removed, and the reactor protection system trip systems are de-energized.

1. Hot Shutdown means conditions as above with reactor coolant temperature greater than 212°F.

2. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F.

BB. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

CC. Surveillance Interval - Each surveillance requirement shall be performed within the specified surveillance interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval.

b. A total maximum combined interval time for any 3 consecutive intervals not to exceed 3.25 times the specified surveillance interval.

DD. Fraction of Rated Power (FRP) - The fraction of rated power is the ratio of core thermal power to rated thermal power of 2527 Mwth.

EE. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

FF. Fuel Design Limiting Ratio for Exxon Fuel (FDLRX) - The fuel design limiting ratio for Exxon fuel is the limit used to assure that the fuel operates within the end-of-life steady state design criteria. FDLRX assures acceptable end-of-life conditions by, among other items, limiting the release of fission gas to the cladding plenum.

GG. Dose Equivalent I-131 - That concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

1.0 DEFINITIONS (Cont'd.)

HH. Process Control Program (PCP) - Contains the sampling, analysis, and formulation determination by which solidification of radioactive wastes from liquid systems is assured.

II. Offsite Dose Calculation Manual (ODCM) - Contains the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, and in the calculation of gaseous and liquid effluent monitor alarm/trip setpoints.

JJ. Channel Functional Test (Radiation Monitor) - Shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify operability including alarm and/or trip functions.

KK. Source Check - The qualitative assessment of instrument response when the sensor is exposed to a radioactive source.

LL. Member(s) of the Public - Shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

MM. Rated Recirculation Pump Speed - is the recirculation pump speed that corresponds to rated core flow (98×10^6 lb/hr) when operating at rated thermal power (dual loop operation).

NN. Dual Loop Operation - reactor power operation with both recirculation pumps running.

OO. Single Loop Operation (SLO) - reactor power operation with one recirculation pump running.

PP. Transient Linear Heat Generation Rate (TLHGR) - The transient linear heat generation rate limit protects against fuel centerline melting and 1% plastic cladding strain during transient conditions throughout the life of the fuel.

QQ. Fuel Design Limiting Ratio for Centerline Melt (FDLRC) - The fuel design limiting ratio for centerline melt is the limit used to assure that the fuel will neither experience centerline melt nor exceed 1% plastic cladding strain for transient overpower events beginning at any power and terminating at 120% of rated thermal power.

RR. Linear Heat Generation Rate (LHGR) - The linear heat generation rate is the operating fuel pin power level.

1.1 SAFETY LIMIT

FUEL CLADDING INTEGRITY

Applicability:

The Safety Limits established to preserve the fuel cladding integrity apply to these variables which monitor the fuel thermal behavior.

Objective:

The objective of the Safety Limits is to establish limits below which the integrity of the fuel cladding is preserved.

Specifications:

A. Reactor Pressure greater than 800 psig and Core Flow greater than 10% of Rated.

The existence of a minimum critical power ratio (MCPR) less than 1.05 shall constitute a violation of the MCPR fuel cladding integrity safety limit.

When in Single Loop Operation, the MCPR safety limit shall be increased by 0.01.

2.1 LIMITING SAFETY SYSTEM SETTING

FUEL CLADDING INTEGRITY

Applicability:

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objective:

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specifications:

A. Neutron Flux Trip Settings

The limiting safety system trip settings shall be as specified below:

1. APRM Flux Scram Trip Setting (Run Mode)

When the reactor mode switch is in the run position, the APRM flux scram setting shall be:

S less than or equal to [$.58W_D$ + 62] during Dual Loop Operation or S less than or equal to [$.58W_D$ + 58.5] during Single Loop Operation with a maximum setpoint of 120% for core flow equal to 98×10^6 lb/hr and greater, where:

S - setting in percent of rated thermal power.

1.1 SAFETY LIMIT (Cont'd.)

2.1 LIMITING SAFETY SYSTEM SETTING (Cont'd.)

W_D = percent of drive flow required to produce a rated core flow of 98 Mlb/hr.

In the event of operation of any fuel assembly with a fuel design limiting ratio for centerline melt (FDLRC) greater than 1.0, the setting shall be modified as follows:

Where: S is less than or equal to $(.58W_D + 62)$/FDLRC during Dual Loop Operation or $(.58\ W_D + 58.5)$/FDLRC during Single Loop Operation

The value of FDLRC shall be set equal to 1.0 unless the actual operating value is greater than 1.0, in which case the actual operating value will be used.

This adjustment may also be performed by increasing the APRM gain by FDLRC which accomplishes the same degree of protection as reducing the trip setting by 1/FDLRC.

2. APRM Flux Scram Trip Setting (Refuel or Startup and Hot Standby Mode)

1.1 SAFETY LIMIT (Cont'd.)

B. Core Thermal Power Limit (Reactor Pressure is less than or equal to 800 psig)

When the reactor pressure is less than or equal to 800 psig or core flow is less than 10% of rated, the core thermal power shall not exceed 25 percent of rated thermal power.

2.1 LIMITING SAFETY SYSTEM SETTING (Cont'd.)

When the reactor mode switch is in the refuel or the startup/hot standby position, the APRM scram shall be set at less than or equal to 15% of rated core thermal power.

3. IRM Flux Scram Trip Setting

The IRM flux scram setting shall be set at less than or equal to 120/125 of full scale

B. APRM Rod Block Setting

The APRM rod block setting shall be:

S is less than or equal to $[.58W_D + 50]$ during Dual Loop Operation or S is less than or equal to $[.58\ W_D + 46.5]$ during Single Loop Operation.

The definitions used above for the APRM scram trip apply.

In the event of operation of any fuel assembly with a fuel design limiting ratio for centerline melt (FDLRC) greater than 1.0, the setting shall be modified as follows:

S is less than or equal to $(.58W_D + 50)/FDLRC$ during Dual Loop Operation or S is less than or equal to $(.58W\ W_D + 46.5)/FDLRC$ during Single Loop Operation

The definitions used above for the APRM scram trip apply.

The value of FDLRC shall be set equal to 1.0 unless the actual operating value is greater than 1.0. In which case the actual operating value will be used.

1.1 SAFETY LIMIT (Cont'd.)

C. Power Transient

1. The neutron flux shall not exceed the scram setting established in Specification 2.1.A for longer than 1.5 seconds as indicated by the process computer.

2. When the process computer is out of service, this safety limit shall be assumed to be exceeded if the neutron flux exceeds the scram setting established by Specification 2.1.A and a control rod scram does not occur.

D. Reactor Water Level (Shutdown Condition)

Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 12 inches above the top of the active fuel when it is seated in the core.

Note: Top of active fuel is defined to be 360 inches above vessel zero (see Bases 3.2).

2.1 LIMITING SAFETY SYSTEM SETTING (Cont'd.)

The adjustment may also be performed by increasing the APRM gain by FDLRC which accomplishes the same degree of protection as reducing the trip setting by 1/FDLRC.

C. Reactor low water level scram setting shall be greater than or equal to 144" above the top of the active fuel at normal operating conditions.

Note: Top of active fuel is defined to be 360 inches above vessel zero (see Bases 3.2).

D. Reactor low water level ECCS initiation shall be 84" (plus 4",minus 0") above the top of the active fuel at normal operating conditions.

Note: Top of active fuel is defined to be 360 inches above vessel zero (see Bases 3.2).

1.1 SAFETY LIMIT BASES (Cont'd.)

power ratio (CPR) which is the ratio of the bundle power which would produce the onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the Minimum Critical Power Ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables. (Figure 2.1-3).

The MCPR Fuel Cladding Integrity Safety Limit assures sufficient conservatism in the operating MCPR limit that in the event of an anticipated operational occurrence from the limiting condition for operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (MCPR=1.00) and the MCPR Fuel Cladding Integrity Safety Limit is based on a detailed statistical procedure which considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the safety limit is the uncertainty inherent in the XN-3 critical power correlation. Refer to XN-NF-524 for the methodology used in determining the MCPR Fuel Cladding Integrity Safety Limit.

The XN-3 critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power as evaluated by the correlation is within a small percentage of the actual critical power being estimated. The assumed reactor conditions used in defining the safety limit introduce conservatism into the limit because boundingly high radial power peaking factors and boundingly flat local peaking distributions are used to estimate the number of rods in boiling transition. Still further conservatism is induced by the tendency of the XN-3 correlation to overpredict the number of rods in boiling transition. These conservatisms and the inherent accuracy of the XN-3 correlation provide a reasonable degree of assurance that during sustained operation at the MCPR Fuel Cladding Integrity Safety Limit there would be no transition boiling in the core. If boiling transition were to occur, however, there is reason to believe that the integrity of the fuel would not necessarily be compromised. Significant test data accumulated by the U.S. Nuclear Regulatory Commission and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach; much of the data indicates that LWR fuel can survive for an extended period in an environment of transition boiling.

During Single Loop Operation, the MCPR safety limit is increased by 0.01 to conservatively account for increased uncertainties in the core flow and TIP measurements.

1.1 SAFETY LIMIT BASES (Cont'd.)

If the reactor pressure should ever exceed the limit of applicability of the XN-3 critical power correlation as defined in XN-NF-512, it would be assumed that the MCPR Fuel Cladding Integrity Safety Limit had been violated. This applicability pressure limit is higher than the pressure safety limit specified in Specification 1.2.

For fuel fabricated by Advanced Nuclear Fuels Corporation (ANF), fuel design criteria have been established to provide protection against fuel centerline melting and 1% plastic cladding strain during transient overpower conditions throughout the life of the fuel. To demonstrate compliance with these criteria, fuel rod centerline temperatures are determined at 120% overpower conditions as a check against calculated centerline melt temperatures. FDLRC is incorporated to protect the above criteria at all power levels considering events which will cause the reactor power to increase to 120% of rated thermal power.

B. Core Thermal Power Limit (Reactor Pressure less than 800 psia)

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a flow of 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow

2.1 LIMITING SAFETY SYSTEM SETTING BASES (Cont'd.)

At times it may be necessary to operate with one reactor coolant recirculation pump out of service. During Single Loop Operation, the normal drive flow relationship during Dual Loop Operation is altered. This is the result of reverse flow through the idle loop jet pumps when the active loop recirculation pump speed is above 20 to 40% of rated. Some of the active loop flow is then diverted from the core and backflows through the idle loop jet pumps; hence, the core receives less flow than would be predicted based upon the Dual Loop drive flow to core flow relationship. If the APRM flow biased trip settings were not altered for Single Loop Operation, the new drive flow to core flow relationship would nonconservatively result in flow biased trips occurring at neutron fluxes higher than normal for a given core flow.

The scram trip setting must be adjusted to ensure that the LHGR transient limit is not violated for any power distribution. This is expressed as the fuel design limiting ratio for centerline melt (FDLRC). The scram setting is adjusted in accordance with the formula in specification 2.1.A.1 when FDLRC is greater than 1.0.

The adjustment may also be accomplished by increasing the APRM gain by FLDRC. This provides the same degree of protection as reducing the trip setting by 1/FDLRC by raising the initial APRM reading closer to the trip setting such that a scram would be received at the same point in a transient as if the trip setting had been reduced.

2. APRM Flux Scram Trip Setting
(Refuel or Start & Hot Standby Mode)

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because

2.1 LIMITING SAFETY SYSTEM SETTING BASES (Cont'd.)

B. APRM Rod Block Trip Setting

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent gross rod withdrawal at constant recirculation flow rate to protect against grossly exceeding the MCPR fuel cladding integrity safety limit. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore the worse case MCPR which could occur during steady-state operation is at 108% of rated thermal power during dual loop operation or 104.5 percent during single loop operation because of the APRM rod block trip setting. As with the APRM flow biased scram, the reduced setpoint during single loop operation accounts for possible reverse flow in the idle loop jet pumps. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward or APRM gain increased if the fuel design limiting ratio for centerline melt (FDLRC) for any fuel assembly exceeds 1.0, thus preserving the APRM rod block safety margin.

C. Reactor Low Water Level Scram - The reactor low water level scram is set at a point which will assure that the water level used in the bases for the safety limit is maintained. The scram setpoint is based on normal operating temperature and pressure conditions because the level instrumentation is density compensated.

D. Reactor Low Low Water Level ECCS Initiation Trip Point - The emergency core cooling subsystems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident and to limit fuel clad temperature to well below the clad melting temperature to assure that core geometry remains intact and to limit any clad metal-water reaction to less than 1%. To accomplish their intended function, the capacity of each emergency core cooling system component was established based on the reactor low water level scram setpoint. To lower the setpoint of the low water level scram would require an increase in the capacity requirement for each of the ECCS components. Thus, the reactor vessel low water level scram was set low enough to permit margin for operation, yet was not set lower because of ECCS capacity requirements.

3.1 LIMITING CONDITIONS FOR OPERATION

REACTOR PROTECTION SYSTEM

Applicability:

Applies to the instrumentation and associated devices which initiates a reactor scram.

Objective:

To assure the operability of the reactor protection system.

Specification:

A. Reactor Protection System

1. The setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.1. The system response times from the opening of the sensor contact up to and including the opening of the trip actuator contacts shall not exceed 50 milliseconds.

2. If during operation, the fuel design limiting ratio for centerline melt (FDLRC) for any fuel assembly exceeds 1.0 when operating above 25% rated thermal power, either:

4.1 SURVEILLANCE REQUIREMENTS

REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective:

To specify the type and frequency of surveillance to be applied to the protection instrumentation.

Specification:

A. Reactor Protection System

1. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.1 and 4.1.2, respectively.

2. Daily during reactor power operation above 25% rated thermal power, the core power distribution shall be checked for:

3.1 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

a. The APRM scram and rod block settings shall be reduced to the values given by the equations in Specifications 2.1.A.1 and 2.1.B. This may be accomplished by increasing APRM gains as described therein.

b. The power distribution shall be changed such that the fuel design limiting ratio for centerline melt (FDLRC) for any fuel assembly no longer exceeds 1.0.

3. Two RPS electric power monitoring channels for each inservice RPS MG set or alternate source shall be OPERABLE at all times.

4.1 SURVEILLANCE REQUIREMENTS (Cont'd.)

a. Maximum fuel design limiting ratio for centerline melt (FDLRC).

b. Deleted.

3. The RPS power monitoring system instrumentation shall be determined OPERABLE:

a. At least once per 6 months by performing a CHANNEL FUNCTIONAL TEST, and

4.1 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

A comparison of Tables 4.1.1 and 4.1.2 indicates that six instrument channels have not been included in the latter Table. These are: Mode Switch in Shutdown, Manual Scram, High Water Level in Scram Discharge Volume Float Switches, Main Steam Line Isolation Valve Closure, Generator Load Rejection, and Turbine Stop Valve Closure. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration is not applicable; i.e., the switch is either on or off. Further, these switches are mounted solidly to the device and have a very low probability of moving; e.g., the switches in the scram discharge volume tank. Based on the above, no calibration is required for these six instrument channels.

B. The FDLRC shall be checked once per day to determine if the APRM gains or scram requires adjustment. This may normally be done by checking in the LPRM readings, TIP traces, or process computer calculations.

Only a small number of control rods are moved daily and thus the peaking factors are not expected to change significantly and thus a daily check of the FDLRC is adequate.

Table 3.2.3

INSTRUMENTATION THAT INITIATES ROD BLOCK

Minimum No. of Operable Inst. Channels Per Trip System (1)	Instrument	Trip Level Setting
1	APRM upscale, (flow bias) (7)	
	Dual Loop Operation	Less than or equal to (.58 W_D plus 50)/FDLRC (See Note 2)
	Single Loop Operation	Less than or equal to (.58 W_D plus 46.5)/FDLRC (See Note 2)
1	APRM upscale (refuel and Startup/Hot Standby mode)	Less than or equal to 12/125 full scale
2	APRM downscale (7)	Greater than or equal to 3/125 full scale
1	Rod block monitor upscale (flow bias) (7)	
	Dual Loop Operation	Less than or equal to (.65 W_D plus 45) (See Note 2)
	Single Loop Operation	Less than or equal to (.65 W_D plus 41) (See Note 2)
1	Rod block monitor downscale (7)	Greater than or equal to 5/125 full scale
3	IRM downscale (3)	Greater than or equal to 5/125 full scale
3	IRM upscale	Less than or equal to 108/125 full scale
3	IRM detector not fully inserted in the core	N/A
2 (5)	SRM detector not in startup position	(See Note 4)
2 (5) (6)	SRM upscale	Less than or equal to 10^5 counts/sec.
1	Scram discharge volume water level - high	Less than or equal to 25 gallons

Notes: (See Next Page)

TABLE 3.2.3 (Notes)

1. For the Startup/Hot Standby and Run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function, except the SRM rod blocks, IRM upscale, IRM downscale and IRM detector not fully inserted in the core need not be operable in the "Run" position and APRM downscale, APRM upscale (flow bias), and RBM downscale need not be operable in the Startup/Hot Standby mode. A RBM upscale need not be operable at less than 30% rated thermal power. One channel may be bypassed above 30% rated thermal power provided that a limiting control rod pattern does not exist. For systems with more than one channel per trip system, if the first column cannot be met for both trip systems, the systems shall be tripped. For the Scram Discharge Volume water level high rod block, there is one instrument channel per bank.

2. W_D percent of drive flow required to produce a rated core flow of 98 Mlb/hr. FDLRC = fuel design limiting ratio for centerline melt.

3. IRM downscale may be bypassed when it is on its lowest range.

4. This function may be bypassed when the count rate is greater than or equal to 100 cps.

5. One of the four SRM inputs may be bypassed.

6. This SRM function may be bypassed in the higher IRM ranges when the IRM upscale Rod Block is operable.

7. Not required while performing low power physics test at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

D. Automatic Pressure Relief Subsystems

1. Except as specified in 3.5.D.2 and 3.5.D.3 below, the Automatic Pressure Relief Subsystem shall be operable whenever the reactor pressure is greater than 90 psig and irradiated fuel is in the reactor vessel.

2. From and after the date that one of the five relief valves of the automatic pressure relief subsystem is made or found to be inoperable

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

D. Surveillance of the Automatic Pressure Relief Subsystem shall be performed as follows:

1. During each operating cycle the following shall be performed:

   a. A simulated automatic initiation which opens all pilot valves, and

   b. With the reactor at pressure each relief valve shall be manually opened. Relief valve opening shall be verified by a compensating turbine bypass valve or control valve closure.

   c. A logic system functional test shall be performed each refueling outage.

2. Whenever HPCI is required to be operable in accordance with 3.5.D.2, HPCI shall be tested to demonstrate operability immediately.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

reactor operation is permissible only during the succeeding seven (7) days provided that during such time the HPCI subsystem is operable. If the following MAPLHGR reduction factors (multipliers) are applied to Figure 3.5-1, the Automatic Pressure Relief Subsystem of ECCS shall be considered operable: (1) 0.89 for 8x8 fuel, or (2) 0.76 for 9x9 fuel.

3. From and after the date that two relief valves are found or made to be inoperable, reactor operation is permissible only during the succeeding seven days provided that during such time the HPCI subsystem is operable and the multipliers specified in 3.5.D.2 are applied.

4. If the requirements of 3.5.D.1 cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to below 90 psig within 24 hours.

E. Isolation Condenser System

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

3. Whenever HPCI is required to be operable in accordance with 3.5.D.3, HPCI shall be tested to demonstrate operaability immediately.

E. Surveillance of the Isolation Condenser System shall be performed as follows:

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

I. Average Planar LHGR

During steady state power operation, the Average Planar Linear Heat Generation Rate (APLHGR) of all the rods in any fuel assembly as a function of average bundle exposure at any axial location, shall not exceed the maximum average planar LHGR shown in Figure 3.5-1 (consisting of two curves). For operation during Single Loop Operation the values of Figure 3.5-1 shall be decreased by a multiplicative factor of 0.91. If, concurrently, one Automatic Pressure Relief Subsystem relief valve is out-of-service, the values of Figure 3.5-1 shall be decreased by a multiplicative factor of 0.89 for 8x8 fuel and 0.76 for 9x9 fuel. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

I. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average bundle exposure shall be determined daily during reactor operation at greater than or equal to 25% rated thermal power.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

J. LOCAL STEADY STATE LHGR

During steady state power operation above 25% rated thermal power, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed its maximum steady state LHGR (SLHGR) value shown in Figure 3.5-1A (consists of two curves). That is, the Fuel Design Limiting Ratio for Exxon Fuel (FDLRX) shall not be greater than 1.0 where $FDLRX = \frac{LHGR}{SLHGR}$

Figure 3.5-1A depicts the SLHGR values for 8x8 and 9x9 fuel as a function of nodal exposure.

If at any time during operation above 25% rated thermal power, it is determined by normal surveillance that FDLRX for any fuel assembly exceeds 1.0, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the FDLRX is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

J. Linear Heat Generation Rate (LHGR)

The Fuel Design Limiting Ratio for Exxon Fuel (FDLRX) shall be checked daily during reactor operation at greater than or equal to 25% rated thermal power.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd)

K. Local Transient LHGR

At any time during power operation, above 25% rated thermal power the fuel design limiting ratio for centerline melt (FDLRC) shall not be greater than 1.0, where

$$FDLRC = \frac{(LHGR)\ (1.2)}{(TLHGR)\ (FRP)}$$

Figure 3.5-1B depicts the TLHGR values for 8x8 and 9x9 fuel as a function of nodal exposure.

If during operation, the FDLRC exceeds 1.0 when operating above 25% rated thermal power, either:

a. The APRM scram and rod block settings shall be reduced to the values given by the equations in Specifications 2.1.A.1 and 2.1.B. This may be accomplished by increasing APRM gains as described therein.

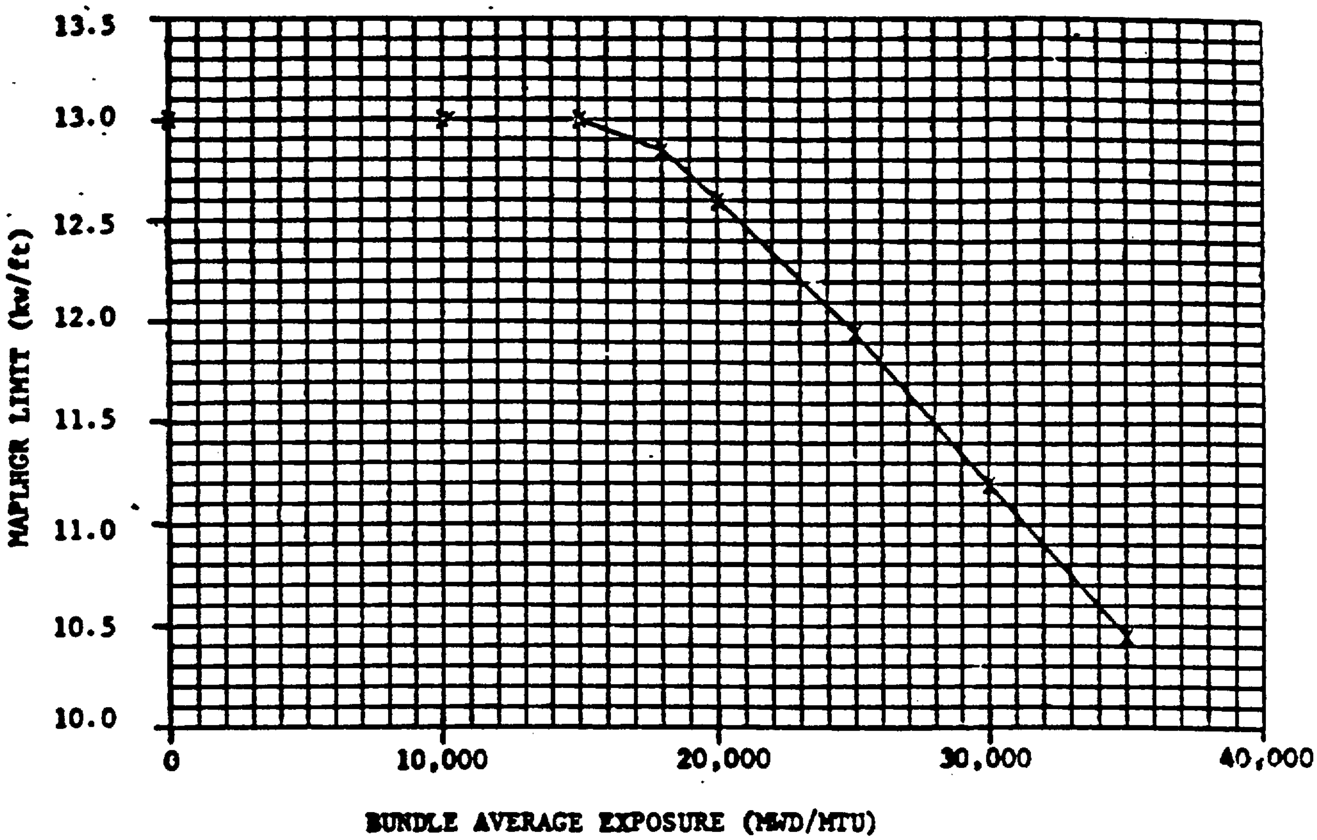
b. The power distribution shall be changed such that the FDLRC no longer exceeds 1.0.

4.5 SURVEILLANCE REQUIREMENT (Cont'd)

K. Transient Linear Heat Generation Rate (LHGR)

The fuel design limiting ratio for centerline melt (FDLRC) shall be checked daily during reactor operation at greater than or equal to 25% rated thermal power

MAPLHGR LIMIT VS. BUNDLE AVERAGE EXPOSURE
ANF 8x8 FUEL

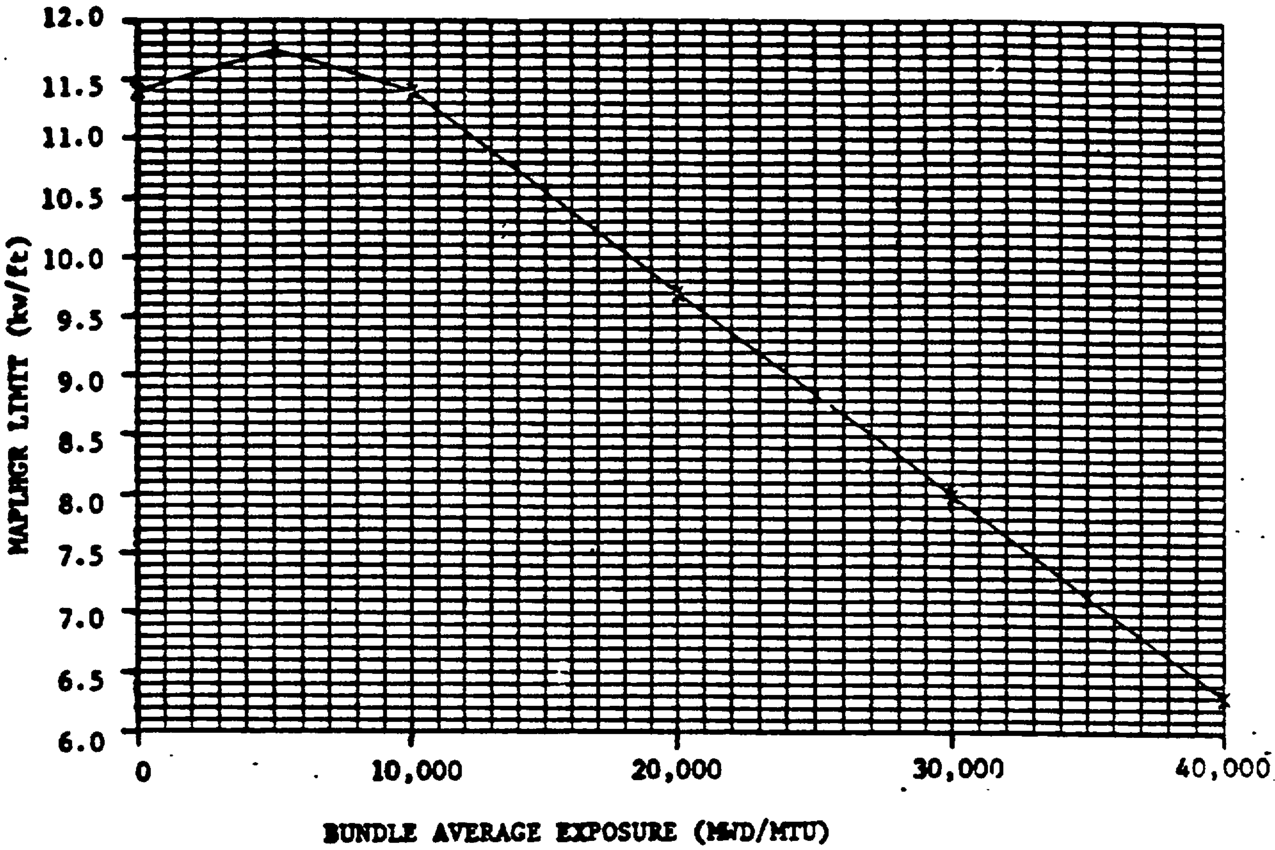


The above graph is based on the following MAPLHGR summary for ANF 8x8 fuel design:

Bundle Average Exposure (MWD/MTU)	MAPLHGR Limit (kw/ft)
0	13.0
10,000	13.0
15,000	13.0
18,000	12.85
20,000	12.60
25,000	11.95
30,000	11.20
35,000	10.45

(Figure 3.5-1)
(Sheet 1 of 2)

MAPLHGR LIMIT VS. BUNDLE AVERAGE EXPOSURE
ANF 9x9 FUEL

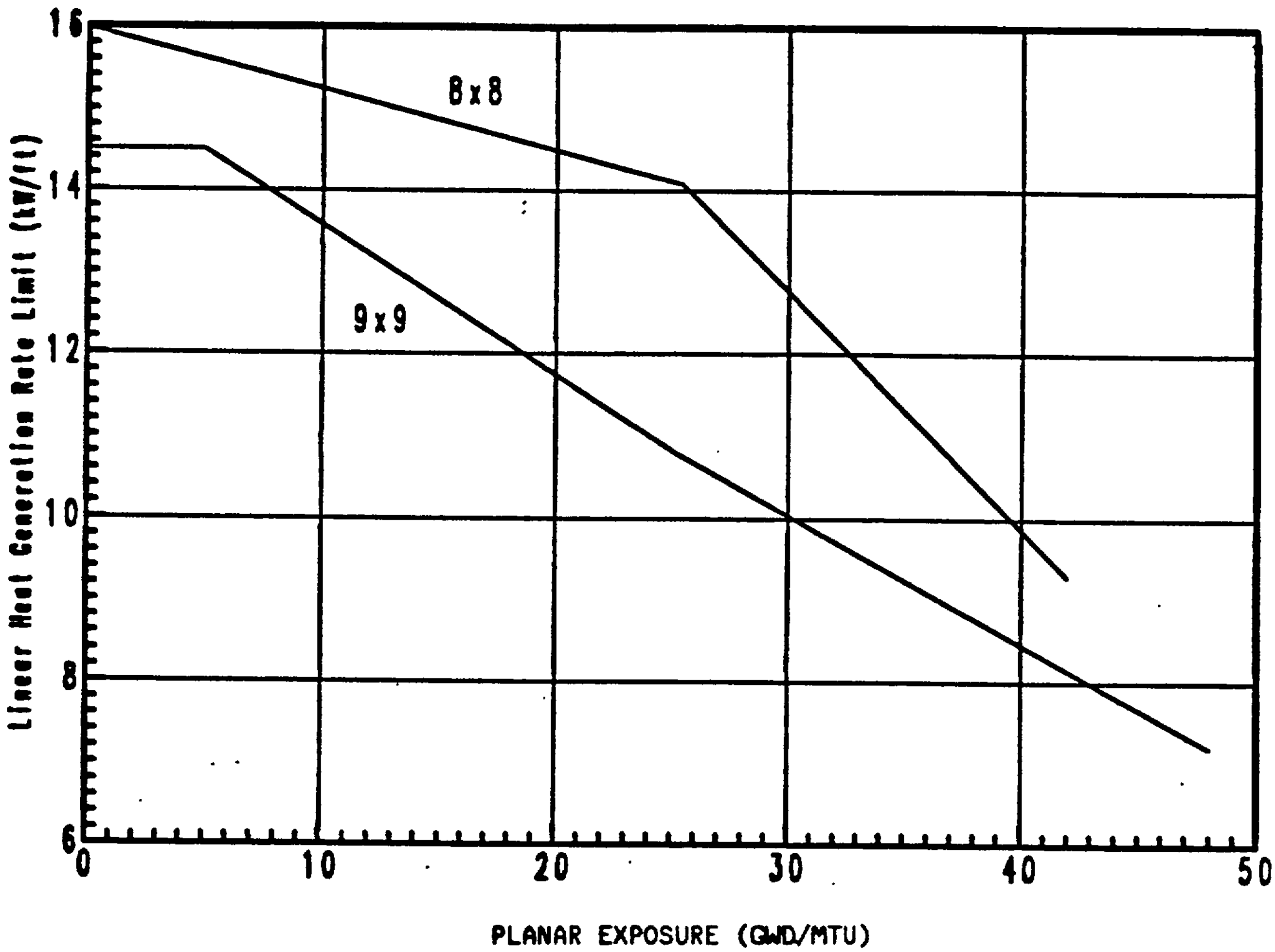


The above graph is based on the following MAPLHGR summary for ANF 9x9 fuel design:

Bundle Average Exposure (MWD/MTU)	MAPLHGR Limit (kw/ft)
0	11.40
5,000	11.75
10,000	11.40
15,000	10.55
20,000	9.70
25,000	8.85
30,000	8.00
35,000	7.15
40,000	6.30

Figure 3.5-1
(Sheet 2 of 2)

STEADY STATE LINEAR HEAT GENERATION RATE LIMIT (SLHGR)
VS. NODAL EXPOSURE



8x8 Fuel

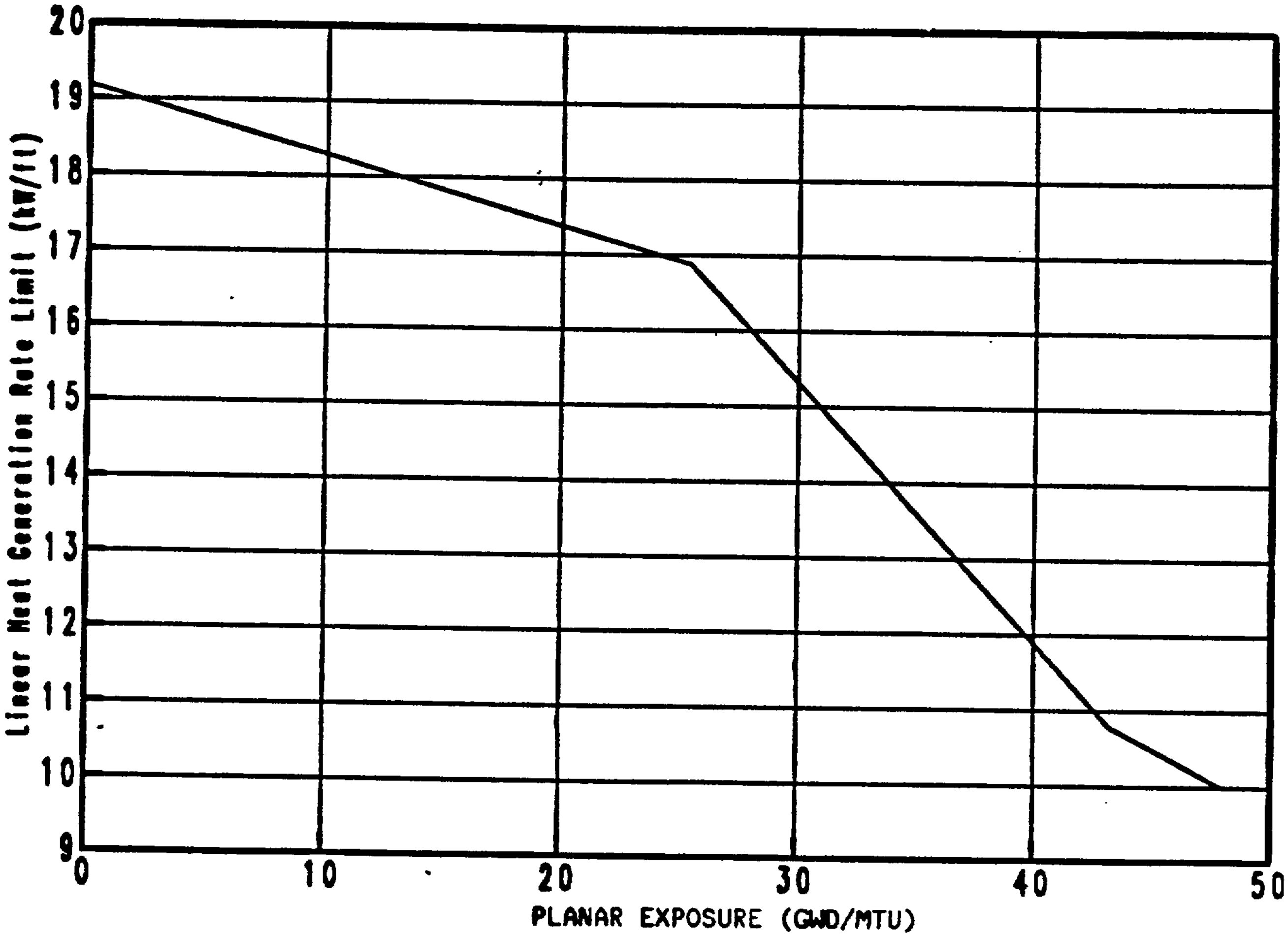
Exposure	LHGR
0.00	16.00
25.40	14.10
42.00	9.30

9x9 Fuel

Exposure	LHGR
0.00	14.50
5.00	14.50
25.20	10.80
48.00	7.20

Figure 3.5-1A

TRANSIENT LINEAR HEAT GENERATION RATE LIMIT (TLHGR) VS. NODAL EXPOSURE
FOR ALL FUEL TYPES



Exposure	LHGR
0.00	19.20
25.40	16.90
43.20	10.80
48.00	10.00

Figure 3.5-1B

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

L. Minimum Critical Power Ratio (MCPR)

During steady state operation at rated core flow, MCPR shall be greater than or equal to 1.39. For core flows other than rated, the MCPR Operating Limit shall be as follows:

1. Manual Flow Control - the MCPR Operating Limit shall be the value from Figure 3.5-2 sheet 1 or the above rated flow value, whichever is greater.

2. Automatic Flow Control - the MCPR Operating Limit is the greatest of the following:
   a. The above rated flow value;
   b. The value from Figure 3.5-2 sheet 1; or
   c. The interpolated value from Figure 3.5-2 sheets 2 and 3.

3. During Single Loop Operation, the rated flow MCPR operating limit shall be increased by an additive factor of 0.01.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

L. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during a reactor power operation at greater than or equal to 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

If at any time during steady state power operation, it is determined that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

M. Condensate Pump Room Flood Protection

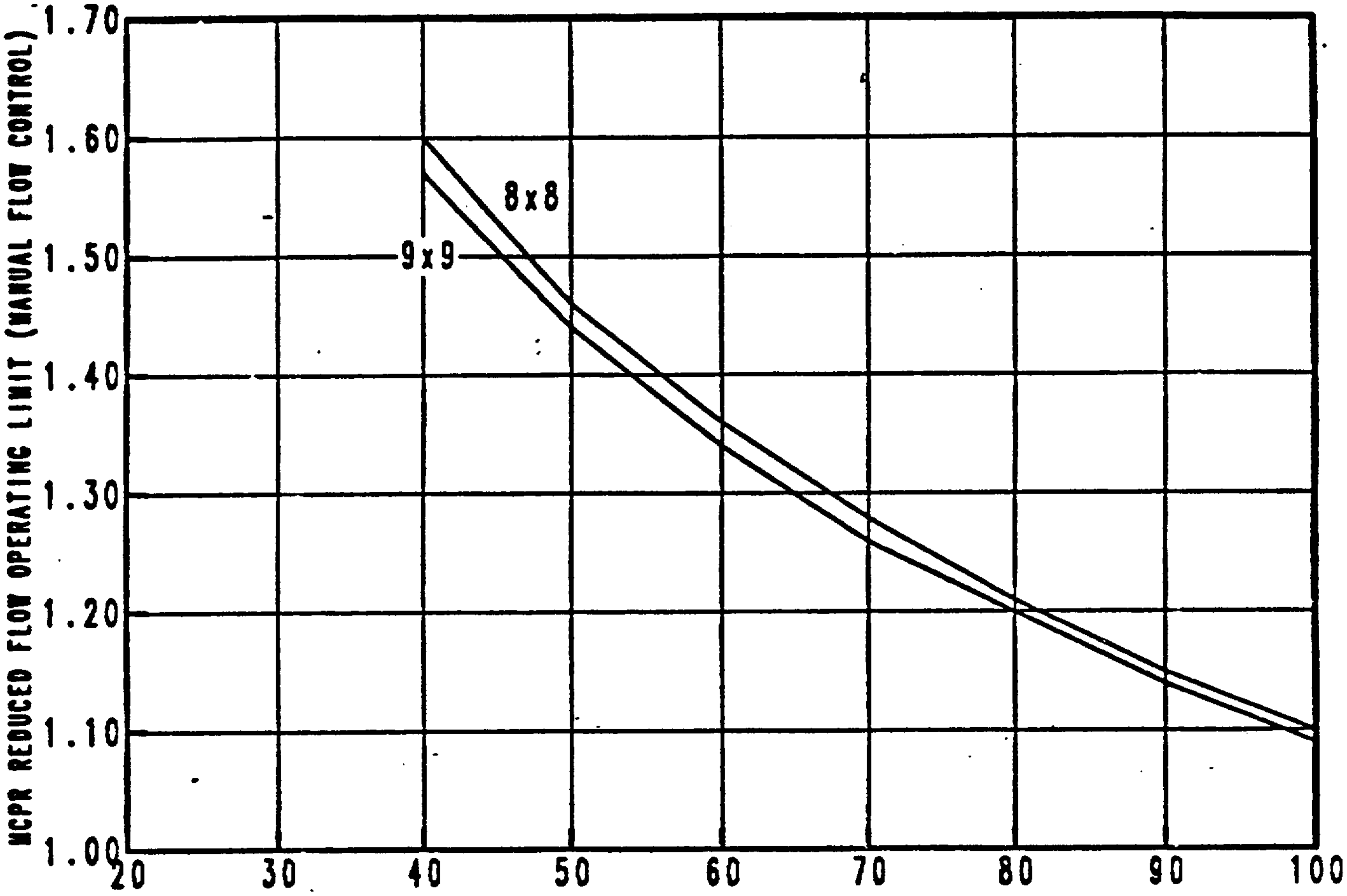
1. The system is installed to prevent or mitigate the consequences of flooding of the condensate pump room and shall be operable prior to startup of the reactor.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

M. Condensate Pump Room Flood Protection

1. The following surveillance requirements shall be observed to assure that the condensate pump room flood protection is operable.

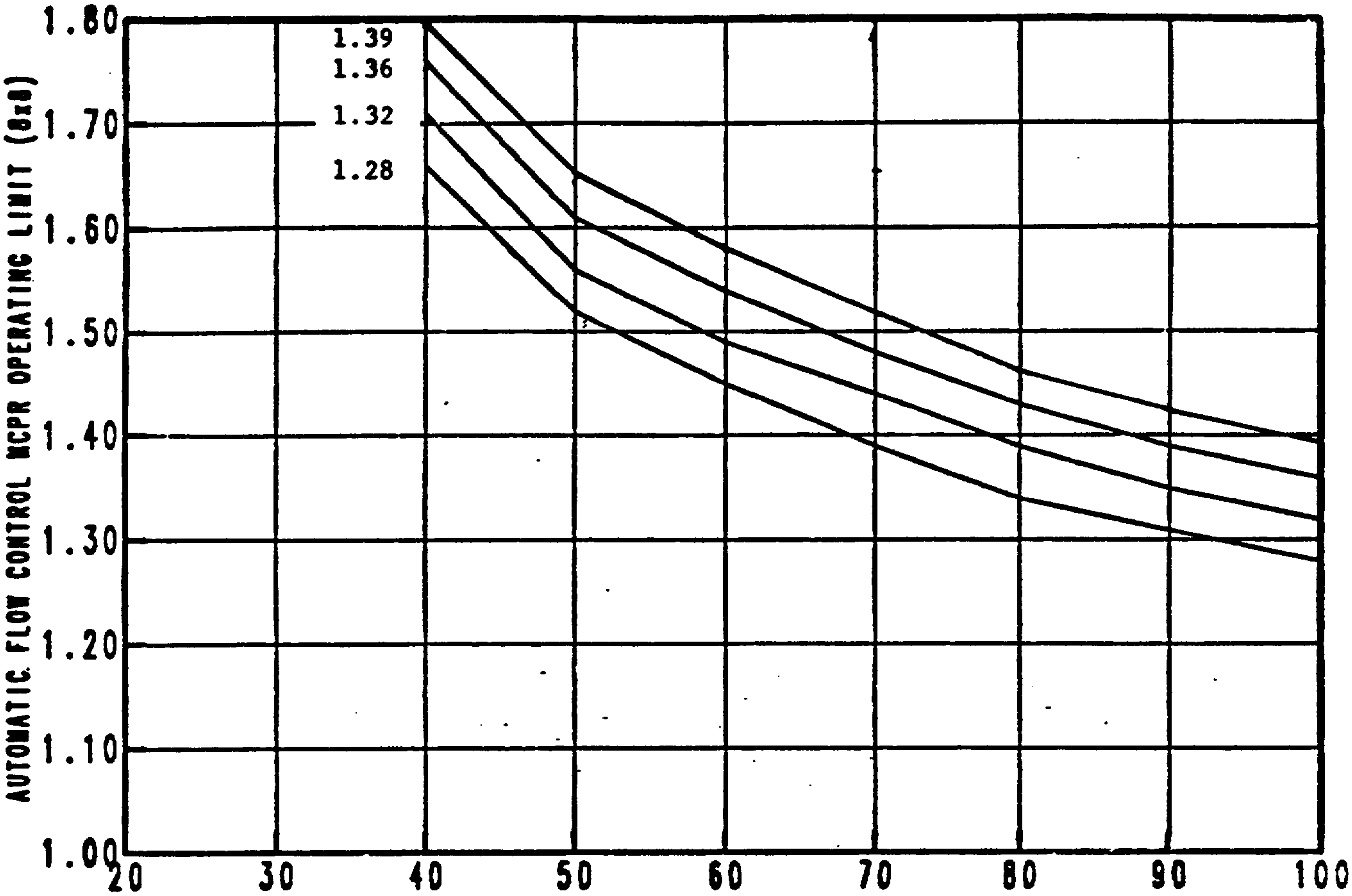
   a. The testable penetrations through the walls of CCSW pump vaults shall be checked during each operating cycle by pressurizing to 15 plus or minus 2 psig and checking for leaks using a soap bubble solution. The criteria for acceptance should be no visible leakage through the soap bubble solution. The bulkhead door shall be checked during each operating cycle by hydrostatically testing the door at 15 plus or minus 2 psig and checking to verify that leakage around the door is less than one gallon per hour.



The above curves are based on the following MCPR Limit summary for reduced Total Core Flow:

Total Core Flow (% Rated)	MCPR Limit 8x8	MCPR Limit 9x9
100	1.10	1.09
90	1.15	1.14
80	1.21	1.20
70	1.28	1.26
60	1.36	1.34
50	1.46	1.44
40	1.60	1.57

Figure 3.5-2 (Sheet 1 of 3)
MCPR Limit for reduced Total Core Flow



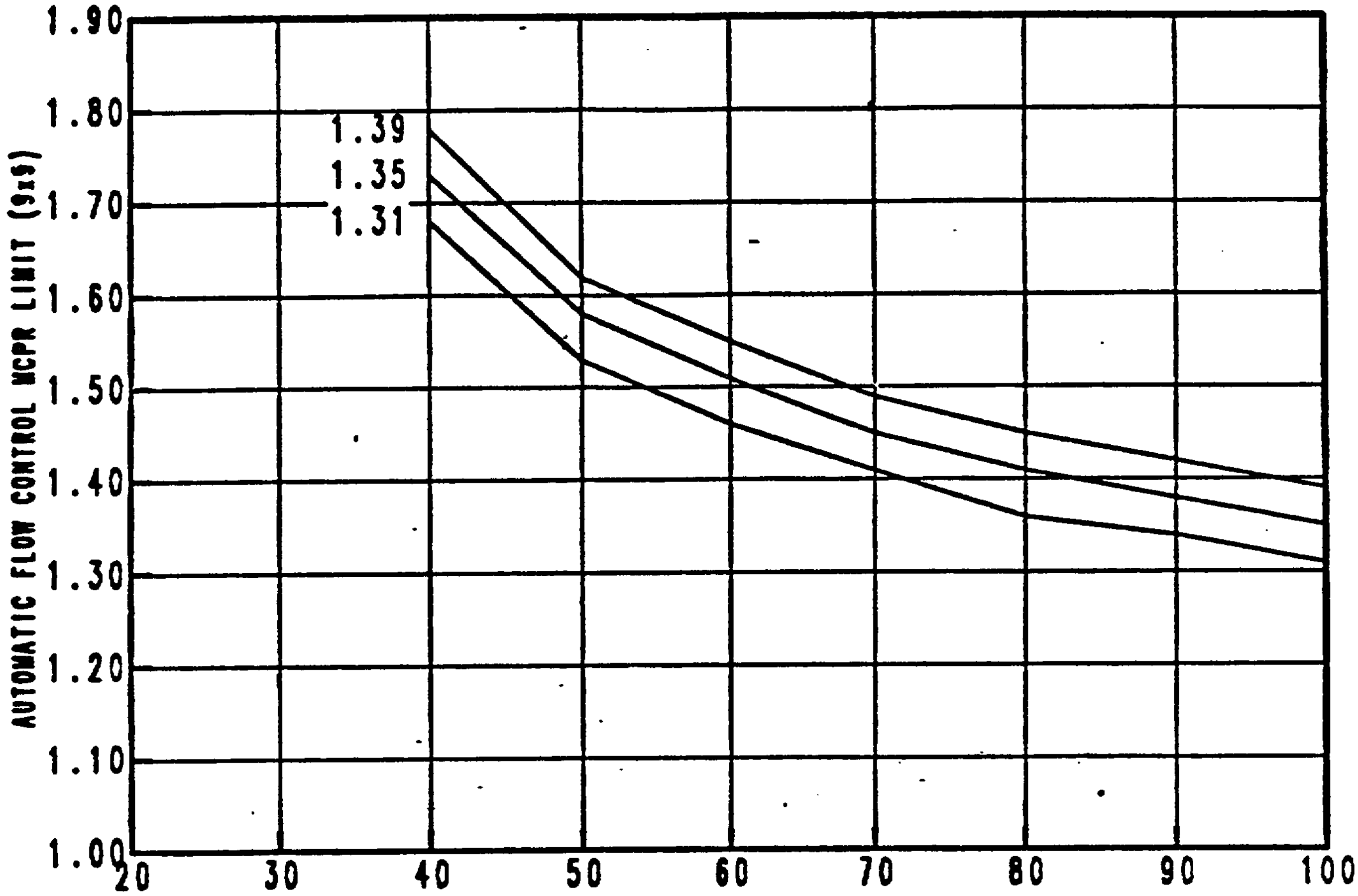
TOTAL CORE FLOW (% RATED, 98 MLB/HR)

The above 8x8 curves are based on the following MCPR operating limit summary for Automatic Flow Control:

Total Core Flow (% Rated)	MCPR Operating Limit for 8x8 fuel* 1.28	1.32	1.36	1.39
100	1.28	1.32	1.36	1.39
90	1.31	1.35	1.39	1.43
80	1.34	1.39	1.43	1.46
70	1.39	1.44	1.48	1.52
60	1.45	1.49	1.54	1.58
50	1.52	1.56	1.61	1.65
40	1.66	1.71	1.76	1.80

* Column headers are MCPR Operating Limits at rated flow.

Figure 3.5-2 (Sheet 2 of 3)
8x8 MCPR Operating Limit For Automatic Flow Control



TOTAL CORE FLOW (% RATED, 98 MLB/HR)

The above 9x9 curves are based on the following MCPR operating limit summary for Automatic Flow Control:

Total Core Flow (% Rated)	MCPR Operating Limit for 9x9 fuel* 1.31	1.35	1.39
100	1.31	1.35	1.39
90	1.34	1.38	1.42
80	1.36	1.41	1.45
70	1.41	1.45	1.49
60	1.46	1.51	1.55
50	1.53	1.58	1.62
40	1.68	1.73	1.78

* Column headers are MCPR Operating Limits at rated flow.

Figure 3.5-2 (Sheet 3 of 3)
9x9 MCPR Operating Limit For Automatic Flow Control

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

b. The CCSW Vault Floor drain shall be checked during each operating cycle by assuring that water can be run through the drain line and actuating the air operated valves by operation of the following sensor:

   i. loss of air

   ii. high level in the condensate pump room (5'0")

c. The condenser pit five foot trip shall have a trip setting of less than or equal to five feet zero inches. The five foot trip circuit for each channel shall be checked once every three months. The 3 and 1 foot alarms shall have a setting of less than or equal to three feet zero inches and less than or equal to 1 foot 0 inches. A logic system functional test, including all alarms, shall be performed during the refueling outage.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. The condenser pit water level switches shall trip the condenser circulating water pumps and alarm in the control room if water level in the condenser pit exceeds a level of 5 feet above the pit floor. If a failure occurs in one of these trip and alarm circuits, the failed circuit shall be immediately placed in a trip condition and reactor operation shall be permissible for the following seven days unless the circuit is sooner made operable.

3. If Specification 3.5.M.1 and 2 cannot be met, reactor startup shall not commence or if operating, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

3.5 LIMITING CONDITION FOR OPERATION BASES

A. Core Spray and LPCI Mode of the RHR System - This specification assures that adequate emergency cooling capability is available.

Based on the loss of coolant analyses included in References (1) and (2) in accordance with 10 CFR 50.46 and Appendix K, core cooling systems provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident, to limit the calculated peak clad temperature to less than 2200°F, to assure that core geometry remains intact, to limit the core wide clad metal-water reaction to less than 1%, and to limit the calculated local metal-water reaction to less than 17%.

The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate. The method and concept are described in Reference (3). Using the results developed in this reference, the repair period is found to be less than 1/2 the test interval. This assumes that the core spray and LPCI subsystems constitute a 1 out of 3 system, however, the combined effect of the two systems to limit excessive clad temperatures must also be considered. The test interval specified in Specification 4.5 was 3 months. Therefore, an allowable repair period which maintains the basic risk considering single failures should be less than 45 days and this specification is within this period. For multiple failures, a shorter interval is specified and to improve the assurance that the remaining

---

(1) "Loss of Coolant Accident Analyses Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations," NEDO-24146A, Revision 1, April 1979.

(2) NEDO-20566, General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50 Appendix K.

(3) APED-"Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards" - April 1969, I.M. Jacobs and P.W. Marriott.

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

systems will function, a daily test is called for. Although it is recognized that the information given in reference 3 provides a quantitative method to estimate allowable repair times, the lack of operating data to support the analytical approach prevents complete acceptance of this method at this time. Therefore, the times stated in the specific items were established with due regard to judgment.

Should one core spray subsystem become inoperable, the remaining core spray and the entire LPCI system are available should the reactor core cooling arise. To assure that the remaining core spray and LPCI subsystems are available they are demonstrated to be operable immediately. This demonstration includes a manual initiation of the pumps and associated valves. Based on judgments of the reliability of the remaining systems; i.e., the core spray and LPCI, a 7-day repair period was obtained.

Should the loss of one LPCI pump occur, a nearly full complement of core and containment cooling equipment is available. Three LPCI pumps in conjunction with the core spray subsystem will perform the core cooling function. Because of the availability of the majority of the core cooling equipment, which will be demonstrated to be operable, a 30-day repair period is justified. If the LPCI subsystem is not available, at least 2 LPCI pumps must be available to fulfill the containment cooling function. The 7-day repair period is set on this basis.

B. Containment Cooling Service Water - The containment heat removal portion of the LPCI/containment cooling subsystem is provided to remove heat energy from the containment in the event of a loss of coolant accident. For the flow specified, the containment long-term pressure is limited to less than 8 psig and, therefore, is more than ample to provide the required heat removal capability. (Ref. Section 5.2.3.2 SAR).

The containment cooling subsystem consists of two sets of 2 service water pumps, 1 heat exchanger and 2 LPCI pumps. Either set of equipment is capable of performing the containment cooling function. Loss of one containment cooling service water pump does not seriously jeopardize the containment cooling capability as any 2 of the remaining three pumps can satisfy the cooling requirements. Since there is some redundancy left a 30-day repair period is adequate. Loss of 1 containment

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

cooling subsystem leaves one remaining system to perform the containment cooling function. The operable system is demonstrated to be operable each day when the above condition occurs. Based on the facts that when one containment cooling subsystem becomes inoperable only one system remains which is tested daily. A 7-day repair period was specified.

C. High Pressure Coolant Injection - The high pressure coolant injection subsystem is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI and core spray subsystems can protect the core.

The HPCI meets this requirement without the use of off-site electrical power. For the pipe breaks for which the HPCI is intended to function the core never uncovers and is continuously cooled and thus no clad damage occurs. (Ref. Section 6.2.5.3 SAR). The repair times for the limiting conditions of operation were set considering the use of the HPCI as part of the isolation cooling system.

D. Automatic Pressure Relief - The relief valves of the automatic pressure relief subsystem are a back-up to the HPCI subsystem. They enable the core spray and LPCI to provide protection against the small pipe break in the event of HPCI failure, by depressurizing the reactor vessel rapidly enough to actuate the core sprays and LPCI. The core spray and LPCI provide sufficient flow of coolant to adequately cool the core.

Analyses have shown that only four of the five valves in the Automatic Depressurization System are required to operate. Loss of one of the relief valves does not significantly affect the pressure-relieving capability, therefore continued operation is acceptable provided the appropriate MAPLHGR reduction factor is applied to assure compliance with the 2200°F PCT limit. Loss of more than one relief valve significantly reduces the pressure relief capability of the ADS; thus; a seven day repair period is specified with the HPCI available, and a 24 hour repair period otherwise.

E. Isolation Cooling System - The turbine main condenser is normally available. The isolation condenser is provided for core decay heat removal following reactor isolation and scram. The isolation condenser has a heat removal capacity sufficient to handle the decay heat production at 300 seconds following a scram. Water will be lost from the reactor vessel through the relief valves in the 300 seconds following isolation and scram. This represents a minor loss relative to the vessel inventory.

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

The system may be manually initiated at any time. The system is automatically initiated on high reactor pressure in excess of 1060 psig sustained for 15 seconds. The time delay is provided to prevent unnecessary actuation of the system during anticipated turbine trips. Automatic initiation is provided to minimize the coolant loss following isolation from the main condenser. To be considered operable the shell side of the isolation condenser must contain at least 11,300 gallons of water. Make-up water to the shell side of the isolation condenser is provided by the condensate transfer pumps from the condensate storage tank. The condensate transfer pumps are operable from on-site power. The fire protection system is also available as make-up water. An alternate method of cooling the core upon isolation from the main condenser is by using the relief valves and HPCI subsystem in a feed and bleed manner. Therefore, the high pressure relief function and the HPCI must be available together to cope with an anticipated transient so the LCO for HPCI and relief valves is set upon this function rather than their function as depressurization means for a small pipe break.

F. Emergency Cooling Availability - The purpose of Specification D is to assure a minimum of core cooling equipment is available at all times. If, for example, one core spray were out of service and the diesel which powered the opposite core spray were out of service, only 2 LPCI pumps would be available. Likewise, if 2 LPCI pumps were out of service and 2 containment service water pumps on the opposite side were also out of service no containment cooling would be available. It is during refueling outages that major maintenance is performed and during such time that all low pressure core cooling systems may be out of service. This specification provides that should this occur, no work will be performed on the primary system which could lead to draining the vessel. This work would include work on certain control rod drive components and recirculation system. Thus, the specification precludes the events which could require core cooling. Specification 3.9 must also be consulted to determine other requirements for the diesel generators.

Dresden Units 2 and 3 share certain process systems such as the makeup demineralizers and the radwaste system and also some safety systems such as the standby gas treatment system, batteries, and diesel generators. All of these systems have been sized to perform their intended function considering the simultaneous operation of both units.

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

For the safety related shared features of each plant, the Technical Specifications for that unit contain the operability and surveillance requirements for the shared feature; thus, the level of operability for one unit is maintained independently of the status of the other. For example, the shared diesel (2/3 diesel) would be mentioned in the specifications for both Units 2 and 3 and even if Unit 3 were in the Cold Shutdown Condition and needed no diesel power, readiness of the 2/3 diesel would be required for continuing Unit 2 operation.

G. Specification 3.5.F.4 provides that should this occur, no work will be performed which could preclude adequate emergency cooling capability being available. Work is prohibited unless it is in accordance with specified procedures which limit the period that the control rod drive housing is open and assures that the worst possible loss of coolant resulting from the work will not result in uncovering the reactor core. Thus, this specification assures adequate core cooling. Specification 3.9 must be consulted to determine other requirements for the diesel generator.

Specification 3.5.F.5 provides assurance that an adequate supply of coolant water is immediately available to the low pressure core cooling systems and that the core will remain covered in the event of a loss of coolant accident while the reactor is depressurized with the head removed.

H. Maintenance of Filled Discharge Pipe - If the discharge piping of the core spray, LPCI, and HPCI are not filled, a water hammer can develop in this piping when the pump and/or pumps are started.

I. Average Planar LHGR

This specification assures that the peak cladding temperature following a postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10CFR50 Appendix K considering the postulated affects of fuel pellet densification.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average LHGR of all the rods in a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within a fuel assembly. Since expected

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than plus or minus 20°F relative to the peak temperature for a typical fuel design, the limit on the average planar LHGR is sufficient to assure that calculated temperatures are below the 10CFR50, Appendix K limit.

The maximum average planar LHGRs shown in Figure 3.5.1 are based on calculations employing the models described in References (1), (2) and (3). Power operation with APLHGRs at or below those shown in Figure 3.5.1 assures that the peak cladding temperature following a postulated loss-of-coolant accident will not exceed the 2200°F limit.

ANF has analyzed the effects Single Loop Operation has on LOCA events (Reference 4). For breaks in the idle loop, the above Dual Loop Operation results are conservative (Reference 1). For breaks in the active loop, the event is more severe primarily due to a more rapid loss of core flow. By applying a multiplicative 0.91 reduction factor to the results of the previous analyses, all applicable criteria are met.

(1) "Loss of Coolant Accident Analyses Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations," NEDO-24146A, Revision 1, April 1979.

(2) XN-NF-81-75 "Dresden Unit 3 LOCA Model Using the ENC EXEM Evaluation Model MAPLHGR Results"

(3) XN-NF-85-63 "Dresden Unit 3 LOCA-ECCS Analysis MAPLHGR results for 9x9 fuel", dated Septebmer 1985.

(4) ANF-84-111, "LOCA-ECCS Analysis for Dresden Units During Single Loop Operation with ANF Fuel," September 1987.

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

J. Local Steady State LHGR

This specification assures that the maximum linear heat generation rate in any fuel rod is less than the design linear heat generation rate even if fuel pellet densification is postulated. This provides assurance that the fuel end-of-life steady state criteria are met.

K. Local Transient LHGR

This specification provides assurance that the fuel will neither experience centerline melt nor exceed 1% plastic cladding strain for transient overpower events beginning at any power and terminating at 120% of rated thermal power.

L. Minimum Critical Power Ratio (MCPR)

The steady-state values for MCPR specified in the Specification were determined using the THERMEX thermal limits methodology described in XN-NF-80-19, Volume 3. The safety limit implicit in the Operating limits is established so that during sustained operation at the MCPR safety limit, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition. The Limiting Transient delta CPR implicit in the operating limits was calculated such that the occurrence of the limiting transient from the operating limit will not result in violation of the MCPR safety limit in at least 95% of the random statistical combinations of uncertainties.

Transient events of each type anticipated during operation of a BWR/3 were evaluated to determine which is most restrictive in terms of thermal margin requirements. The generator load rejection/turbine trip without bypass is typically the limiting event. The thermal margin effects of the event are evaluated with the THERMEX Methodology and appropriate MCPR limits consistent with the XN-3 critical power correlation are determined. Several factors influence which transient results in the largest reduction in critical power ratio, such as the cycle-specific fuel loading, exposure and fuel type. The current cycle's reload licensing analyses identifies the limiting transient for that cycle.

As described in Specification 4.3.C.3 and the associated Bases, observed plant data were used to determine the average scram performance used in the transient analyses for determining the MCPR Operating Limit. If the current cycle scram time performance falls outside of the distribution assumed in the analyses, an adjustment of the MCPR limit may be required to

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

maintain margin to the MCPR Safety Limit during transients. Compliance with the assumed distribution and adjustment of the MCPR Operating Limit will be performed as directed by the nuclear fuel vendor in accordance with station procedures.

For core flows less than rated, the MCPR Operating Limit established in the specification is adjusted to provide protection of the MCPR Safety Limit in the event of an uncontrolled recirculation flow increase to the physical limit of pump flow. This protection is provided for manual and automatic flow control by choosing the MCPR operating limit as the value from Figure 3.5-2 Sheet 1 or the rated core flow value, whichever is greater. For Automatic Flow Control, in addition to protecting the MCPR Safety Limit during the flow run-up event, protection is provided against violating the rated flow MCPR Operating Limit during an automatic flow increase to rated core flow. This protection is provided by the reduced flow MCPR limits shown in Figure 3.5-2 Sheet 2 or 3 where the curve corresponding to the current rated flow MCPR limit is used (linear interpolation between the MCPR limit lines depicted is permissible). Therefore, for Automatic Flow Control, the MCPR Operating Limit is chosen as the value from Figure 3.5-2 Sheet 1, Sheet 2, Sheet 3 or the rated flow value, whichever is greatest.

Analyses have demonstrated that transient events in Single Loop Operation are bounded by those at rated conditions; however, due to the increase in the MCPR fuel cladding integrity safety limit in Single Loop Operation, an equivalent adder must be uniformly applied to all MCPR LCO to maintain the same margins to the MCPR fuel cladding integrity safety limit.

M. Flood Protection

Condensate pump room flood protection will assure the availability of the containment cooling service water system (CCSW) during a postulated incident of flooding in the turbine building. The redundant level switches in the condenser pit will preclude any postulated flooding of the turbine building to an elevation above river water level. The level switches provide alarm and circulating water pump trip in the event a water level is detected in the condenser pit.

4.5 SURVEILLANCE REQUIREMENT BASES

(A thru F)

The testing interval for the core and containment cooling systems is based on quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example the core spray final admission valves do not open until reactor pressure has fallen to 350 psig thus during operation even if high drywell pressure were stimulated the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

The systems can be automatically actuated during a refueling outage and this will be done. To increase the availability of the individual components of the core and containment cooling systems the components which make up the system i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month. Likewise the pumps and motor-operated valves are also tested each month to assure their operability. The combination of a yearly simulated automatic actuation test and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems.

With components or subsystems out-of-service overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining cooling equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, etc., caused the out-of-service period, then the demonstration of operability should be thorough enough to assure that a similar problem does not exist on the remaining components. For example, if an out-of-service period were caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

The requirement of 180 psig at 3500 gpm at the containment cooling service water (CCSW) pump discharge provides adequate margin to ensure that the LPCI/CCSW system provides the design

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

bases cooling water flow and maintains 20 psig differential pressure at the containment cooling heat exchanger. This differential pressure precludes reactor coolant from entering the river water side of the containment cooling heat exchangers.

The verification of Main Steam Relief Valve operability during manual actuation surveillance testing must be made independent of temperatures indicated by thermocouples downstream of the relief valves. It has been found that a temperature increase may result with the valve still closed. This is due to steam being vented through the valve actuation mechanism during the surveillance test. By first opening a turbine bypass valve, and then observing its closure response during relief valve actuation, positive verification can be made for the relief valve opening and passing steam flow. Closure response of the turbine control valves during relief valve manual actuation would likewise serve as an adequate verification for relief valve opening. This test method may be performed over a wide range of reactor pressure greater than 150 psig. Valve operation below 150 psig is limited by the spring tension exhibited by the relief valves.

G. Deleted

H. Maintenance of Filled Discharge Pipe

The surveillance requirements to assure that the discharge piping of the core spray, LPCI, and HPCI systems are filled provides for a visual observation that water flows from a high point vent. This ensures that the line is in a full condition. Between the monthly intervals at which the lines are vented, instrumentation has been provided to monitor the presence of water in the discharge piping. This instrumentation will be calibrated on the same frequency as the safety system instrumentation. This period of periodic testing ensures that during the intervals between the monthly checks the status of the discharge piping is monitored on a continuous basis.

I. Average Planar LHGR

At core thermal power levels less than or equal to 25 percent, operating plant experience and thermal hydraulic analyses indicate that the resulting average planar LHGR is below the maximum average planar LHGR by a considerable margin; therefore,

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

evaluation of the average planar LHGR below this power level is not necessary. The daily requirement for calculating average planar LHGR above 25 percent rated thermal power is sufficient since power distribution shifts are slow when there have not been significant power or control rod changes.

J. Local Steady State LHGR

The LHGR for all fuel shall be checked daily during reactor operation at greater than or equal to 25 percent power to determine if fuel burnup or control rod movement has caused changes in power distribution. A limiting LHGR value is precluded by a considerable margin when employing a permissible control rod pattern below 25% rated thermal power.

K. Local Transient LHGR

The fuel design limiting ratio for centerline melt (FDLRC) shall be checked daily during reactor operation at greater than or equal to 25% power to determine if fuel burnup or control rod movement has caused changes in power distribution. The FDLRC limit is designed to protect against centerline melting of the fuel during anticipated operational occurrences.

L. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 percent, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicates that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR.

The daily requirement for calculating MCPR above 25 percent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes.

M. Flood Protection

The watertight bulkhead door and the penetration seals for pipes and cables penetrating the vault walls have been

4.5 <u>SURVEILLANCE REQUIREMENT BASES</u> (Cont'd.)

designed to withstand the maximum flood conditions. To assure that their installation is adequate for maximum flood conditions, a method of testing each seal has been devised.

To test a pipe seal, another test seal is installed in the opposite side of the penetration creating a space between the two seals that can be pressurized. Compressed air is then supplied to a fitting on the test seal and the space inside the sleeve is pressurized to approximately 15 psi. The outer face of the permanent seal is then tested for leaks using a soap bubble solution.

On completion of the test, the test seal is removed for use on other pipes and penetrations of the same size.

In order to test the watertight bulkhead doors, a test frame must be installed around each door. At the time of the test, a reinforced steel box with rubber gasketing is clamped to the wall around the door. The fixture is then pressurized to approximately 15 psig to test for leak tightness.

Floor drainage of each vault is accomplished through a carbon steel pipe which penetrates the vault. When open, this pipe will drain the vault floor to a floor drain sump in the condensate pump room.

Equipment drainage from the vault coolers and the CCSW pump bedplates will also be routed to the vault floor drains. The old equipment drain pipes will be permanently capped to preclude the possibility of back-flooding the vault.

As a means of preventing backflow from outside the vaults in the event of a flood, a check valve and an air operated valve are installed in the 2" vault floor drain line 6'0" above the floor of the condensate pump room.

The check valve is a 2" swing check designed for 125 psig service. The air operated valve is a control valve designed for a 50 psi differential pressure. The control valve will be in the normally open position in the energized condition and will close upon any one of the following:

a. Loss of air or power

b. High level (5'0") in the condensate pump room

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

Closure of the air operated valve on high water level in the condensate pump room is effected by use of a level switch set at a water level of 5'0". Upon actuation, the switch will close the control valve and alarm in the control room.

The operator will also be aware of problems in the vaults/ condensate pump room if the high level alarm on the equipment drain sump is not terminated in a reasonable amount of time. It must be pointed out that these alarms provide information to the operator but that operator action upon the above alarms is not a necessity for reactor safety since the other provisions provide adequate protection.

A system of level switches has been installed in the condenser pit to indicate and control flooding of the condenser area. The following switches are installed:

	Level	Function
a.	1'0" (1 switch)	Alarm, Panel Hi-Water-Condenser Pit
b.	3'0" (1 switch)	Alarm, Panel High-Circ. Water Condenser Pit
c.	5'0" (2 redundant switch pairs)	Alarm and Circ. Water Pump Trip

Level (a) indicates water in the condenser pit from either the hotwell or the circulating water system. Level (b) is above the hotwell capacity and indicates a probable circulating water failure.

Should the switches at level (a) and (b) fail or the operator fail to trip the circulating water pumps on alarm at level (b), the actuation of either level switch pair at level (c) shall trip the circulating water pumps automatically and alarm in the control room. These redundant level switch pairs at level (c) are designed and installed to IEEE-279, "Criteria for Nuclear Power Plant Protection Systems." As the circulating water pumps are tripped, either manually or automatically, at level (c) of 5'0", the maximum water level reached in the condenser pit due to pumping will be at the 491'0" elevation (10' above condenser pit floor elevation 481'0"; 5' plus an additional 5' attributed to pump coastdown).

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

In order to prevent overheating of the CCSW pump motors, a vault cooler is supplied for each pump. Each vault cooler is designed to maintain the vault at a maximum 105°F temperature during operation of its respective pump. For example, if CCSW pump 2B-1501 starts, its cooler will also start and compensate for the heat supplied to the vault by the 2B pump motor keeping the vault at less than 105°F.

Each of the coolers is supplied with cooling water from its respective pump's discharge line. After the water has been passed through the cooler, it returns to its respective pump's suction line. In this way, the vault coolers are supplied with cooling water totally inside the vault. The cooling water quantity needed for each cooler is approximately 1% to 5% of the design flow of the pumps so that the recirculation of this small amount of heated water will not affect pump or cooler operation.

Operation of the fans and coolers is required during pump operability testing and thus additional surveillance is not required.

Verification that access doors to each vault are closed, following entrance by personnel, is covered by station operating procedures.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

e. The suction valve in the idle loop shall be closed and electrically isolated except when the idle loop is being prepared for return to service; and

f. If the tripped pump is out of service for more than 24 hours, implement the following additional restrictions:

i. The flow biased RBM Rod Block LSSS shall be reduced by 4.0% (Specification 3.2.C.1);

ii. The flow biased APRM Rod Block LSSS shall be reduced by 3.5% (Specification 2.1.B);

iii. The flow biased APRM scram LSSS shall be reduced by 3.5% (Specification 2.1.A.1);

iv. The MCPR Safety Limit shall be increased by 0.01 (Specification 1.1.A);

v. The MCPR Operating Limit shall be increased by 0.01 (Specification 3.5.L.3);

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

vi. The MAPLHGR Operating Limit shall be reduced by a multiplicative factor of 0.91 (Specification 3.5.I). If concurrently, one Automatic Pressure Relief Subsystem relief valve is out-of-service, the MAPLHGR Operating Limit shall be reduced by a multiplicative factor of 0.89 for 8x8 fuel and 0.76 for 9x9 fuel.

4. Core thermal power shall not exceed 25% of rated without forced recirculation. If core thermal power is greater than 25% of rated without forced recirculation, action shall be initiated within 15 minutes to restore operation to within the prescribed limits and core thermal power shall be returned to within the prescribed limit within two (2) hours.

I. Snubbers (Shock Suppressors)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

I. Snubbers (Shock Suppressors)

The following surveillance requirements apply to safety related snubbers.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

In addition, during the start-up of Dresden Unit 2, it was found that a flow mismatch between the two sets of jet pumps caused by a difference in recirculation loops could set up a vibration until a mismatch in speed of 27% occurred. The 10% and 15% speed mismatch restrictions provide additional margin before a pump vibration problem will occur.

Reduced flow MCPR Operating Limits for Automatic Flow Control are not applicable for Single Loop Operation. Therefore, sustained reactor operation under such conditions is not permitted.

Regions I and II of Figure 3.6.2 represent the areas of the power/flow map with the least margin to stable operation. Although calculated decay ratios at the intersection of the natural circulation flow line and the APRM Rod block line indicate that substantial margin exists to where unstable operation could be expected. Specifications 3.6.H.3.b., 3.6.H.3.c. and 4.6.H.3. provide additional assurance that if unstable operation should occur, it will be detected and corrected in a timely manner.

During the starting sequence of the inoperable recirculation pump, restricting the operable recirculation pump speed below 65% of rated prevents possible damage to the jet pump riser braces due to excessive vibration.

The closure of the suction valve in the idle loop prevents the loss of LPCI through the idle recirculation pump into the downcomer.

Analyses have been performed which support indefinite operation in single loop provided the restrictions discussed in Specification 3.6.H.3.d. are implemented within 24 hours.

The LSSSs are corrected to account for backflow through the idle jet pumps above 20-40% of rated recirculation pump speed. This assures that the original drive flow biased rod block and scram trip settings are preserved during Single Loop Operation.

The MCPR safety limit has been increased by 0.01 to account for core flow and TIP reading uncertainties which are used in the statistical analysis of the safety limit. In addition, the MCPR Operating Limit has also been increased by 0.01 to maintain the same margin to the safety limit as during Dual Loop Operation.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

The mulplicative 0.91 reduction of MAPLHGR Operating Limit accounts for the more rapid loss of core flow during Single Loop Operation than during Dual Loop Operation.

The more conservative MAPLHGR reduction factors of 0.89 for 8x8 fuel and 0.76 for 9x9 fuel are applied if one relief valve and one recirculation loop are inoperable at the same time. The small break LOCA is the concern for one relief valve out-of-service; the large break LOCA is the concern for Single Loop Operation. Selecting the more conservative MAPLHGR multipliers will cover both the relief value out-of-service and Single Loop Operation.

Specification 3.6.H.4 increased the margin of safety for thermal-hydraulic stability and for startup of recirculation pumps from natural circulation conditions.

I. Snubbers (Shock Suppressors)

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

Because the snubber protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacements. In case a shutdown is required, the allowance of 36 hours to reach a cold shutdown condition will permit an orderly shutdown consistent with standard operating procedures. Since plant startup should not commence with knowingly defective safety related equipment, Specification 3.6.I.4 prohibits startup with inoperable snubbers.

When a snubber is found inoperable, a review shall be performed to determine the snubber mode of failure. Results of the review shall be used to determine if an engineering evaluation of the safety-related system or component is necessary. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the support component or system.

All safety related hydraulic snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate hydraulic fluid level and proper attachment of snubber to piping and structures.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

All safety related mechanical snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation and attachments to the piping and anchor for indication of damage or impaired operability.

The inspection frequency is based upon maintaining a constant level of snubber protection. Thus, the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

To further increase the assurance of snubber reliability, functional tests will be performed once each refueling cycle. A representative sample of 10% of the safety-related snubbers will be functionally tested. Observed failures on these samples will require testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as different entities for the above surveillance programs.

Hydraulic snubber testing will include stroking of the snubbers to verify piston movement, lock-up, and bleed. Functional testing of the mechanical snubbers will consist of verification that the force that initiates free movement of the snubber in either tension or compression is less than the maximum breakaway friction force and verification that the activation (restraining action) is achieved within the specified range of acceleration or velocity, as applicable based on snubber design, in both tension and compression.

3.10 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

G. Fuel Storage Reactivity Limit

1. The new fuel storage facility shall be such that the K_{eff} dry is less than 0.90 and flooded is less than 0.95.

2. Whenever a fuel assembly is stored in the spent fuel storage pool, the peak assembly reactivity in a reactor lattice distribution shall be limited to less than or equal to the following values:

Assembly Type	K_{inf}
GE 7x7	1.26
GE 8x8	1.32
ANF 8x8	1.33
ANF 9x9	1.27

Whenever storing other assembly types or fuel rods in the spent fuel storage pool, their peak reactivity shall be bounded by the most limiting K_{inf} value listed above.

H. Loads Over Spent Fuel Storage Pool

No loads heavier than the weight of a single spent fuel assembly and handling tool shall be carried over fuel stored in the spent fuel storage pool.

4.10 SURVEILLANCE REQUIREMENTS (Cont'd.)

G. Fuel Storage Reactivity Limit

1. Prior to storing Fuel in the new fuel storage facility, an analysis must be performed to demonstrate that the criteria in 3.10.G.1 are satisfied.

2. Prior to storing Fuel in the spent fuel storage pool, an analysis must be performed to demonstrate that the criteria in 3.10.G.2 are satisfied.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 94 TO FACILITY OPERATING LICENSE NO. DPR-25

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNIT NO. 3

DOCKET NO. 50-249

1.0 INTRODUCTION

By letter dated March 9, 1988 (Ref. 1), Commonwealth Edison Company (CECo) proposed to amend Appendix A and Section 3.E of Facility Operating License No. DPR-25 to support Cycle 11 operation of Dresden Unit 3 with an entire core of Advanced Nuclear Fuels (ANF) fuel. In a letter dated June 17, 1988, CECo submitted two Technical Specifications pages that were inadvertently deleted from the original submittal. These pages were related to reactor operation with relief valves out of service. The March 9 submittal addressed all aspects of the changes involved related to the relief valves except for the inadvertently deleted pages. All residual General Electric (GE) fuel is scheduled to be discharged during the end-of-cycle (EOC) 10 outage. The requested amendment furnished information to support (1) use of ANF 9x9 fuel with axially zoned burnable absorber (Gd_2O_3) rods, (2) modified limits for single loop operation (SLO) based on ANF analyses, (3) provisions for extended operation with a relief valve out-of-service, and (4) operation, including coastdown, with reduced feedwater heating.

In support of the Dresden 3 Cycle 11 (D3C11) reload CECo submitted topical reports which described the reload analysis (Ref. 2), the plant transient analysis (Ref. 3), analysis of operation with one relief valve out-of-service (Ref. 4), and the LOCA-ECCS analysis during SLO with ANF fuel (Ref. 5).

2.0 EVALUATION OF RELOAD

2.1 Reload Description

The D3C11 reload will include 72 fresh ANF 9x9 fuel assemblies designated XN-4H and 96 fresh ANF 9x9 fuel assemblies designated XN-4L. These assemblies have a central region enrichment of 3.35 weight percent U-235 and 6 inch natural uranium ends to yield an average assembly enrichment of 3.13 weight percent U-235. The remainder of the core is comprised of 556 previously irradiated ANF fuel assemblies designated XN-1 (8x8), XN-2 (8x8), XN-3 (9x9), and XN-3A (9x9). The core will be operated under the Single Rod Sequence (SRS) control strategy to assure that the control rod withdrawal error will not be limiting.

2.2 Fuel Design

The mechanical design of the XN-4H and XN-4L 9x9 reload fuel is described in References 6 and 7. The ANF fuel to be returned to the Dresden 3 core has been approved for operation in previous cycles. The XN-4H and XN-4L fuel assemblies are identical except for a difference in the Gd_2O_3 concentration in the central region of the gadolinia-bearing fuel rods. Both assembly types contain nine gadolinia-bearing fuel rods with 3.0% Gd_2O_3 in the top six inches and bottom 12 inches of the enriched region of these rods. The central region of the gadolinia-bearing rods in the XN-4H assemblies contains 4.5% Gd_2O_3 whereas the XN-4L assemblies contain 4.0% Gd_2O_3 in this region. Both fuel types contain 79 fuel rods (8 are tie rods) and two water rods. Based on the previous review of the generic submittal (Ref. 6) and the information submitted with the D3C11 reload application, the staff finds the mechanical design of the ANF fuel for the D3C11 reload is acceptable.

During the review of the Cycle 10 reload submittal, the staff placed an exposure cap on 8x8 and 9x9 fuel due to rod bow considerations. The limit was set at 30,000 MWD/MTU for 8x8 fuel and 23,000 MWD/MTU for 9x9 fuel (batch average exposure). The expected peak assembly exposures at the end of Cycle 11 are 34,400 MWD/MTU and 23,500 MWD/MTU for the 8x8 and 9x9 assemblies, respectively. Based on additional information on rod bowing (Ref. 8) which has been reviewed by the staff and on the staff's safety evaluation of XN-NF-82-06(P), Supplement 1, Revision 2 (Refs. 9 & 10), these expected Cycle 11 fuel exposures are acceptable.

2.3 Thermal-Hydraulic Design

Single phase flow tests of full scale assemblies have been performed in order to determine the component hydraulic resistances for the D3C11 fuel types. Based on the similar hydraulic performance illustrated in hydraulic demand curves for ANF 8x8 and ANF 9x9 fuel (Ref. 2), the staff concludes that the two fuel types are hydraulically compatible.

The XN-3 correlation used to develop the minimum critical power ratio (MCPR) safety limit has been approved for application to both the ANF 8x8 and the new 9x9 fuel type (Refs. 11 & 12). ANF has calculated the MCPR safety limit to be 1.05 for all fuel types in the D3C11 core. Since the calculations considered each of the constituent fuel types, conservative local power distributions for each type, the worst (bounding) radial power distribution at which each fuel type is expected to operate and used approved methodology (Ref. 13), the staff finds the safety limit acceptable for all Cycle 11 fuel types. The proposed operating limit MCPR for D3C11 is 1.39, which is the same value as for current (Cycle 10) operation, and bounds the delta-CPR results of the limiting plant transients as discussed in Section 2.5 of this safety evaluation.

The thermal-hydraulic stability of the Cycle 11 core was analyzed using the methods identified in XN-NF-80-19(P)(A), Volume 4, Revision 1 (Ref. 14). Reference 14 cites the use of the COTRAN and COTRANSA models for use in the analysis of core thermal-hydraulic stability. The resultant maximum decay ratios for natural recirculation flow determined analytically using the approved COTRAN code at various power and flow conditions are 0.35 (47.6% rated power and 31.5% rated flow) at the 100% flow control line (FCL) and 0.55 (58% rated power) at the average power range monitor (APRM) rod block intercept. Since both of these decay ratios are less than the surveillance criterion of 0.75 as calculated by COTRAN, no stability Technical Specification surveillance requirement is needed for Cycle 11 operation. A test comparing stability between dual loop operation (DLO) and single loop operation (SLO) was performed during Cycle 10 from which it was concluded that the operating region of concern exhibits adequate margin to power/flow instabilities in SLO and DLO. Although the staff concurs that present positions indicate stability monitoring Technical Specifications are not required during DLO, staff positions regarding calculated acceptable decay ratios and Technical Specifications requirements described in NRC Generic Letter 86-02 are under review due to the recent LaSalle instability event. Any new staff findings as a result of this review will likely be applied generically to all BWRs including Dresden 3. In addition, since ANF 9X9 fuel has not yet received generic approval by the staff, there is the possibility of additional stability testing as the amount of 9X9 fuel in the Dresden 3 core increases in future reloads. For SLO, the current stability surveillances required by Technical Specifications are adequate for detecting any core wide or local instabilities. Therefore, the thermal-hydraulic design of Cycle 11 is acceptable.

2.4 Nuclear Design

The nuclear design for D3C11 has been performed with ANF methodologies previously reviewed and approved (Ref. 15). The fuel loading pattern is given in Figure 4.4 of Reference 2. The beginning-of-cycle (BOC) shutdown margin is 1.24% delta-k and at minimum conditions is 1.11% delta-k, well in excess of the required 0.42% delta-k. The standby liquid control system (which is designed to inject a quantity of boron solution that produces a concentration of no less than 600 ppm of boron in the reactor core) was calculated to provide a shutdown margin of 6.19% delta-k for cold conditions with all control rods in their full power positions. This meets the shutdown margin requirement of 3.0% delta-k and is, therefore, acceptable. Since these results have been obtained by previously approved methods and meet the appropriate requirements, the staff concludes that the nuclear design of Cycle 11 is acceptable.

For D3C11, there will be 12 ASEA-ATOM (A-A) control blades designated CR-82B inserted which are similar to the CR-82 design previously generically reviewed and approved by the NRC (Ref. 16). These new control blades incorporate several enhancements compared to the CR-82 design and do not have any significant impact on the mechanical characteristics. The staff, therefore, finds them acceptable for use in Cycle 11.

2.5 Transient and Accident Analyses

Corewide transients were analyzed with the same methodology used to establish thermal margin requirements for Cycle 10 operation (Refs. 17 & 18). The XCOBRA-T hot channel model was used to calculate the delta-CPR values. The XCOBRA-T model has been reviewed by the staff and found to be acceptable (Ref. 19).

The licensee evaluated several categories of potential corewide transients for Cycle 11 and provided specific results for three transients, generator load rejection without bypass (LRWB), feedwater controller failure (FWCF), and loss of feedwater heating (LFWH). The limiting transient is identified as the LRWB. Since this limiting transient is a rapid pressurization event, the ANF methodology for including uncertainties in determining operating limits for rapid pressurization transients in BWRs (Refs. 20 & 21) was applied. This methodology uses a conservative deterministic multiplier of 110% on the calculated transient power to account for COTRANSA code uncertainties and treats the uncertainties in the important input variables (scram speed and scram delay) statistically. At rated power, the delta-CPR was 0.23 for ANF 8x8 fuel and 0.26 for ANF 9x9 fuel for the LRWB transient. Therefore, the delta-CPR results of the analyses for the limiting corewide transients are acceptably bounded by the proposed Cycle 11 MCPR limiting condition of operation (LCO) of 1.39.

The most limiting event for reactor vessel over-pressurization is the main steamline isolation valve (MSIV) closure without direct scram (single failure) on valve position. The maximum value of the sensed pressure in the steam dome was 1297 psig which corresponds to a maximum vessel pressure of 1324 psig at the lower plenum. These values are less than the Technical Specification limit of 1345 psig as measured by the steam dome pressure indicator and the 1375 psig ASME vessel pressure limit. This is acceptable.

The licensee has also evaluated the effect of a relief valve out-of-service (RVOOS) on the plant transients (Ref. 4). The results indicate that with one RVOOS there is no effect on delta-CPR calculated for the limiting transients and an insignificant effect on peak pressure for all fuel types in Cycle 11.

The licensee has determined the required reduced flow MCPR operating limit for off-rated conditions to complement the Cycle 11 MCPR full flow operating limits during the automatic flow control (AFC) condition and in manual flow control (MFC). The results are given in Tables 5.4, 5.5, and 5.7 of Reference 3 and are acceptable.

For the control rod withdrawal error (RWE) local transient, the licensee has determined that a rod block monitor (RBM) upper setting of 110% of full power results in a delta-CPR of 0.31 for 9x9 fuel and 0.30 for 8x8 fuel. The Technical Specification MCPR LCO of 1.39 for both fuel types in Cycle 11, therefore, bounds the RWE results.

Analyses with a feedwater heater out-of-service (FHOOS) were also performed to support coastdown operation for EOC 11. The results show that the delta-CPRs for the transients analyzed with a FHOOS are bounded by the delta-CPRs for transients at normal feedwater temperature (Ref. 3).

The licensee also evaluated the control rod drop accident (RDA) and the loss of coolant accident (LOCA) which are described as follows.

The RDA evaluation yields a value of 187 cal/gm for the maximum deposited fuel rod enthalpy. This is well below the NRC required limit of 280 cal/gm, and is, therefore, acceptable.

ANF has previously performed LOCA analyses for Dresden 3 using 8x8 fuel (Ref. 22) and 9x9 fuel (Ref. 23) which provided maximum average planar linear heat generation rate (MAPLHGR) limits. These limits remain applicable for the fuel in Cycle 11 during dual loop operation. ANF has also evaluated the effect of a RVOOS on the MAPLHGR limits (Ref. 4). The limiting postulated small break LOCA was analyzed since relief valves do not actuate in large breaks. Based on the results of this latter analysis, MAPLHGR multipliers of 0.89 and 0.76 were calculated for 8x8 and 9x9 fuel types, respectively. The results of a LOCA analysis for Dresden 3 during SLO (Ref. 5), which were performed by ANF using the generically approved EXEM/BWR Evaluation Model, established the multiplier to be applied to the MAPLHGRs of the ANF fuel during SLO. These results support a MAPLHGR multiplier of 0.91 for all fuel types in the Cycle 11 core during SLO. The LOCA analyses were performed with reviewed and accepted methods and the results are well within the limits of 10 CFR 50.46. Therefore, the staff concludes that the MAPLHGR limits proposed for Cycle 11 are acceptable.

2.6 Extended Load Line Limit Analysis (ELLLA)

The extended load line limit analysis (ELLLA) provides a basis to support plant normal operation in the region of the power/flow map above the 100% power/100% flow load line and bounded by the 108% APRM rod block line and the 100% rated power line. This added capability increases operating flexibility to permit flow compensation for xenon buildup following startups and for fuel depletion later in cycle, and to improve the efficiency of achieving and maintaining 100% power. The results of the previous ELLLA performed by ANF as part of the Cycle 10 reload analyses are also applicable to Cycle 11 since the cycle specific analyses for Cycle 11 have been performed consistently with respect to power/flow region assumptions. It is concluded that changes in core behavior caused by the extended operating range have been acceptably accounted for in D3C11.

2.7 Single Loop Operation

Current Technical Specifications for Dresden 3 permit plant operation with a single recirculation loop out-of-service for an extended period of time. GE analyses have demonstrated that transient events during single loop operation (SLO) are bounded by those at rated conditions. ANF analyses have confirmed the GE conclusions. Since the ANF fuel was designed to be compatible with the previous co-resident GE fuel in thermal-hydraulic, nuclear and mechanical design performance, and since the ANF methodology has given results which are consistent with those of GE for normal two-loop operation, the staff concludes that the GE analyses for SLO are also applicable to SLO with fuel and analyses provided by ANF.

For SLO, GE found that an increase of 0.01 in the MCPR safety limit was needed to account for increased flow measurement uncertainties and increased traveling incore probe (TIP) uncertainties associated with single pump operation. ANF has also evaluated these effects and found that the 0.01 increase in the allowed safety limit MCPR is applicable to ANF fuel during SLO. Therefore, the staff concludes that increasing the safety limit MCPR by 0.01 for SLO with ANF fuel during Cycle 11 is acceptable.

ANF has also performed LOCA analyses for SLO conditions, as discussed in Section 2.5, to determine an appropriate SLO MAPLHGR multiplier for ANF 8x8 and 9x9 fuels.

2.8 Technical Specification Changes

To support D3C11 operation with a mixed core of ANF 8x8 and ANF 9x9 fuel consistent with the safety analyses, the following Technical Specification changes have been requested:

(1) Specification 3.5.K: A new Section for transient LHGR limits is added. This provides assurance that the fuel will neither experience centerline melt nor exceed 1% plastic cladding strain for transient overpower events

beginning at any power and terminating at 120% of rated thermal power and is, therefore, acceptable.

(2) Specification 1.1A, 3.5.L.3, 3.6.H.3.f.iv, 3.6.H.3.f.v: The MCPR LCO adder during SLO is changed to 0.01 from 0.03. This results in an increase in the MCPR safety limit for SLO of 0.01 (relative to two loop operation). As discussed in Section 2.7, the increase of safety limit MCPR by 0.01 for SLO is to account for the increased uncertainties in the total core flow and TIP reading and is acceptable.

(3) Specification 3.5.L: The MCPR LCO is changed to 1.39 for both 8x8 and 9x9 fuel. This has been shown to bound the limiting transients and accidents in Cycle 11 and is, therefore, acceptable.

(4) Specification 3.5.L.1 and Figure 3.5-2: The Figure (Sheets 1, 2, and 3) is revised to incorporate changes in reduced flow MCPR values. This has been evaluated in Section 2.5 and found to be acceptable.

(5) Specification 1.0: The definitions of the Fraction of Limiting Power Density (FLPD) and the Maximum Fraction of Limiting Power Density (MFLPD) are deleted and replaced by the definitions of the Steady State Linear Heat Generation Rate (SLHGR), the Fuel Design Limiting Ratio for Exxon Fuel (FDLRX), the Transient Linear Heat Generation Rate (TLHGR) and the Fuel Design Limiting Ratio for Centerline Melt (FDLRC). These changes are administrative in nature and delete information no longer applicable or provide clarification to current specifications.

(6) Specification 2.1.A.1, 2.1.B, 3.1.A.2, 4.1.A.2.a, and Table 3.2.3: References to MFLPD, MFLPD/FRP, and FRP/MFLPD are changed to the indicated FDLRC or 1/FDLRC. These are also administrative changes and are acceptable.

(7) Specification 3.5.I: Figure 3.5-1 (Sheets 3, 4, and 5) are deleted. These administrative changes are acceptable.

(8) Specification 3.5.J, 4.5.J: The Section 3.5.J title is changed to "LOCAL STEADY STATE LHGR," references to FDLRX are added, GE LHGR design value of 13.4 KW/ft is deleted from Figure 3.5-1A, and "STEADY STATE" is added to title. These administrative changes are acceptable.

(9) Specification 3.K.5: A new Section on local transient LHGR and FDLRC is added as well as Figure 3.5-1B showing the Transient LHGR Limit curve. These administrative changes are acceptable.

(10) Specification 3.5.I, 3.6.H.3.f.vi: The SLO MAPLHGR multiplier is changed to 0.91 from 0.70. This has been justified by the results of the LOCA analyses for SLO discussed in Section 2.5 and is acceptable.

(11) Specification 3.5.L: MCPR Penalty based on scram time performance is deleted. This is acceptable since the MCPR LCO of 1.39 conservatively bounds the delta-CPR results of the plant transient analyses for Cycle 11.

(12) Specification 3.5.D, 4.5.D, 3.5.I, 3.6.H.3.f.vi: Wording is changed to allow extended operation with one RVOOS and limited (7 days) operation with two RVOOS provided HPCI is operable and MAPLHGR adjustment factors are applied. Operation is allowed with one RVOOS provided appropriate MAPLHGR reductions discussed in Section 2.5 are implemented. Analyses have shown that allowing two RVOOS (and assuming the HPCI is inoperable) may cause the 2200° F PCT limit to be exceeded. To allow a longer repair time, HPCI operability must be credited. Since HPCI must be tested upon finding two RVs inoperable, this change allows the same 7 day period recently approved for Quad Cities Unit 1 Cycle 10, provided HPCI is shown to be operable.

(13) Throughout the Technical Specifications and Bases, references to Exxon Nuclear Company (ENC) have been changed to Advanced Nuclear Fuels Corporation (ANF). In addition, various sections have been revised to

reflect the appropriate ANF methodologies and to delete GE methods and references where appropriate. These are acceptable administrative changes.

2.9 License Change

The following license restriction has been supported for Cycle 11 operation:
"Section 3.E Restriction

Operation in the coastdown mode is permitted to 40% power."

This restriction drops the portion of 3.E regarding off-normal feedwater heating which required a determination if the MCPR Operating Limit and calculated peak pressure for the worst case abnormal operating transient remain bounding. This has been evaluated in Section 2.5, Transient and Accident Analysis, of this Safety Evaluation and found to be acceptable. Thus modifying Section 3.E of the license to delete the requirement to prepare a safety evaluation for coastdown operation with off-normal feedwater temperature is acceptable.

3.0 SUMMARY

Based on the review of the fuel, nuclear, and thermal-hydraulic design as well as the transient and accident analysis presented by the licensee, the staff concludes that the proposed reload of Dresden 3 Cycle 11 and associated Technical Specification changes are acceptable.

4.0 REFERENCES

1. Letter from J. A. Silady (CECo) to T. E. Murley (NRC), Dresden Nuclear Power Station Unit 3 Proposed License Amendment and Analysis for Cycle 11 Reload, March 9, 1988.

2. ANF-87-097, "Dresden Unit 3 Cycle 11 Reload Analysis," September 1987.

3. ANF-87-096, "Dresden Unit 3 Cycle 11 Plant Transient Analysis," September 1987.

4. XN-NF-84-49, "Analysis of Dresden Units 2 and 3 Operation With One Relief Valve Out-of-Service," September 1984.

5. ANF-87-111, "LOCA-ECCS Analysis for Dresden Units During Single Loop Operation with ANF Fuel," September 1987.

6. XN-NF-85-067(P)(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," September 1986.

7. ANF-87-110(P), "Dresden Unit 3 Cycle 11 Nuclear Design Report," July 1987.

8. Letter from G. N. Ward (ANF) to G. C. Lainas (NRC), "Additional Information on Rod Bowing," GNW:021:87, March 11, 1987.

9. XN-NF-82-06(P), Supplement 1, Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup, Supplement 1, Extended Burnup Qualification of ENC 9x9 BWR Fuel," January 1987.

10. Letter from A. Thadani (NRC) to D. A. Adkisson (ANF), "Acceptance for Referencing of Licensing Topical Report XN-NF-82-06(P), Supplement 1, Revision 2," May 3, 1988.

11. Letter from H. Bernard (NRC) to G. F. Owsley (ENC), "Acceptance for Referencing of Topical Report XN-NF-512, Revision 1," July 22, 1982.

12. Letter from C. O. Thomas (NRC) to J. C. Chandler (ENC), "Acceptance for Referencing of Licensing Topical Report XN-NF-734, Confirmation of the XN-3 Critical Power Correlation for 9x9 Fuel Assemblies," February 1, 1985.

13. XN-NF-524(P)(A), Revision 1, "Exxon Nuclear Critical Power Methodology for BWRs," November 1983.

14. XN-NF-80-19(P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," September 1985.

15. XN-NF-80-19(P)(A), Volume 1 (and Supplements 1 & 2), "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," March 15, 1983.

16. TR-VR-85-225-A, "Topical Report of ASEA-ATOM BWR Control Blades for US BWRs," October 1985.

17. XN-NF-79-71(P), Revision 2 (and Supplements), "ENC Plant Transient Methodology for Boiling Water Reactors," November 1981.

18. XN-NF-85-62, "Dresden Unit 3 Cycle 10 Plant Transient Analysis," September 1985.

19. Letter from G. C. Lainas (NRC) to G. N. Ward (ENC), "Acceptance for Referencing of Licensing Topical Report XN-NF-84-105(P), XCOBRA-T: A Computer Code for BWR Transient Thermal Hydraulic Core Analysis," October 27, 1986.

20. XN-NF-80-19(P)(A), Volume 3, Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors - THERMEX: Thermal Limit Methodology Summary Description," January 1987.

21. XN-NF-79-71(P), Revision 2, Supplement 3, "Revised Methodology for Including Uncertainties in Determining Operating Limits for Rapid Pressurization Transients in BWRs," March 1985.

22. CN-NF-81-75(P), "Dresden Unit 3 LOCA Analysis Using the ENC EXEM Evaluation Model," November 1981.

23. XN-NF-85-63, "Dresden Unit 3 LOCA-ECCS Analysis MAPLHGR Results for 9X9 Fuel," September 1985.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.32 the Commission has determined that granting this amendment will have no significant impact on the environment (53 FR 18361).

6.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security nor to the health and safety of the public.

Principal Contributor: L. Kopp, NRR/SRXB

Dated: June 20, 1988