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NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

January 19, 1977

Mr Dennis L Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors
U S Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr Ziemann:

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

Response to 12/14/76 Questions on
Single Recirculation Pump Operation

This letter is in response to your December 14, 1976 request for additional information regarding our September 7, 1976 submittal on single recirculation pump operation with the equalizer valve closed. The nature of the questions suggests that the review of our submittal has been expanded beyond ECCS considerations to involve areas which have been previously analyzed.

The title of the report accompanying our September 7, 1976 letter, "License Amendment Submittal for Single-Loop Operation", is misleading. Single-Loop Operation is not being newly licensed. It was a design feature licensed with the plant and verified by the startup test program. It was an allowable mode of operation until issuance of an amendment to the Monticello license on October 30, 1975. The NRC safety evaluation of our August 4, 1975 license amendment request stated the following: "An evaluation was not provided for ECCS performance during reactor operation with one recirculation loop out of service. Therefore, continuous operation in excess of 24 hours under such conditions will not be permitted until the necessary analyses have been performed, evaluated and determined acceptable." Our September 7, 1976 submittal was prepared to provide the ECCS performance information that you requested.

Your recent questions and the respective answers are as follows:

1. The idle loop startup transient has been analyzed in your FSAR from an initial power of 60%. In NEDO-21252, Page 4-1, it states that "operation with one recirculation loop results in a maximum power output which is 20 to 30% below that from (sic) which can be attained for two-pump operation." Is 60% power the most severe initial power for the idle loop startup transient analysis? If not, revise the analysis using the most severe initial power level.

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NORTHERN STATES POWER COMPANY

Answer #1 - The most severe case of the idle loop startup transient is that case where initial power is at the highest level where a scram does not occur during the transient. That threshold corresponds to 60% power. It is true that greater than 60% power can be achieved with single-loop operation; however, an idle loop startup transient would then result in a neutron flux scram and less severe results.

2. What effect will reverse flow have on jet pump vibration, specifically risers, supports, and baffle plates?

3. What effect will asymmetric flow have on instrument housings located in the lower plenum?

Answer #2 and #3 - Single recirculation pump trips were included in the Monticello startup test program. Vibration transducers mounted on jet pumps, incore instrument guide tubes in the lower plenum area and numerous other locations inside the reactor vessel indicated movement during flow reversals and asymmetric flows. Measurements fell within pre-established limiting criteria. Results are reported in NEDO-10563. The Monticello results were considered confirmatory to and compatible with vibration tests at similar facilities; the results of all these tests have undergone extensive AEC review in the past.

We trust that this additional information will allow completion of our September 7, 1976 amendment request.

Yours very truly,

L. O. Mayer

L O Mayer, PE
Manager of Nuclear Support Services

LOM/ak

cc: J G Keppler
G Charnoff
MPCA
Attn: J W Ferman

(25) K. Eccleston

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

Docket No. 50-263

August 24, 1981

Mr. L. O. Mayer, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

RE: MONTICELLO NUCLEAR GENERATING PLANT

Dear Mr. Mayer:

On July 2, 1981, we sent a letter to all licensees who have requested approval to operate on a continuing basis at power levels above 50% with only one recirculation loop in the event the other loop is inoperative. You and other BWR licensees received a copy of one of these letters since we expect most BWR facilities would like to have this flexibility. In the letter we proposed a meeting to obtain a better understanding of what might have caused variations in jet pump flow and related parameters at Browns Ferry Unit No. 1 during single loop operation and how this incident should affect approval of single loop operation at other facilities.

You have indicated to your NRC project manager that you are interested in attending the proposed meeting. The meeting will be held at 9:00 A.M., Wednesday, September 9, 1981 in room P-118, Phillips Building, 7920 Norfolk Avenue, Bethesda, Maryland. You are requested to advise your project manager of the people who will be attending this meeting from your organization.

Sincerely,

for Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosure:
Meeting Agenda

cc w/enclosure
See next page

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Proposed Meeting with BWR Applicants and Licensees on Single Loop Operation

Purpose of Meeting:

1. To determine what may have caused the jet pump flow and other variations experienced by Browns Ferry Unit 1 during single loop operation and

2. Evaluate whether the Browns Ferry experience should result in power limits for other BWRs operating on a single loop.

Agenda:

1. Discussion of what may have caused the unexpected variations in operating parameters when Browns Ferry Unit 1 exceeded about 60 percent rated power while operating with only one recirculation loop.

2. Discussion of parameters affected at Browns Ferry 1 (i.e., jet pump flow, neutron flux, core flow, core pressure drop, etc.)

3. Discussion of whether the Browns Ferry 1 experience would be expected at other BWRs operating on one recirculation loop. If ~~so,~~ are safety limits likely to be violated or cause complications with respect to core stability, core flow symmetry, pump cavitation or damage to the jet pumps and reactor vessel internals.

4. Discussion of the benefits vs. potential problems and cost of testing single loop operation in another BWR that is instrumented to detect what parameters are affected.

5. Evaluation of whether single loop operation at power levels about 50 to 55 percent is a safe and prudent means of reactor operation.

Mr. L. O. Mayer
Northern States Power Company

cc:

Gerald Charnoff, Esquire
Sahw, Pittman, Potts and
Trowbridge
1800 M Street, N. W.
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Arthur Renquist, Esquire
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 23, 1981

Docket No. 50-263

Mr. L. O. Mayer, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

RE: MONTICELLO NUCLEAR GENERATING PLANT

Dear Mr. Mayer:

My letter to you of August 24, 1981 informed you that we were proposing a meeting with licensees who have requested approval to operate on a single recirculation loop. The purpose of the meeting is to determine what may have caused variations in jet pump flow at Browns Ferry Unit No. 1 while operating on a single loop and what impact this should have on approval of other facilities to operate on one loop. Licensees and applicants who have not requested approval for single loop operation were also invited to the meeting.

As you were advised by your licensing project manager, the meeting scheduled for September 9, 1981 had to be postponed to allow more time for analysis of relevant operating data. We apologize for this inconvenience. The meeting will be held at 9:00 AM on Thursday, October 22, 1981 in Room P-118, Phillips Building, 7920 Norfolk Avenue, Bethesda, Maryland. It would be appreciated if you would inform your licensing project manager of the number of people who will be attending this meeting from your organization.

Sincerely,

TA Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

cc: See Next Page

~~8110150069~~

Mr. L. O. Mayer
Northern States Power Company

cc:

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N/ⓓ

NSP

Northern States Power Company

414 Nicollet Mall
Minneapolis, Minnesota 55401
Telephone (612) 330-5500

July 2, 1982

Director
Office of Nuclear Reactor Regulation
U S Nuclear Regulatory Commission
Washington, DC 20555

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

Revision 1 to License Amendment Request Dated September 7, 1976
Single Loop Operation

Attached are 3 originals and 37 conformed copies of a request for change of the Technical Specifications, Appendix A of the Full Term Operating License for the Monticello Nuclear Generating Plant. This submittal supersedes our request dated September 7, 1976.

Because this request is a revision of an earlier amendment request, an additional license amendment fee is not required.

The proposed change will allow the plant to remain operational at a substantial power level with one recirculation pump in operation and the equalizer valve closed. Exhibits A and B present the proposed change to the Technical Specifications. Exhibit C is an updated analysis report which presents a safety evaluation in support of the change. Your review of this matter at an early date is requested.

DM Musolf

D M Musolf
Acting Head-Nuclear Support Services

DMM/SAF/bd

cc: Regional Admin-III, NRC
Resident Inspector, NRC
NRR Project Manager, NRC
G Charnoff
MPCA
Attn: J W Ferman

Attachment

A001

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

Docket No. 50-263

REQUEST FOR AMENDMENT TO
OPERATING LICENSE NO. DPR-22

Revision 1 to
License Amendment Request Dated September 7, 1976

Northern States Power Company, a Minnesota corporation, request authorization for changes to the Technical Specifications as shown on the attachments labeled Exhibit A and Exhibit B. Exhibit A describes the proposed changes along with reasons for the change. Exhibit B is a set of Technical Specification pages incorporating the proposed changes. Exhibit C is the Analysis report which supports the change.

This request contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By D M Musolf
D M Musolf
Acting Head-Nuclear Support Services

On this 2nd day of July, 1982, before me a notary public in and for said County, personally appeared D M Musolf Acting Head-Nuclear Support Services, and being first duly sworn acknowledged that he is authorized to execute this document on behalf of Northern States Power Company, that he knows the contents thereof and that to the best of his knowledge, information and belief, the statements made in it are true and that it is not interposed for delay.

Betty J Dean

BETTY J. DEAN
NOTARY PUBLIC - MINNESOTA
RAMSEY COUNTY
My Commission Expires Dec 16 1987

EXHIBIT A

Monticello Nuclear Generating Plant
Docket No. 50-263

Revision 1 to
License Amendment Requested dated September 7, 1976
Proposed Changes to Technical Specification
Appendix A of Operating
License No. DPR-22

Pursuant to 10 CFR 50.59 and 50.90, the holders of Operating License DPR-22 hereby proposed the following changes to Appendix A, Technical Specifications.

1. Pages 6, 7, 8 of Section 2.1; Pages 17 and 20 of the Section 2.3 Bases; Pages 56 and 57 of Section 3.2/4.2; Pages 114 and 114a of Section 3.5/4.5; Page 119 of the Section 3.5 Bases; Pages 211, 213 and 214 of Section 3.11; Page 215, 216 and 217 of the Section 3.11/4.11 Bases

   PROPOSED CHANGES

   Incorporate the changes as indicated in the proposed revised pages submitted as Exhibit B.

   REASON FOR CHANGE

   These proposed changes are additions to the existing Technical Specifications which are associated with a mode of operation involving only one reactor recirculation pump with the equalizer valves closed. It is desirable to have provisions for this mode of operation because reactor operation can safely continue at a substantial power level when equipment outages exist. The plant was initially designed and license to allow operation with only one recirculation pump. An in-depth analysis has now been completed and new, conservative limits are proposed such that the flexibility of one-pump operation can be restored.

SAFETY EVALUATION

The safety evaluations in support of the proposed changes are included as Exhibit C entitled, "Monticello Nuclear Generating Plant Single-Loop Operation, NEDO-24271."

2. Pages 14 and 15 of the Section 2.3 Bases and Page 20 of the Section 2.3 Bases

PROPOSED CHANGE

Delete the third paragraph on page 14 and revise the second paragraph as shown in Exhibit B. Delete the first two paragraphs on page 15.

REASON FOR CHANGE

A discussion of the conservatisms and methodology in the analyses is contained in reference (1). To eliminate confusion and possible conflicts the detailed discussion in the bases should be deleted.

SAFETY EVALUATION

This change does not affect the commitments required by the Technical Specifications.

EXHIBIT B

Revision 1 License Amendment Request Dated - Sept 7, 1976

Exhibit B, attached, consists of the following revised pages of the Appendix A Technical Specifications which incorporate the proposed changes.

Pages

6
7
8
14
15
17
20
56
57
114
114a (new page)
119
211
213
214
215
216
217

2.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variables associated with fuel thermal behavior.

Objectives:

To establish limits below which the integrity of the fuel cladding is preserved.

Specification:

A. Core Thermal Power Limit (Reactor Pressure $>$ 800 Psia and Core Flow is $>$ 10% of Rated)

When the reactor pressure is $>$ 800 Psia and core flow is $>$ 10% of rated, the existence of a minimum critical power ratio (MCPR) less than 1.07 for two recirculation loop operation or less than 1.08 for single loop operation for 8x8 and 8x8R fuel shall constitute violation of the fuel cladding integrity safety limit.

2.3 FUEL CLADDING INTEGRITY

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objectives:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

Specification:

The Limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM - The APRM flux scram trip setting shall be:

$S \leq 0.65 (W-dw) + 55\%$

where,

S = Setting of percent of rated thermal power, rated power being 1670 MWT

W = recirculation drive flow in percent

dw = single loop operation recirculation reverse flow in the idle loop.

dw = 0 For two recirculation loop operation

dw = 5.4 For one recirculation loop operation

B. Core Thermal Power Limit (Reactor Pressure $\leq$ 800 psia or Core Flow $\leq$ 10% of rated)

When the reactor pressure is $\leq$ 800 psia or core flow is $\leq$ 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

C. Power Transients

To insure that the safety limit established in Specification 2.1.A is not exceeded, each required scram shall be initiated by its primary source signal as indicated by the plant process computer

except in the event of operation with a maximum fraction of limiting power density for any fuel type in the core greater than the fraction of rated power, when the setting shall be modified as follows:

$$S \leq [0.65\ (W-dw) + 55\%]\ \frac{FRP}{MFLPD}$$

where,

FRP = fraction of rated thermal power, rated power being 1670 MWt

MFLPD = maximum fraction of limiting power density for any fuel type in the core.

2. IRM - Flux Scram setting shall be 20% of rated neutron flux

B. APRM Rod Block - The APRM rod block setting shall be:

$$S \leq 0.65\ (W-dw) + 43\%$$

where,

S = Setting of percent of rated thermal power, rated power being 1670 MWT

W = recirculation drive flow in percent

dw = Single loop operation recirculation reverse flow in the idle loop.

dw = 0 For two recirculation loop operation

dw = 5.4 For one recirculation loop operation

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. This level shall be continuously monitored whenever the recirculation pumps are not operating.

except in the event of operation with a maximum fraction of limiting power density for any fuel type in the core greater than the fraction of rated power, when the setting shall be modified as follows:

$$S \leq [0.65\ (W\text{-}dw) + 43\%]\ \frac{FRP}{MFLPD}$$

where,

FRP = fraction of rated thermal power, rated power being 1670 MWt

MFLPD = maximum fraction of limiting power density for any fuel type in the core.

C. Reactor Low Water Level Scram setting shall be $\geq$ 10'6" above the top of the active fuel.

D. Reactor Low Low Water Level ECCS initiation shall be $\geq$ 6'6" $\leq$ 6'10" above the top of the active fuel.

Bases:

2.3 The abnormal operational transients applicable to operation of the Monticello Unit have been analyzed throughout the spectrum of planned operating conditions up to the thermal power level of 1670 MWt. The analyses were based upon plant operation in accordance with the operating map given in Figure 3-2-3 of the FSAR. The licensed maximum power level 1670 MWt represents the maximum steady-state power which shall not knowingly be exceeded.

Transient analysis performed each reload are given in Reference 1. Models and model conservatisms are also described in this reference. As discussed in Reference 2, the core wide transient analysis for one recirculation pump operation is conservatively bounded by two-loop operation analysis and the flow-dependent rod block and scram setpoint equations are adjusted for one-pump operation.

Bases Continued:

Deviations from as-left settings of setpoints are expected due to inherent instrument error, operator setting error, drift of the setpoint, etc. Allowable deviations are assigned to the limiting safety system settings for this reason. The effect of settings being at their allowable deviation extreme is minimal with respect to that of the conservatisms discussed above. Although the operator will set the setpoints within the trip settings specified, the actual values of the various setpoints can vary from the specified trip setting by the allowable deviation.

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip setting or when a sufficient number of devices have been affected by any means such that the automatic function is incapable of preventing a safety limit from being exceeded while in a reactor mode in which the specified function must be operable. Sections 3.1 and 3.2 list the reactor modes in which the functions listed above are required.

The bases for individual trip settings are discussed in the following paragraphs.

A. Neutron Flux Scram The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (1670 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operation transients, the thermal power of the fuel will be less than

Bases Continued:

backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated power per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The IRM scram remains active until the mode switch is placed in the run position. This switch occurs when reactor pressure is greater than 850 psig.

The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps. During steady-state operation with one recirculation pump operating the equalizer line shall be closed. Analysis of transients from this operating condition are less severe than the same transients from the two pump operation.

The operator will set the APRM neutron flux trip setting no greater than that stated in Specification 2.3.A.1. However, the actual setpoint can be as much as 3% greater than that stated in Specification 2.3.A.1 for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on page 18.

B. APRM Control Rod Block Trips 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 rod withdrawal beyond a given point at constant recirculation flow rate, and thus to protect against the condition of a MCPR less than the Safety Limit (T.S.2.1.A). 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

Bases Continued:

that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of the neutron scram protection over the entire range of applicability of the fuel cladding integrity safety limit.

The operator will set this pressure trip at greater than or equal to 825 psig. However, the actual trip setting can be as much as 10 psi lower due to the deviations discussed on page 18.

References

1. "Generic Reload Fuel Application", NEDE 24011-P-A-1, July 1979

2. "Monticello Nuclear Generating Plant Single-Loop Operation" NEDO 24271, June 1980

Table 3.2.3
Instrumentation That Initiates Rod Block

Function	Trip Settings	Reactor Modes in Which Function Must be Operable or Operating and Allowable Bypass Conditions**			Total No. of Instrument Channels per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (Notes 1,6)	Required Conditions*
		Refuel	Startup	Run			
1. SRM							
a. Upscale	$\leq 5x10^5$ cps	X	X(d)		2	1 (Note 3)	A or B or C
b. Detector not fully inserted		X(a)	X(a)		2	1 (Note 3)	A or B or C
2. IRM							
a. Downscale	$\geq$ 3/125 full scale	X(b)	X(b)		4	2 (Note 4)	A or B or C
b. Upscale	$\leq$ 108/125 full scale	X	X		4	2 (Note 4)	A or B or C
3. APRM							
a. Upscale (flow referenced)	See Technical Specifications 2.3.B.			X	3	1 (Note 7)	D or E
b. Downscale	$\geq$ 3/125 full scale			X	3	1 (Note 7)	D or E

Table 3.2.3 - continued
Instrumentation That Initiates Rod Block

Function	Trip Settings	Reactor Modes in Which Function Must be Operable or Operating and Allowable Bypass Conditions**			Total No. of Instrument Channels per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (Notes 1,6)	Required Conditions*
		Refuel	Startup	Run			
4. RBM							
a. Upscale (flow referenced)	See Technical Specifications 2.3.B			X(c)	1	1 (Note 5)	D or E
b. Downscale	≥3/125 full			X(c)	1	1 (Note 5)	D or E
5. Scram Discharge Volume							
Water Level-High	≤18 gal		X	X	1	1	B and D, or A

Notes:

(1) There shall be two operable or operating trip systems for each function. If the minimum number of operable or operating instrument channels cannot be met for one of the two trip systems, this condition may exist up to seven days provided that during this time the operable system is functionally tested immediately and daily thereafter.

(2) (deleted)

(3) Only one of the four SRM channels may be bypassed.

(4) There must be at least one operable or operating IRM channel monitoring each core quadrant.

(5) One of the two RBMs may be bypassed for maintenance and/or testing for periods not in excess of 24 hours in any 30 day period. An RBM channel will be considered inoperable if there are less than half the total number of normal inputs from any LPRM level.

3.0 LIMITING CONDITIONS FOR OPERATION

I. Recirculation System

1. Except as specified in 3.5.1.2 below, whenever irradiated fuel is in the reactor, with reactor coolant temperature greater than 212°F and both reactor recirculation pumps operating, the recirculation system cross tie valve interlocks shall be operable.

2. The recirculation system cross tie valve interlocks may be inoperable if at least one cross tie valve is maintained fully closed.

3. Reactor operation with one loop recirculation may continue at up to 50% of rated power if the following conditions are met within 24 hours after one pump operation commences. If the conditions cannot be met or two pump operation cannot be restored by the end of 24 hours, an orderly reactor shutdown shall be initiated.

   a. The Minimum Critical Power Ratio (MCPR) Safety Limit will be increased per T.S. 2.1.A

   b. The MCPR Limiting Condition for Operation (LCO) will be changed per T.S. 3.11.C.

   c. The Maximum Average Planar Linear Heat Generation (MAPLHGR) will be changed as noted in Table 3.11.1

4.0 SURVEILLANCE REQUIREMENTS

I. Recirculation System

1. Once per month, when irradiated fuel is in the reactor with reactor coolant temperature greater than 212°F and both reactor recirculation pumps operating, the recirculation system cross time valve interlocks shall be demonstrated to be operable by verifying that the cross tie valves cannot be opened using the normal control switch.

2. When a recirculation system cross tie valve interlock is inoperable, the position of at least one fully closed cross tie valve shall be recorded daily.

3. When in one loop operation, the following surveillances will be completed:

   a. APRM flux noise will be measured once per shift and the recirculation pump speed will be reduced if the flux noise average over ½ hour exceeds 5% peak to peak as measured on the APRM chart recorder.

   b. The core plate delta P noise will be measured once per shift and the recirculation pump speed will be reduced if the noise exceeds 1 psi peak to peak.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
d. The APRM scram and rod block setpoints and the RBM setpoints shall be reduced as noted in T.S. 2.3.A and T.S. 2.3.B. e. The suction valve or the main discharge and main discharge bypass valves in the idle loop is closed and electrically isolated until the idle loop is being prepared for return to service. f. The equalizer line shall be isolated.	

Bases Continued 3.5:

G. Emergency Cooling Availability

The purpose of Specification G is to assure that sufficient core cooling equipment is available at all times. It is during refueling outages that major maintenance is performed and during such time that all core and containment cooling subsystems may be out of service. Specification 3.5.G.3 allows all core and containment cooling subsystems to be inoperable provided no work is being done which has the potential for draining the reactor vessel. Thus events requiring core cooling are precluded.

Specification 3.5.G.4 recognizes that concurrent with control rod drive maintenance during the refueling outage, it may be necessary to drain the suppression chamber for maintenance or for the inspection required by Specification 4.7.A.1. In this situation, a sufficient inventory of water is maintained to assure adequate core cooling in the unlikely event of loss of control rod drive housing or instrument thimble seal integrity.

H. Deleted

I. Recirculation System

The capacity of the Emergency Core Coolant System is based on the potential consequences of a double ended recirculation line break. Such a break involves 3.9 sq. ft. when the cross tie valves are closed and 5.3 sq. ft. when the cross tie valves are open. Specification 3.11.A is based on an ECCS evaluation assuming a break area of 3.9 sq. ft.; the limitations of 3.11.A do not apply to the larger break area. Therefore, at least one cross tie valve must remain closed during power operation to reduce the potential break area.

An analysis of one-pump operation (equalizer valve closed) identifies certain limitations peculiar to that mode of operation. Reference the September 7, 1976 License Amendment Request from NSP to NRR. Operation with only one pump is not a normal mode; it will generally involve a forced outage of equipment. There may be insufficient time to make adjustments to the RBM and APRM flow referenced rod block and scram prior to commencing one-pump operation. The reduction in power with the reduced core flow will cause the APLHGR to reduce accordingly, naturally moving in the direction of the new limit. Specification 3.5.I.3 allows 24 hours before these new limits are required to be implemented.

3.11 REACTOR FUEL ASSEMBLIES

Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed for two recirculation loop operation the limiting value given in Table 3.11.1 based on a straight line interpolation between data points and for one recirculation loop operation the values in Table 3.11.1 reduced by 0.85 for all fuel types. When core flow is less than 90% of rated core flow, the APLHGR shall not exceed 95% of the limiting value given in Table 3.11.1. When core flow is less than 70% of rated core flow, the APLHGR shall not exceed 90% of the limiting value given in Table 3.11.1. If any time during operation it is determined that the limit for APLHGR is being exceeded, action shall be initiated within 15

4.11 REACTOR FUEL ASSEMBLIES

Applicability

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective

The objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at 25% rated thermal power.

3.0 LIMITING CONDITIONS FOR OPERATION

C. Minimum Critical Power Ratio (MCPR)

1. During power operation the Operating MCPR Limit shall be ≥ 1.43 for 8x8 and 8x8R fuel, ≥ 1.47 for P8x8R fuel at rated power and flow for two recirculation loop operation, provided $\tau_B \geq \tau_{AVE}$* (see section 3.3.C.3). If at any time during 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. Surveillance and corresponding action shall continue until reactor operation is 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 conditions within 36 hours. For core flows other than rated the Operating MCPR Limit shall be the above applicable MCPR value time K_f where K_f is as shown in Figure 3.11.3.

   For one recirculation loop operation the MCPR limits at rated flow are 0.01 higher than the comparable two-loop values.

2. If the gross radioactivity release rate of noble gases at the steam jet air ejector monitors exceeds, for a period greater than 15 minutes, the equivalent of 236,000 uCi/sec following a 30-minute decay, the Operating MCPR Limits specified in 3.11.C.1 shall be adjusted to ≥ 1.48 for all fuel types, times the appropriate K_f. Subsequent operation with the adjusted MCPR values shall be per paragraph 3.11.C.1.

   For one recirculation loop operation the MCPR limits at rated flow are 0.01 higher than the comparable two-loop values.

*If $\tau_{AVE} > \tau_a$, the operating MCPR Limit shall be a linear interpolation between the limits in 3.11.C.1 and 1.48 for 8x8 and 8x8R fuel and 1.52 for P8x8R fuel.

4.0 SURVEILLANCE REQUIREMENTS

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at 25% rated thermal power and following any change in power level or distribution which has the potential of bringing the core to its operating MCPR Limit.

TABLE 3.11.1

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE vs. EXPOSURE

Exposure MWD/STU	MAPLHGR FOR EACH FUEL TYPE (kw/ft) (Note 1)							
	8DB262	8DB250	8DB219L	8DRB265L	P8DRB265L	8DRB282	P8DRB282	P8DRB284LB
200	11.1	11.2	11.4	11.5	11.6	11.2	11.2	11.4
1,000	11.3	11.3	11.5	11.6	11.6	11.2	11.2	11.4
5,000	11.9	11.9	11.9	11.7	11.8	11.6	11.8	11.8
10,000	12.1	12.1	12.0	11.8	11.9	11.7	11.9	11.9
15,000	12.1	12.1	11.9	11.7	11.9	11.7	11.8	11.9
20,000	12.0	11.9	11.8	11.6	11.8	11.5	11.7	11.7
25,000	11.6	11.5	11.3	11.3	11.3	11.3	11.3	11.4
30,000	10.3	10.6	10.2	10.3	10.5	10.4	10.7	10.6
35,000	9.3	9.3	9.3	9.2	9.5	9.2	9.5	9.5
(36,000)	9.1	9.0	9.1	9.0	9.3	9.0	9.3	9.3
40,000	8.9*							
45,000	8.0*							
50,000	7.3*							

(1) For two recirculation loop operation. For one recirculation loop operation multiply these values by 0.85

*For extended burnup program test bundles

Bases 3.11

A. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10CFR50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak cladding temperature by less than $\pm$ 20° relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10CFR50 Appendix K limit. The limiting value for APLHGR is given by this specification.

Reference 6 demonstrates that for lower initial core flow rates the potential exists for earlier DNB during postulated LOCA's. Therefore a more restrictive limit for APLHGR is required during reduced flow conditions.

Those abnormal operational transients, analyzed in FSAR Section 14.5, which result in an automatic reactor scram are not considered a violation of the LCO. Exceeding APLHGR limits in such cases need not be reported.

Reduction factors for one recirculation loop operation were derived in Reference 8.

B. LHGR

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 1 and in References 2 and 3, and assumes a linearly increasing variation and axial gaps between core bottom and top and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking.

Those abnormal operational transients, analyzed in FSAR Section 14.5, which result in an automatic reactor scram are not considered a violation of the LCO. Exceeding LHGR limits in such cases need not be reported.

Bases Continued

C. Minimum Critical Power Ratio (MCPR)

The ECCS evaluation presented in Reference 4 and Reference 6 assumed the steady state MCPR prior to the postulated loss-of-coolant accident to be 1.24 for all fuel types for normal and reduced flow. The Operating MCPR Limit for two recirculation loop operation is determined from the analysis of transients discussed in Bases Sections 2.1 and 2.3. By maintaining an operating MCPR above these limits, the Safety Limit (T.S. 2.1.A) is maintained in the event of the most limiting abnormal operational transient.

For one recirculation loop operation the MCPR limits at rated flows are 0.01 higher than the comparable two-loop values.

Use of GE's new ODYN code Option B will require average scram time to be a factor in determining the MCPR (Reference 7). In order to increase the operating envelope for MCPR below $MCPR_A$ (ODYN code Option A), the cycle average scram time (τ_{AVE}) must be determined (see Bases 3.3.C). If τ_{AVE} is below the adjusted analysis scram time, the $MCPR_B$ Limit can be used. If $\tau_{AVE} > \tau_B$ a linear interpolation must be used to determine the appropriate MCPR. For example:

$$MCPR = MCPR_B + \frac{\tau_{AVE} - \tau_B}{0.9 - \tau_B}(MCPR_A - MCPR_B)$$

$MCPR_A$ and $MCPR_B$ have been determined from the most limiting accident analyses.

For operation with less than rated core flow the Operating MCPR Limit is adjusted by multiplying the above limit by K_f. Reference 5 discusses how the transient analysis done at rated conditions encompasses the reduced flow situation when the proper K_f factor is applied.

Noble gas activity levels above that stated in 3.11.C.2 are indicative of fuel failure. Since the failure mode cannot be positively identified, a more conservative Operating MCPR Limit must be applied to account for a possible fuel loading error.

Those abnormal operational transients, analyzed in FSAR Section 14.5, which result in an automatic reactor scram are not considered a violation of the LCO. Exceeding MCPR limits in such cases need not be reported.

Bases Continued

References

1. "Fuel Densification Effects in General Electric Boiling Water Reactor Fuel," Supplements 6,7, and 8, NEDM-10735, August, 1973.

2. Supplement 1 to Technical Report on Densification of General Electric Reactor Fuels, December 14, 1974 (USAEC Regulatory Staff).

3. Communication: VA Moore to IS Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

4. "Loss-of-Coolant Accident Analysis Report for the Monticello Nuclear Generating Plant," NEDO-24050 -1, December, 1980, L O Mayer (NSP) to Director of Nuclear Reactor Regulation (USNRC), February 6, 1981.

5. "General Electric BWR Generic Reload Application for 8x8 Fuel," NEDO-20360, Revision 1, November 1974.

6. "Revision of Low Core Flow Effects on LOCA Analysis for Operating BWR's," R L Gridley (GE) to D G Eisenhut (USNRC), September 28, 1977.

7. "Response to NRC Request for Information on ODYN Computer Mode," R H Buchholz (GE) to P S Check (USNRC), September 5, 1980.

8. "Monticello N.G.P. Single-loop Operation NEDO 24271, June 1980"

Bases 4.11

The APLHGR, LHGR and MCPR shall be checked daily to determine if fuel burnup, or control rod movement have caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are removed daily, a daily check of power distribution is adequate. For a limiting value to occur below 25% of rated thermal power, an unreasonably large peaking factor would be required, which is not the case for operating control rod sequences. In addition, the MCPR is checked whenever changes in the core power level or distribution are made which have the potential of bringing the fuel rods to their thermal-hydraulic limits.

NEDO-24271
80NED277
Class 1
June 1980

EXHIBIT C

Revision 1
License Amendment Request dated Sept 7, 1976

MONTICELLO NUCLEAR GENERATING PLANT
SINGLE-LOOP OPERATION

NUCLEAR ENERGY ENGINEERING DIVISION • GENERAL ELECTRIC COMPANY
SAN JOSE, CALIFORNIA 95125

GENERAL ELECTRIC

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ILLUSTRATIONS

TABLES

1. INTRODUCTION AND SUMMARY

The current technical specifications for the Monticello Nuclear Generating Plant do not allow plant operation beyond a relatively short period of time if an idle recirculation loop cannot be returned to service. The Monticello Nuclear Generating Plant (Technical Specification 3.6 G) shall not be operated for a period in excess of 24 hours with one recirculation loop out of service.

The capability of operating at reduced power with a single recirculation loop is highly desirable, from a plant availability/outage planning standpoint, in the event maintenance of a recirculation pump or other component renders one loop inoperative. To justify single-loop operation, the safety analyses documented in the Final Safety Evaluation Reports and Reference 1 were reviewed for one-pump operation. Increased uncertainties in the core total flow and TIP readings resulted in an 0.01 incremental increase in the MCPR fuel cladding integrity safety limit during single-loop operation. This 0.01 increase is reflected in the MCPR operating limit. No other increase in this limit is required as core-wide transients are bounded by the rated power/flow analyses performed for each cycle, and the recirculation flow-rate dependent rod block and scram setpoint equations given in the technical specifications are adjusted for one-pump operation. The least stable power/flow condition, achieved by tripping both recirculation pumps, is not affected by one-pump operation.

During single-loop operation the flow control should be in master manual since control oscillations might occur in the recirculation flow control system under automatic flow control conditions.

Derived MAPLHGR reduction factors are 0.85, 0.85, and 0.85 for the 8x8, 8x8R and P8x8R fuel types, respectively.

The analyses were performed assuming the equalizer valve was closed. The discharge valve in the idle recirculation loop is normally closed, but if its closure is prevented, the suction valve in the loop should be closed to prevent the partial loss of Low Pressure Coolant Injection (LPCI) flow through the recirculation pump into the downcomer degrading the intended LPCI performance.

2. MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT

Except for core total flow and TIP reading, the uncertainties used in the statistical analysis to determine the MCPR fuel cladding integrity safety limit are not independent on whether coolant flow is provided by one or two recirculation pumps. Uncertainties used in the two-loop operation analysis are documented in the FSAR for initial cores and in Table 5-1 of Reference 1 for reloads. A 6% core flow measurement uncertainty has been established for single-loop operation (compared to 2.5% for two-loop operation). As shown below, this value conservatively reflects the one standard deviation (one sigma) accuracy of the core flow measurement system documented in Reference 2. The random noise component of the TIP reading uncertainty was revised for single recirculation loop operation to reflect the operating plant test results given in Subsection 2.2 below. This revision resulted in a single-loop operation process computer uncertainty of 9.1% for reload cores. The comparable two-loop process computer uncertainty value is 8.7% for reload cores. The net effect of these two revised uncertainties is a 0.01 incremental increase in the required MCPR fuel cladding integrity safety limit.

2.1 CORE FLOW UNCERTAINTY

2.1.1 Core Flow Measurement During Single Loop Operation

The jet pump core flow measurement system is calibrated to measure core flow when both sets of jet pumps are in forward flow; total core flow is the sum of the indicated loop flows. For single-loop operation, however, the inactive jet pumps will be backflowing. Therefore, the measured flow in the backflowing jet pumps must be subtracted from the measured flow in the active loop. In addition, the jet pump flow coefficient is different for reverse flow than for forward flow, and the measurement of reverse flow must be modified to account for this difference.

For single-loop operation, the total core flow is derived by the following formula:

$$\begin{pmatrix}\text{Total Core}\\ \text{Flow}\end{pmatrix} = \begin{pmatrix}\text{Active Loop}\\ \text{Indicated Flow}\end{pmatrix} - C\begin{pmatrix}\text{Inactive Loop}\\ \text{Indicated Flow}\end{pmatrix}$$

where C (= 0.95) is defined as the ratio of "Inactive Loop True Flow" to "Inactive Loop Indicated Flow," and "Loop Indicated Flow" is the flow indicated by the jet pump "single-tap" loop flow summers and indicators, which are set to indicate forward flow correctly.

The 0.95 factor was the result of a conservative analysis to appropriately modify the single-tap flow coefficient for reverse flow.* If a more exact, less conservative core flow is required, special in-reactor calibration tests would have to be made. Such calibration tests would involve calibrating core support plate ΔP versus core flow during two-pump operation along the 100% flow control line, operating on one pump along the 100% flow control line, and calculating the correct value of C based on the core flow derived from the core support plate ΔP and the loop flow indicator readings.

2.1.2 Core Flow Uncertainty Analysis

The uncertainty analysis procedure used to establish the core flow uncertainty for one-pump operation is essentially the same as for two-pump operation, except for some extensions. The core flow uncertainty analysis is described in Reference 2. The analysis of one-pump core flow uncertainty is summarized below.

For single-loop operation, the total core flow can be expressed as follows (Figure 2-1):

$$W_C = W_A - W_I$$

where

W_C = total core flow;
W_A = active loop flow; and
W_I = inactive loop (true) flow.

---

*The expected value of the "C" coefficient is ~0.88.

By applying the "propagation of errors" method to the above equation, the variance of the total flow uncertainty can be approximated by:

$$\sigma^2_{W_C} = \sigma^2_{W_{sys}} + \left(\frac{1}{1-a}\right)^2 \sigma^2_{W_{A_{rand}}} + \left(\frac{a}{1-a}\right)^2 \left(\sigma^2_{W_{I_{rand}}} + \sigma^2_C\right)$$

where

σ_{W_C} = uncertainty of total core flow;

$\sigma_{W_{sys}}$ = uncertainty systematic to both loops;

$\sigma_{W_{A_{rand}}}$ = random uncertainty of active loop only;

$\sigma_{W_{I_{rand}}}$ = random uncertainty of inactive loop only;

σ_C = uncertainty of "C" coefficient; and

a = ratio of inactive loop flow (W_I) to active loop flow (W_A).

Resulted from an uncertainty analysis, the conservative, bounding values of $\sigma_{W_{sys}}$, $\sigma_{W_{A_{rand}}}$, $\sigma_{W_{I_{rand}}}$ and σ_C are 1.6%, 2.6%, 3.5% and 2.8%, respectively. Based on above uncertainties and a bounding value of 0.36 for "a", the variance of the total flow uncertainty is approximately:

$$\sigma^2_{W_C} = (1.6)^2 + \left(\frac{1}{1-0.36}\right)^2 (2.6)^2 + \left(\frac{0.36}{1-0.36}\right)^2 \left[(3.5)^2 + (2.8)^2\right]$$

$$= (5.0\%)^2$$

When the effect of 4.1% core bypass flow split uncertainty at 12% (bounding case) bypass flow fraction is added to the above total core flow uncertainty, the active coolant flow uncertainty is:

$$\sigma^2_{\substack{\text{active}\\\text{coolant}}} = (5.0\%)^2 + \left(\frac{0.12}{1-0.12}\right)^2 (4.1\%)^2 = (5.0\%)^2$$

which is less than the 6% core flow uncertainty assumed in the statistical analysis.

In summary, core flow during one-pump operation is measured in a conservative way and its uncertainty has been conservatively evaluated.

2.2 TIP READING UNCERTAINTY

To ascertain the TIP noise uncertainty for single recirculation loop operation, a test was performed at an operating BWR. The test was performed at a power level 59.3% of rated with a single recirculation pump in operation (core flow 46.3% of rated). A rotationally symmetric control rod pattern existed prior to the test.

Five consecutive traverses were made with each of five TIP machines, giving a total of 25 traverses. Analysis of their data resulted in a nodal TIP noise of 2.85%. Use of this TIP noise value as a component of the process computer total uncertainty results in a one-sigma process computer total uncertainty value for single-loop operation of 9.1% for reload cores.

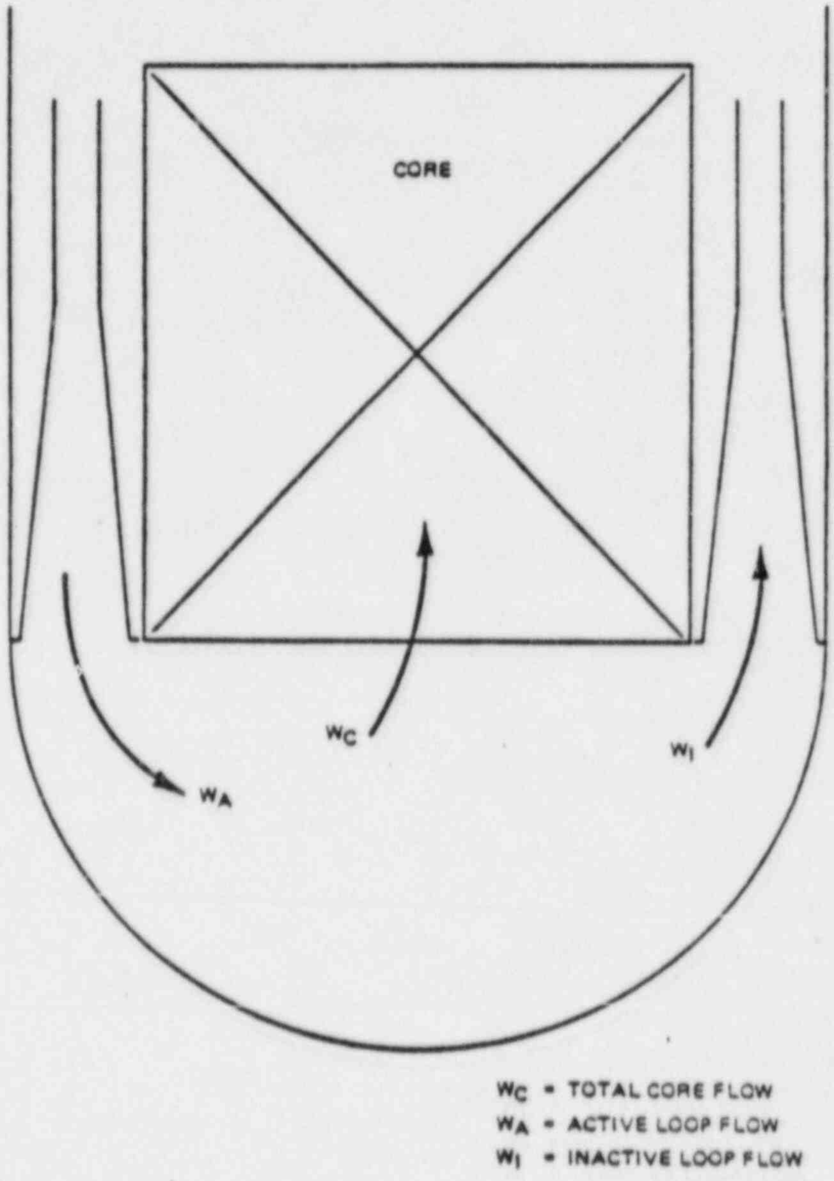

Figure 2-1. Illustration of Single Recirculation Loop Operation Flows

3. MCPR OPERATING LIMIT

3.1 CORE-WIDE TRANSIENTS

Operation with one recirculation loop results in a maximum power output which is 20 to 30% below that which is attainable for two-pump operation. Therefore, the consequences of abnormal operational transients from one-loop operation will be considerably less severe than those analyzed from a two-loop operational mode. For pressurization, flow decrease and cold water increase transients, previously transmitted Reload/FSAR results bound both the thermal and overpressure consequences of one-loop operation.

Figure 3-1 shows the consequences of a typical pressurization transient (turbine trip) as a function of power level. As can be seen, the consequences of one-loop operation are considerably less because of the associated reduction in operating power level.

The consequences from flow decrease transients are also bounded by the full power analysis. A single pump trip from one-loop operation is less severe than a two-pump trip from full power because of the reduced initial power level.

Cold water increase transients can result from either recirculation pump speedup or restart, or introduction of colder water into the reactor vessel by events such as loss of feedwater heater. The K_f factors are derived assuming that both recirculation loops increase speed to the maximum permitted by the M-G set scoop tube position. This condition produces the maximum possible power increase and, hence, maximum ΔCPR for transients initiated from less than rated power and flow. When operating with only one recirculation loop, the flow and power increase associated with the increased speed on only one M-G set will be less than that associated with both pumps increasing speed; therefore, the K_f factors derived with the two-pump assumption are conservative for single-loop operation. Inadvertent restart of the idle recirculation pump would result in a neutron flux transient which would exceed the flow reference scram. The resulting scram is expected to be less severe than the rated power/flow case documented in the FSAR. The latter event (loss of feedwater heating) is generally the most severe cold water increase event with respect to increase in core power. This event is caused by positive reactivity insertion from core flow inlet subcooling; therefore, the event is primarily dependent on the initial power level. The higher the initial power level, the greater the CPR change during the transient. Since the initial power level during one-pump operation will be significantly lower, the one-pump cold water increase case is conservatively bounded by the full power (two-pump) analysis.

From the above discussions, it can be concluded that the transient consequence from one-loop operation is bounded by previously submitted full power analysis.

3.2 ROD WITHDRAWAL ERROR

The rod withdrawal error at rated power is given in the FSAR for the initial core and in cycle-dependent reload supplemental submittals. These analyses are performed to demonstrate that, even if the operator ignores all instrument indications and the alarm which could occur during the course of the transient, the rod block system will stop rod withdrawal at a minimum critical power ratio (MCPR) which is higher than the fuel cladding integrity safety limit. Correction of the rod block equation (below) and lower power assures that the MCPR safety limit is not violated.

One-pump operation results in backflow through 10 of the 20 jet pumps while the flow is being supplied into the lower plenum from the 10 active jet pumps. Because of the backflow through the inactive jet pumps, the present rod block equation was conservatively modified for use during one-pump operation because the direct active-loop flow measurement may not indicate actual flow above about 35% drive flow without correction.

A procedure has been established for correcting the rod block equation to account for the discrepancy between actual flow and indicated flow in the active loop. This preserves the original relationship between rod block and actual effective drive flow when operating with a single loop.

The two-pump rod block equation is:

$$RB = mW + \left[RB_{100} - m(100)\right]$$

The one-pump equation becomes:

$$RB = mW + \left[RB_{100} - m(100)\right] - m\Delta W$$

where

ΔW = difference, determined by utility, between two-loop and single-loop effective drive flow at the same core flow;

RB = power at rod block in %;

m = flow reference slope for the rod block monitor (RBM);

W = drive flow in % of rated; and

RB_{100} = top level rod block at 100% flow.

If the rod block setpoint (RB_{100}) is changed, the equation must be recalculated using the new value.

The APRM trip settings are flow biased in the same manner as the rod block monitor trip setting. Therefore, the APRM rod block and scram trip settings are subject to the same procedural changes as the rod block monitor trip setting discussed above.

3.3 OPERATING MCPR LIMIT

For single-loop operation, the rated condition steady-state MCPR limit is increased by 0.01 to account for the increase in the fuel cladding integrity safety limit (Section 2). At lower flows, the steady-state MCPR operating limit is conservatively established by multiplying the rated flow steady-state limit by the K_f factor. This ensures that the 99.9% statistical limit requirement is always satisfied for any postulated abnormal operational transient.

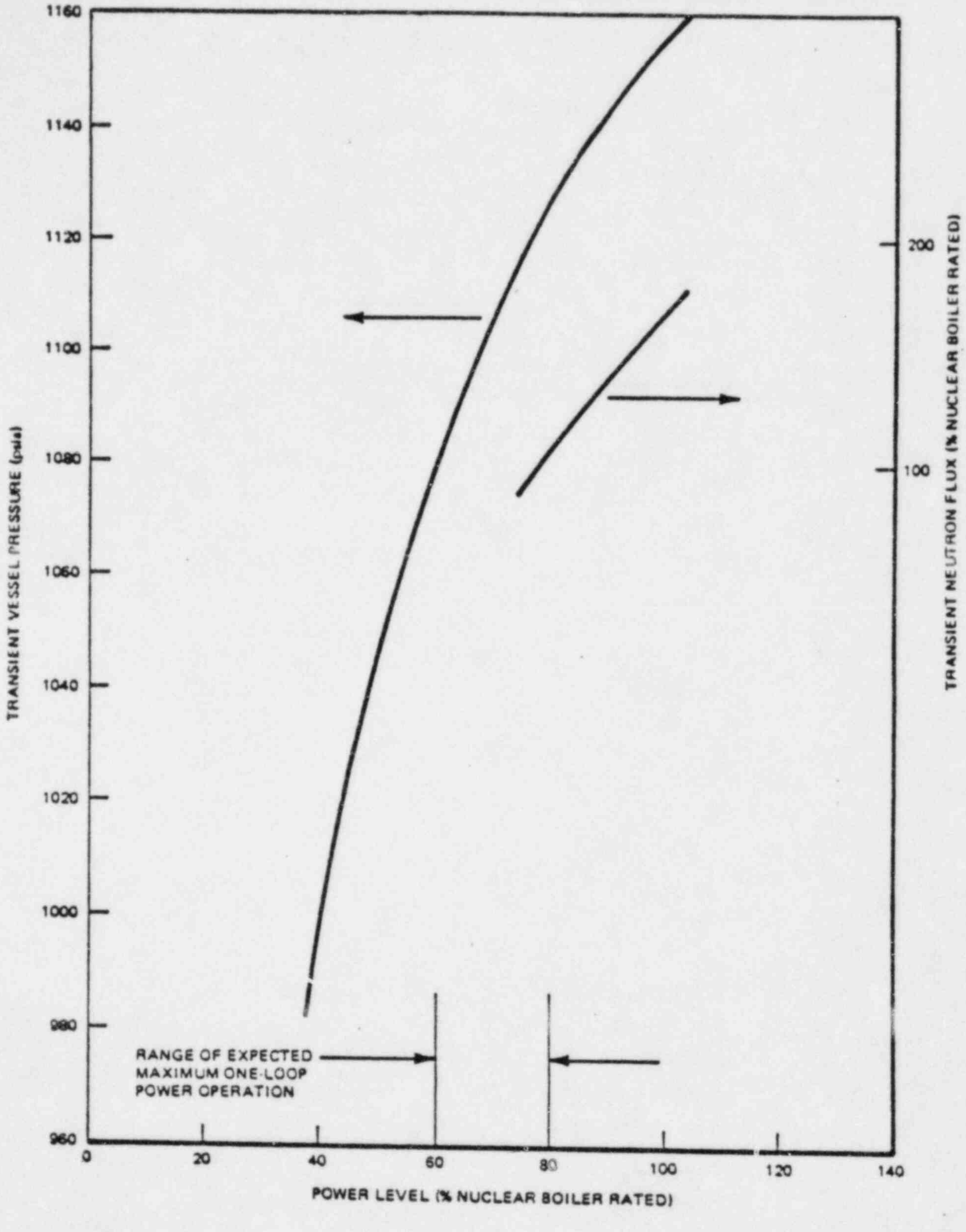

Figure 3-1. Main Turbine Trip with Bypass Manual Flow Control

4. STABILITY ANALYSIS

The least stable power/flow condition attainable under normal conditions occurs at natural circulation with the control rods set for rated power and flow. This condition may be reached following the trip of both recirculation pumps. As shown in Figure 4-1, operation along the minimum forced recirculation line with one pump running at minimum speed is more stable than operating with natural circulation flow only, but is less stable than operating with both pumps operating at minimum speed.

During single-loop operation, the flow control should be in master manual, since control oscillations might occur in the recirculation flow control system under automatic flow control conditions.

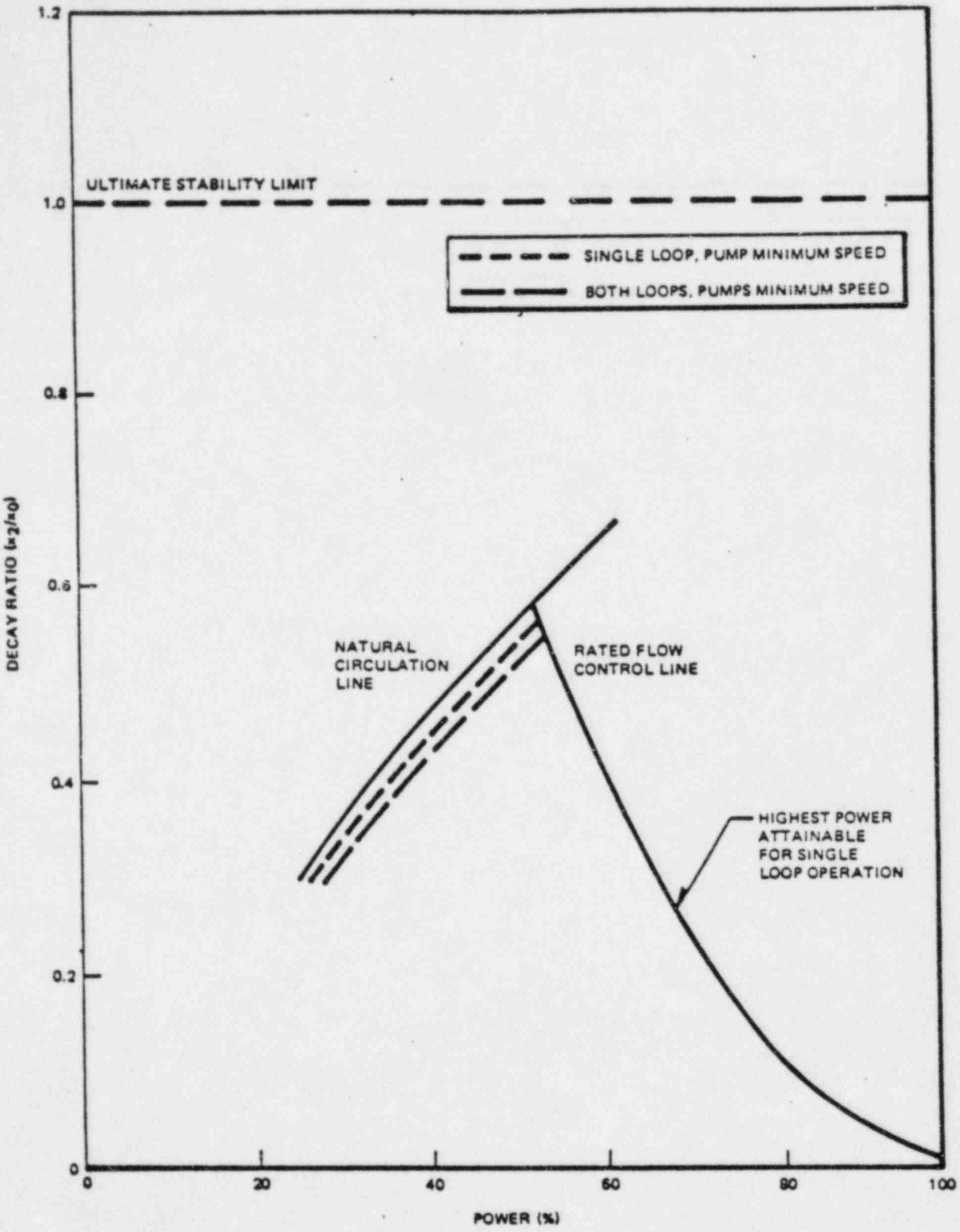

Figure 4-1. Decay Ratio Versus Power Curve for Two-Loop and Single-Loop Operation

5. ACCIDENT ANALYSES

The broad spectrum of postulated accidents is covered by six categories of design basis events. These events are the loss-of-coolant, recirculation pump seizure, control rod drop, main steamline break, refueling, and fuel assembly loading accidents. The analytical results for the loss-of-coolant and recirculation pump seizure accidents with one recirculation pump operating are given below. The results of the two-loop analysis for the last four events are conservatively applicable for one-pump operation.

5.1 LOSS-OF-COOLANT ACCIDENT ANALYSIS

A single-loop operation analysis utilizing the models and assumptions documented in Reference 3 was performed for the Monticello Nuclear Generating Plant. Using this method, SAFE/REFLOOD computer code runs were made for a full spectrum of break sizes for the suction breaks. Because the reflood time for the single-loop analysis is similar to the two-loop analysis, the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) curves currently applied were modified by derived reduction factors for use during one recirculation pump operation.

5.1.1 Break Spectrum Analysis

A break spectrum analysis was performed using the SAFE/REFLOOD computer codes and the assumptions given in Section II.A.7.2.2. of Reference 3.

Since the suction break is the most limiting, the suction break spectrum reflood times for one recirculation loop operation are compared to the standard previously performed two-loop operation in Figure 5-1. The uncovered time (reflood time minus recovery time) for the suction break spectrum is compared in Figure 5-2.

For the Monticello Nuclear Generating Plant, the maximum reflooding time for the standard two-loop analysis is 345 seconds with a boiling transition time within 9 sec, occurring at 40% of the Design Basis Accident (DBA) suction break, which is the most limiting break for the two-loop operation. For the single-loop analysis, the maximum reflooding time is 351 seconds, occurring at 40% DBA suction break. These uncovered times can be considered similar.

5.1.2 Single-Loop MAPLHGR Determination

The small differences in uncovered time and reflood time for the limiting break size would result in a small increase in the calculated peak cladding temperature. Therefore, as noted in Reference 3, the one- and two-loop SAFE/REFLOOD results can be considered similar and the generic alternative procedure described in Section II.A.7.4. of this reference was used to calculate the MAPLHGR reduction factors for single-loop operation.

MAPLHGR reduction factors were determined for the cases given in Table 5-1. The most limiting reduction factors for each fuel type is shown in Table 5-2.

One-loop operation MAPLHGR values are derived by multiplying the current two-loop operation MAPLHGR values by the reduction factor for that fuel type. As discussed in Reference 3, single recirculation loop MAPLHGR values are conservative when calculated in this manner.

5.1.3 Small Break Peak Cladding Temperature

Section II.A.7.4.4.2 of Reference 3 discusses the small sensitivity of the calculated peak clad temperature (PCT) to the assumptions used in the one-pump operation analysis and the duration of nucleate boiling. Since the slight increase (~50°F) in PCT is overwhelmingly offset by the decreased MAPLHGR (equivalent to 300° to 500°F ~PCT) for one pump operation, the calculated PCT values for small breaks will be well below the 2200°F 10CFR50.46 cladding temperature limit.

5.2 ONE-PUMP SEIZURE ACCIDENT

The one-pump seizure accident is a relatively mild event during two recirculation pump operation, as documented in References 1 and 2. Similar analyses were performed to determine the impact this accident would have on one recirculation pump operation. These analyses were performed with the models documented in Reference 1 for a large core BWR/4 plant (Reference 4). The analyses were initialized from steady-state operation at the following initial conditions, with the added condition of one inactive recirculation loop. Two sets of initial conditions were assumed:

(1) Thermal Power = 75% and core flow = 58%

(2) Thermal Power = 82% and core flow = 56%

These conditions were chosen because they represent reasonable upper limits of single-loop operation within existing MAPLHGR and MCPR limits at the same maximum pump speed. Pump seizure was simulated by setting the single operating pump speed to zero instantaneously.

The anticipated sequence of events following a recirculation pump seizure which occurs during plant operation with the alternate recirculation loop out of service is as follows:

(1) The recirculation loop flow in the loop in which the pump seizure occurs drops instantaneously to zero.

(2) Core voids increase which results in a negative reactivity insertion and a sharp decrease in neutron flux.

(3) Heat flux drops more slowly because of the fuel time constant.

(4) Neutron flux, heat flux, reactor water level, steam flow, and feedwater flow all exhibit transient behaviors. However, it is not anticipated that the increase in water level will cause a turbine trip and result in scram.

It is expected that the transient will terminate at a condition of natural circulation and reactor operation will continue. There will also be a small decrease in system pressure.

The minimum CPR for the pump seizure accident for the large core BWR/4 plant was determined to be greater than the fuel cladding integrity safety limit; therefore, no fuel failures were postulated to occur as a result of this analyzed event.

These results are applicable to the Monticello Nuclear Generating Plant.

Table 5-1

MAPLHGR MULTIPLIFR CASES

Fuel Type	Cases Calculated
8x8	100% DBA Suction Break
	40% DBA Suction Break*
8x8R/P8x8R	100% DBA Suction Break
	40% DBA Suction Break*

*Most limiting break for MAPLHGR reduction factors.

Table 5-2

LIMITING MAPLHGR REDUCTION FACTORS

Fuel Type	Reduction Factors
8x8	0.85
8x8R	0.85
P8x8R	0.85

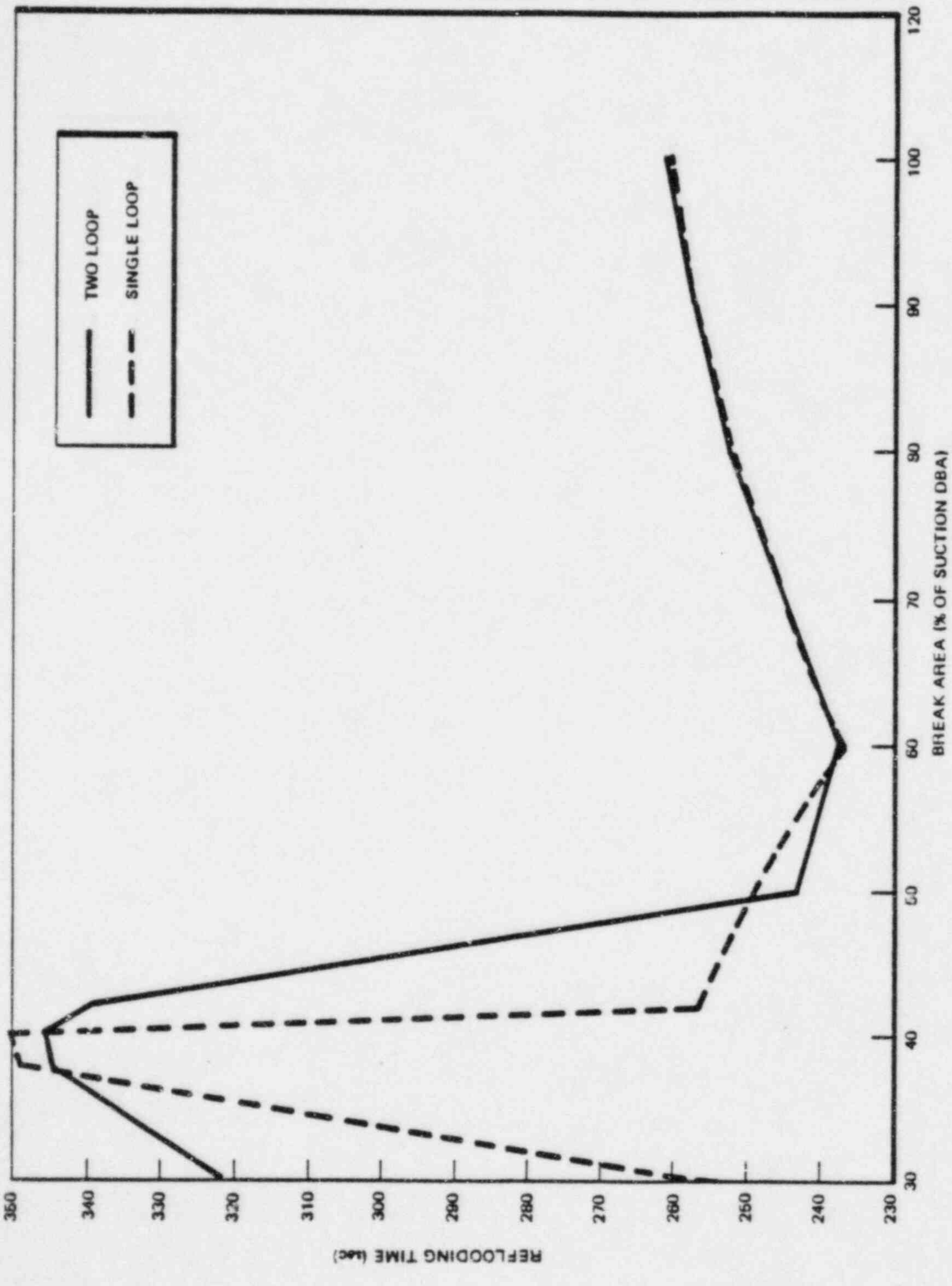

Figure 5-1. Monticello Reflooding Time vs. Break Area

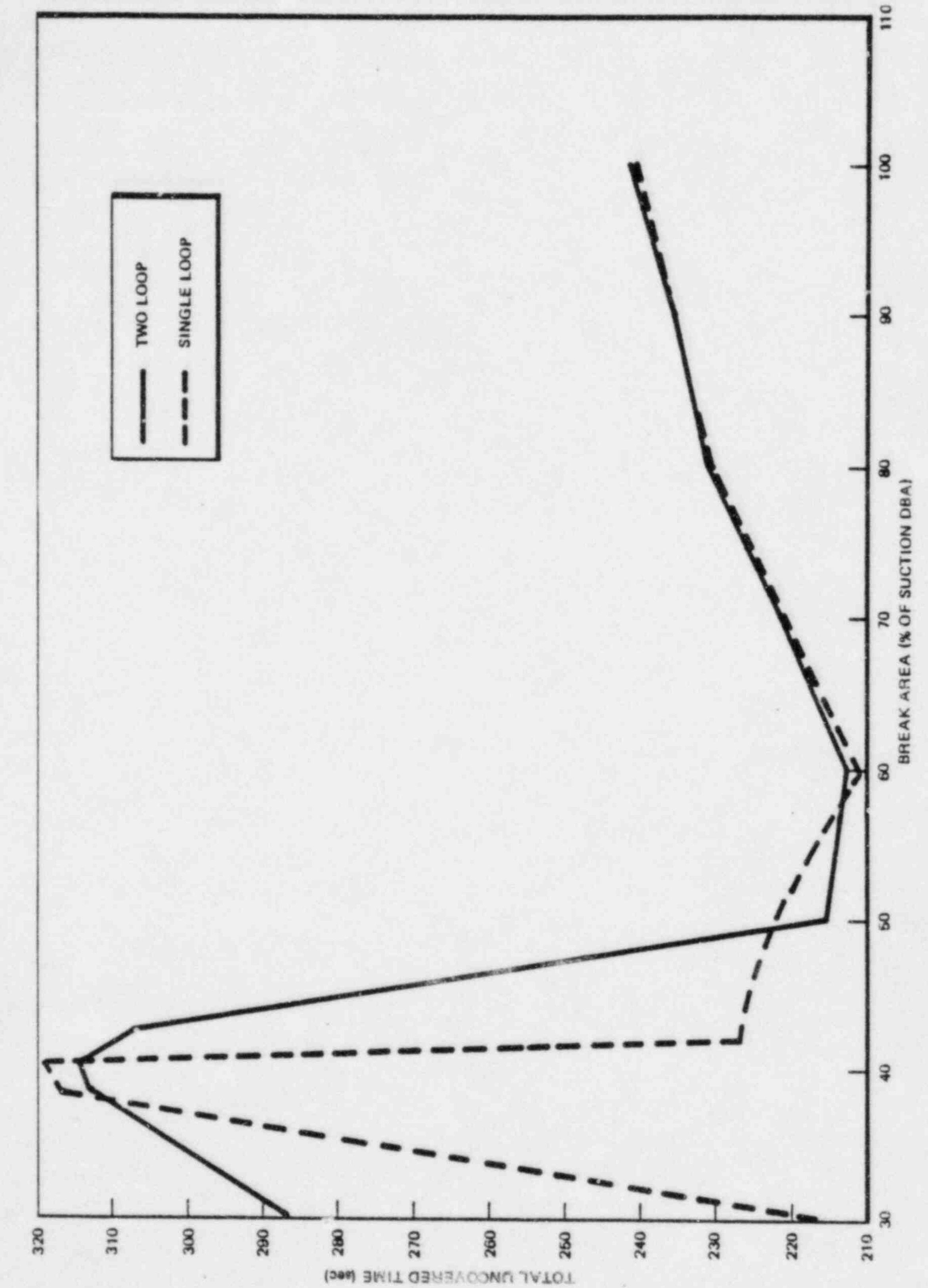

Figure 5-2. Monticello Total Uncovered Time vs. Break Area

6. REFERENCES

1. "Generic Reload Fuel Application, General Electric Company", August 1979 (NEDE-24011-P-A-1).

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

3. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K Amendment No. 2 - One Recirculation Loop Out-of-Service", General Electric Company, Revision 1, July 1978 (NEDO-20566-2).

4. Enclosure to Letter #TVA-BFNP-TS-117, O. E. Gray III to Harold R. Denton, September 15, 1978.

Northern States Power Company

414 Nicollet Mall
Minneapolis, Minnesota 55401
Telephone (612) [illegible]

October 5, 1982

Director
Office of Nuclear Reactor Regulation
U S Nuclear Regulatory Commission
Washington, DC 20555

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

[illegible]
[illegible] 7, [illegible]

On September 20, 1982, the NRC Project Manager for our Monticello Plant requested additional information related to our application for a Single Loop Operation License Amendment. The purpose of this letter is to provide the requested information.

Figure 4-1 of the Monticello Nuclear Generating Plant Single Loop Operation Report, Exhibit C of our July 2, [illegible] [illegible] shows that single loop operation is [illegible] [illegible] Figure 6 of the Supplemental Reload Licensing Submittal for Monticello Reload 9 (Cycle 10), Exhibit C of our June 25, 1982 License Amendment Request, states that natural circulation was analyzed for Cycle 10 and found acceptable. Therefore, no specific stability analysis for single loop operation for Cycle 10 was necessary.

The MAPLHGR reduction factors as listed in Revision 1 of our License Amendment Request for Single Loop Operation have been reviewed by General Electric and found applicable to Cycle 10.

David Musolf
Manager of Nuclear Support Services

DMM/SAP/js

cc: Regional Admin III, NRC
NRR Project Manager, NRC
G Charnoff
[illegible]
[illegible] J W Ferman
Resident Inspector

[illegible], 1982

Director
Office of Nuclear Reactor Regulation
U S Nuclear Regulatory Commission
Washington, DC 20555

[illegible]LLO NUCLEAR GENERATING PLANT
Docket No. 50-[illegible] License No. DPR-22

Supplemental Information for Revision No. 1 to
License Amendment [illegible] Dated September 7, 1976
[illegible] Loop Operation

On September 20, 1982, the NRC Project Manager for our Monticello Plant requested additional information related to our application for a Single Loop Operation License Amendment. The purpose of this letter is to provide the requested information.

[illegible]

[illegible] specific stability analyses for single loop operation for Cycle 10 [illegible]

The MAPLHGR reduction factors as listed in Revision 1 of our License Amendment request for Single Loop Operation have been reviewed by General Electric and found applicable to Cycle 10.

David Musolf
Manager of Nuclear Support Services

DMM/SAZ/jo

cc: Regional Admin III, NRC
NRR Project Manager, NRC
G Charnoff
MPCA
Attn: J W Ferman
Resident Inspector

[illegible]
PDR

MIDLAND 1&2-FSAR

15.5 DESIGN FEATURES

(A9)

NSP

Northern States Power Company

414 Nicollet Mall
Minneapolis, Minnesota 55401
Telephone (612) 330-5500

September 29, 1983

Director
Office of Nuclear Reactor Regulation
U S Nuclear Regulatory Commission
Washington, DC 20555

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

Single Loop Operation License Amendment
Request Rev 1 Additional Information

References 1) License Amendment Request Rev 1 submitted July 2, 1982

2) Conference call on September 26, 1983 between NRC, NSP and Lawrence Livermore Laboratories.

Two issues related to single loop operation were identified by the NRC Staff during their review of our Technical Specification change request (reference 1):

> Describe how the change from normal two recirculation cooling loop operation to one loop operation would be accomplished, with what physical and administrative controls, and while complying with branch technical position EICSB 12 regarding multiple setpoints and their control, and with IEEE STD 279-4.15.
>
> Describe changes made to the flow computer to automatically account for magnitude and sense change for reverse flow in the idle loop jet pumps during single loop operation.

These issues were discussed during a recent telephone conference call (reference 2). The purpose of this letter is to document the information provided during this call.

The Monticello technical staff will write a procedure which administratively implements the requirements of the new technical specifications. The multiple setpoints will not be used. Rather the APRM Scram and Rod Block settings would be effectively reset by gain adjustment. An independent verification of the new gain settings for single loop operation will be made by an individual with equal or greater knowledge or by the shift supervisor on the next shift.

A001
1/0

NORTHERN STATES POWER COMPANY

The APRM Scram and Rod Block flow bias is generated by circuitry which measures driving flow. The circuitry is calibrated such that during normal two loop operation, 100% drive flow equals rated core flow. However, in the case of single loop operation, the relationship of the drive flow to rated core flow is affected by the back flow through the idle jet pumps. Therefore, the APRM Scram and Rod Block settings are reduced by a conservative factor (dw=5.4) to account for the reduced flow conditions in single loop operation. With this factor applied, no further changes are required in the driving flow measurement system.

We believe this information will allow the NRC Staff to complete their review of this license amendment request. Please contact us if you have any questions related to this matter.

David Musolf

David Musolf
Manager - Nuclear Support Services

DMM/SAF/js

cc: Regional Administrator-III
NRR Project Manager, NRC
Resident Inspector, NRC
G Charnoff

(30) H. Nicolaras

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

October 5, 1983

Docket No. 50-263

Mr. D. M. Musolf
Nuclear Support Services Department
Northern States Power Company
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Dear Mr. Musolf:

The Commission has requested the Office of the Federal Register to publish the enclosed "Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for Prior Hearing." This notice is associated with your application of July 2, 1982 as supplemented on October 5, 1982. The amendment would change the Technical Specifications to incorporate revised safety and operating limits associated with the operation of Monticello Nuclear Generating Plant with one recirculation loop out of service. The proposed changes would provide for Average Power Range Monitor (APRM) flux scram trip and rod block settings, an increase in the safety limit Minimum Critical Power Ratio (MCPR) value and revisions to the allowable Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) values suitable for use with an idle recirculation loop. Presently, the Monticello Technical Specifications would require plant shutdown if an idle recirculation loop cannot be returned to service within 24 hours. The amendment would authorize the plant to operate up to 50% of rated power for extended periods of time.

Sincerely,

Helen Nicolaras

Helen Nicolaras, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosure:
Notice of Consideration

cc w/enclosure:
See next page

8310240149

Mr. D. M. Musolf
Northern States Power Company
Monticello Nuclear Generating Plant

cc:

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and
Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
Box 1200
Monticello, Minnesota 55362

Plant Manager
Monticello Nuclear Generating Plant
Northern States Power Company
Monticello, Minnesota 55362

Russell J. Hatling, Chairman
Minnesota Environmental Control
Citizens Association (MECCA)
Energy Task Force
144 Melbourne Avenue, S. E.
Minneapolis, Minnesota 55414

Executive Director
Minnesota Pollution Control Agency
1935 W. County Road B2
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Mr. Steve Gadler
2120 Carter Avenue
St. Paul, Minnesota 55108

Mr. John W. Ferman, Ph.D.
Nuclear Engineer
Minnesota Pollution Control Agency
1935 W. County Road B2
Roseville, Minnesota 55113

Commissioner of Health
Minnesota Department of Health
717 Delaware Street, S.E.
Minneapolis, Minnesota 55440

Auditor
Wright County Board of Commissioners
Buffalo, Minnesota 55313

U.S. Environmental Protection Agency
Region V Office
Regional Radiation Representative
230 South Dearborn Street
Chicago, Illinois 60604

James G. Keppler
Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT

TO FACILITY OPERATING LICENSE AND

OPPORTUNITY FOR PRIOR HEARING

The United States Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License No. DPR-22, issued to Northern States Power Company (the licensee), for operation of tthe Monticello Nuclear Generating Plant located in Wright County, Minnesota.

The amendment would revise the provisions of the Technical Specifications to incorporate revised safety and operating limits associated with the operation of Monticello Nuclear Generating Plant with one recirculation loop out of service. The changes proposed by the licensee would provide for reduced Average Power Range Monitor (APRM) flux scram trip and rod block settings, an increase in the safety limit Minimum Critical Power Ratio (MCPR) value and revisions to the allowable Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) values suitable for use with an idle recirculation loop. Presently, the Monticello Technical Specifications would require plant shutdown if an idle recirculation loop cannot be returned to service within 24 hours. The amendment would authorize the plant to operate up to 50% of rated power for extended periods of time. Supporting the amendment request, is a report prepared by General Electric that presents the analysis for core performance, in accordance with the licensee's application for amendment dated July 2, 1982 as supplemented on October 5, 1982.

Prior to issuance of the proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

By November 14, 1983 , the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written petition for leave to intervene. Request for a hearing and petitions for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR §2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition would specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to fifteen (15) days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than fifteen (15) days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter, and the bases for each contention set forth with reasonable specificity. Contentions shall be limited to matters within the scope of the amendment under consideration. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to present evidence and cross-examine witnesses.

A request for a hearing or a petition for leave to intervene shall be filed with the Secretary of the Commission, United States Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. by the above date. Where petitions are filed during the last ten (10) days of the notice period, it is requested that the petitioner or representative for the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at (800) 325-6000 (in Missouri (800) 342-6700). The Western Union operator should be given Datagram Identification Number 377 and the following message addressed to Domenic B. Vassallo: (petitioner's name and telephone number); (date petition was mailed); (plant name); and (publication date and page number of this FEDERAL REGISTER notice). A copy of the petition should also be sent to the Executive Legal Director, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and Gerald Charnoff, Esq., Shaw, Pittman, Potts and Trowbridge, 1800 M Street, N. W., Washington, D. C. 20036, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board designated to rule on the petition and/or request, that the petitioner has made a substantial showing of good cause for the granting of a late petition and/or request. That determination will be based upon a balancing of the factors specified in 10 CFR §2.714(a)(1)(i)-(v) and §2.714(d).

For further details with respect to this action, see the application for amendment dated July 2, 1982, as supplemented October 5, 1982, which is available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Environmental Conservation Library, Minneapolis Public Library, 300 Nicollet Mall, Minneapolis, Minnesota.

Dated at Bethesda, Maryland this 5th day of October, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

DBVassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

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

(31) NJHCD File-
Duane Arnold

November 8, 1983

Docket No. 50-331

Mr. Lee Liu
Chairman of the Board and
Chief Executive Officer
Iowa Electric Light and Power Company
Post Office Box 351
Cedar Rapids, Iowa 52406

Dear Mr. Liu:

The Commission has requested the Office of the Federal Register to publish the enclosed "Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for Prior Hearing." This notice relates to your application dated June 24, 1983, which would modify the operating license and Technical Specifications (TSs) for Duane Arnold Energy Center to permit unit operation up to 50% of rated thermal power with one recirculation loop out of service. The proposed license changes would delete the license condition which requires the unit to be in cold shutdown within the succeeding 24 hours if an idle recirculation loop cannot be returned to service within 24 hours. The proposed changes would also modify the TSs as necessary to provide for appropriate Average Power Range Monitor flux scram trip and rod block settings, an increase in the safety limit Minimum Critical Power Ratio value and revisions to the allowable Average Planar Linear Heat Generation Rate values suitable for use with an idle recirculation loop.

Sincerely,

Mohan C. Thadani

Mohan C. Thadani, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosure:
Notice of Consideration

cc w/enclosure:
See next page

~~8311260010~~

Mr. Lee Liu
Iowa Electric Light and Power Company
Duane Arnold Energy Center

cc:

Mr. Jack Newman, Esquire
Harold F. Reis, Esquire
Lowenstein, Newman, Reis and Axelrad
1025 Connecticut Avenue, N. W.
Washington, D. C. 20036

Office for Planning and Programming
523 East 12th Street
Des Moines, Iowa 50319

Chairman, Linn County
Board of Supervisors
Cedar Rapids, Iowa 52406

Iowa Electric Light and Power Company
ATTN: D. L. Mineck
Post Office Box 351
Cedar Rapids, Iowa 52406

U. S. Environmental Protection
Agency
Region VII Office
Regional Radiation Representative
324 East 11th Street
Kansas City, Missouri 64106

U. S. Nuclear Regulatory Commission
Resident Inspector's Office
Rural Route #1
Palo, Iowa 52324

James G. Keppler
Regional Administrator
Region III Office
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Mr. Thomas Houvenagle
Regulatory Engineer
Iowa Commerce Commission
Lucas State Office Building
Des Moines, Iowa 50319

UNITED STATES NUCLEAR REGULATORY COMMISSION

IOWA ELECTRIC LIGHT AND POWER COMPANY

DOCKET NO. 50-331

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE AND OPPORTUNITY FOR PRIOR HEARING

The United States Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License No. DPR-49, issued to Iowa Electric Light and Power Company (the licensee), for the operation of the Duane Arnold Energy Center (DAEC) located in Linn County, Iowa.

The amendment proposed by the licensee would revise the operating license and the provisions in the Technical Specifications relating to changes to permit reactor operation at power levels up to 50% of rated thermal power with one recirculation loop out of service. Presently, DAEC operating license requires a unit to be in cold shutdown within the succeeding 24 hours if an idle recirculation loop cannot be returned to service within 24 hours. The change proposed by the licensee would delete this license condition and modify the Technical Specifications (TSs) as necessary to provide for appropriate Average Power Range Monitor (APRM) flux scram trip and rod block settings, an increase in the safety limit Minimum Critical Power Ratio (MCPR) value and revisions to the allowable Average Planar Linear Heat Generation Rate (APLHGR) values suitable for use with an idle recirculation loop, in accordance with the licensee's application for amendment dated June 24, 1983.

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Prior to issuance of the proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

By December 16, 1983 the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written petition for leave to intervene. Request for a hearing and petitions for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR §2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition would specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to fifteen (15) days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than fifteen (15) days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter, and the bases for each contention set forth with reasonable specificity. Contentions shall be limited to matters within the scope of the amendment under consideration. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to present evidence and cross-examine witnesses.

A request for a hearing or a petition for leave to intervene shall be filed with the Secretary of the Commission, United States Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. by the above date. Where petitions are filed during the last ten (10) days of the notice period, it is requested that the petitioner or representative for the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at (800) 325-6000 (in Missouri (800) 342-6700). The Western Union operator should be given Datagram Identification Number 377 and the following message addressed to Domenic B. Vassallo: (petitioner's name and telephone number); (date petition was mailed); (plant name); and (publication date and page number of this FEDERAL REGISTER notice). A copy of the petition should also be sent to the Executive Legal Director, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, and Jack Newman, Esquire, Harold F. Reis, Esquire, Lowenstein, Newman, Reis and Axelrad, 1025 Connecticut Avenue, N. W., Washington, D. C. 20036, attorneys for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board designated to rule on the petition and/or request, that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR §2.714(a)(1)(i)-(v) and §2.714(d).

For further details with respect to this action, see the application for amendment dated June 24, 1983, which is available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and the Cedar Rapids Public Library, 426, Third Avenue, S. E., Cedar Rapids, Iowa 52401.

Dated at Bethesda, Maryland this 8th day of November, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

DBVassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

(B2) [illegible handwriting]

Iowa Electric Light and Power Company

October 17, 1980
LDR-80-277

LARRY D. ROOT
ASSISTANT VICE PRESIDENT
OF NUCLEAR DIVISION

Mr. Harold Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton:

Transmitted herewith, in accordance with requirements of 10CFR50.59 and 50.90 is an application for amendment to Appendix A (Technical Specification) to operating license DPR-49 for the Duane Arnold Energy Center (DAEC), which provides for single recirculation loop operation of the DAEC. This application supplements application RTS-74, submitted January 12, 1977.

This application has been reviewed by the DAEC Operations Committee and the DAEC Safety Committee.

In accordance with 10CFR50.30, three signed and 37 additional copies of this application are transmitted herewith. This application, consisting of the foregoing letter and enclosures, is true and accurate to the best of my knowledge and belief.

IOWA ELECTRIC LIGHT AND POWER COMPANY

BY: Larry D. Root
Larry D. Root
Assistant Vice President
Nuclear Generation

Subscribed to and Sworn to Before Me
this 17 day of October
19 80.

Mary E. Benfield
Notary Public in and For The
State of Iowa

MARY E. BENFIELD
COMMISSION EXPIRING
SEPTEMBER 30, 1983

LDR/YB/ld
Enclosure
cc: Y. Balas
D. Arnold
L. Liu
S. Tuthill
K. Meyer
D. Mineck
J. Van Sickel
NRC Resident Office
K. Eccleston (NRC)
File: A-117

8010240220

PROPOSED CHANGE RTS 124 TO

THE DUANE ARNOLD ENERGY CENTER

TECHNICAL SPECIFICATIONS

The holders of license DPR-49 for the Duane Arnold Energy Center propose to amend Appendix A (Technical Specifications) to said license by deleting current pages and replacing them with the attached, new pages. A list of the affected pages is included.

The current DAEC Technical Specifications do not allow plant operation beyond 24 hours if an idle recirculation loop cannot be returned to service. The ability to operate at reduced power with a single loop is highly desirable from availability/outage planning standpoint in the event that maintenance or component inavailability renders one loop inoperable. Such events have occurred three times during the current cycle and have caused the licensee to apply for temporary amendments, sometimes on an emergency basis. Therefore, the holders of this license propose that the Technical Specifications be revised as indicated in the attached pages to allow single-loop operation. Supporting analysis is given in NEDO-24272, which is reference 11 on P. 3.12-11 and a copy is enclosed.

The other change consists of renumbering pages to delete blank pages which were created in previous changes to the Technical Specifications.

~~8610240225~~

<u>Affected Pages</u>

1.1-1	1.1-24*
1.1-2	1.1-25*
1.1-3	1.1-26*
1.1-5	1.1-27*
1.1-6	1.1-28*
1.1-7	3.2-16
1.1-8	3.6-7
1.1-9	3.6-29
1.1-10	3.12-1
1.1-11	3.12-3
1.1-12	3.12-4
1.1-13	3.12-5a*
1.1-14	3.12-6
1.1-15	3.12-7
1.1-16	3.12-8
1.1-17	3.12-9
1.1-18	3.12-11
1.1-19*	3.12-13
1.1-20*	3.12-14
1.1-21*	3.12-15
1.1-22*	3.12-16
1.1-23*	3.12-17
	3.12-5

* These pages have been deleted.

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the inter-related variables associated with fuel thermal behavior.

Objective:

To establish limits which ensure the integrity of the fuel cladding.

Specifications:

A. Reactor Pressure > 785 psig and Core Flow > 10% of Rated.

The existence of a minimum critical power ratio (MCPR) less than 1.07 for two recirculation loop operation (1.08 for single-loop operation) shall constitute violation of the fuel cladding integrity safety limit.

B. Core Thermal Power Limit (Reactor Pressure < 785 psig or Core Flow < 10% of Rated

When the reactor pressure is ≤ 785 psig or core flow is less than 10% of rated, the core thermal power shall not exceed 25 percent of rated thermal power.

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

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:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trips

1. APRM High Flux Scram When In Run Mode.

For operation with the fraction of rated power (FRP) greater than or equal to the maximum fraction of limiting power density (MFLPD), the APRM scram trip setpoint shall be as shown on Fig. 2.1-1 and shall be:

$S \leq (0.66W + 54)$

with a maximum setpoint of 120% rated power at 100% rated recirculation flow or greater.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

16.C Power Transient

To ensure that the Safety Limits established in Specification 1.1.A and 1.1.B are not exceeded, each required scram shall be initiated by its primary source signal. A Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the Primary Source Signal.

D. With irradiated fuel in the reactor vessel, the water level shall not be less than 12 in. above the top of the normal active fuel zone. Top of the active fuel zone is defined to be 344.5 inches above vessel zero (See Bases 3.2)

Where: S = Setting in percent of rated power (1,593 MWt)

W = Recirculation loop flow in percent of rated flow. Rated recirculation loop flow is that recirculation loop flow which corresponds to 49×10^6 lb/hr core flow.

For a MFLPD greater than FRP, the APRM scram setpoint shall be:

$S \leq (0.66W + 54) \frac{FRP}{MFLPD}$ for two recirculation loop operation and

$S \leq (0.66W + 50.7) \frac{FRP}{MFLPD}$ for one recirculation loop operation.

NOTE: These settings assume operation within the basic thermal design criteria. These criteria are LHGR $\leq$ 18.5 KW/ft (7x7 array) or 13.4 KW/ft (8x8 array) and MCPR $\geq$ values as indicated in Table 3.12-2 times K_f, where K_f is defined by Figure 3.12-1. Therefore, at full power, operation is not allowed with MFLPD greater than unity even if the scram setting is reduced. If it is determined that either of these design criteria is being violated during operation, action must be taken immediately to return to operation within these criteria.

2. APRM High Flux Scram

When in the REFUEL or STARTUP and HOT STANDBY MODE. The APRM scram shall be set at less than or equal to 15 percent of rated power.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
	3. APRM Rod Block When in Run Mode. For operation with MFLPD less than or equal to FRP the APRM Control Rod Block setpoint shall be as shown on Fig. 2.1-1 and shall be: $S \leq (0.66W + 42)$ The definitions used above for the APRM scram trip apply. For a MFLPD greater than FRP, the APRM Control Rod Block setpoint shall be: $S \leq (0.66W + 42)\frac{FRP}{MFLPD}$ for two recirculation loop operation, and $S \leq (0.66W + 38.7)\frac{FRP}{MFLPD}$ for one recirculation loop operation. 4. IRM - the IRM scram shall be set at less than or equal to 120/125 of full scale. B. Scram and Isolation on reactor low water level $\geq$ 513.5 inches above vessel zero (+12" on level instruments) C. Scram - turbine stop valve closure $\leq$ 10 percent valve closure D. Turbine control valve fast closure shall occur within 30 milliseconds of the start of turbine control valve fast closure.

1.1 BASES: FUEL CLADDING INTEGRITY

A. Fuel Cladding Integrity Limit at Reactor Pressure ≥ 785 psig and Core Flow ≥ 10% of Rated

The fuel cladding integrity safety limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedure used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore the fuel cladding integrity safety limit is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is generically determined in Reference 1, for two recirculation loop operation. This safety limit MCPR is increased by 0.01 for single-loop operation as discussed in Reference 2.

B. Core Thermal Power Limit (Reactor Pressure $\leq$ 785 psig or Core Flow $\leq$ 10% of Rated)

At pressures below 785 psig, the core evaluation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low power and all 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 power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of 28 x 10^3lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28 x 10^3lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia or core flow less than 10% is conservative.

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 1.1.A or 1.1.B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following close of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

The computer provided with Duane Arnold has a sequence annunciation program which will indicate the sequence in which events such as scram, APRM trip initiation, pressure scram initiation, etc., occur. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 1.1.C will be relied on to determine if a Safety Limit has been violated.

D. Reactor Water Level (Shutdown Condition)

During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the fuel* provides adequate margin. This level will be continuously monitored.

*Top of the active fuel zone is defined to be 344.5 inches above vessel zero (See Bases 3.2).

1.1 REFERENCES

1. "Generic Reload Fuel Application," NEDE-24011-P-A and NEDO-24011-A.*

2. "Duane Arnold Energy Center Single-Loop Operation," NEDO-24272 July 1980.

*Approved Revision at time reload analyses are performed.

2.1 BASES: LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Duane Arnold Energy Center have been analyzed throughout the spectrum of planned operating conditions up to the thermal power condition of 1658 MWt. The analyses were based upon plant operation in accordance with the operating map given in Figure 3.7-1 of the FSAR. In addition, 1658 MWt is the licensed maximum power level of the Duane Arnold Energy Center, and this represents the maximum steady state power which shall not knowingly be exceeded.

Transient analyses performed each reload are given in Reference 1. Models and model conservatisms are also described in this reference. As discussed in Reference 2, the core wide transient analyses for one recirculation pump operation is conservatively bounded by two-loop operation analyses and the flow-dependent rod block and scram setpoint equations are adjusted for one-pump operation.

Steady-state operation without forced recirculation will not be permitted, except during special testing. The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps.

Trip Settings

The bases for individual trip settings are discussed in the following paragraphs.

A. Neutron Flux Trips

1. APRM High Flux Scram (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (1593 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin from fuel damage. Therefore, the use of flow referenced scram trip provides even additional margin. An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MFLPD and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1, when the maximum fraction of limiting power density is greater than the fraction of rated power.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR greater than or equal to safety limit when the transient is initiated from MCPR $\geq$ values as indicated in Table 3.12.2.

2. <u>APRM High Flux Scram (Refuel or Startup & Hot Standby Mode)</u>

For operation in these modes the APRM scram setting of 15 percent of rated power and the IRM High Flux Scram provide 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 and the Rod Sequence Control System. Worths of individual rods are very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise.

Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 880 psig.

3. APRM Rod Block (Run Mode)

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 rod withdrawal beyond a given power level at constant recirculation flow rate, and thus prevents a MCPR less than safety limit. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents excessive reactor power level increase resulting from 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 worst case MCPR which could occur during steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. 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 if the maximum fraction of limiting power density exceeds the fraction of rated power, thus preserving the APRM rod block safety margin. As with the scram setting, this may be accomplished by adjusting the APRM gain.

4. IRM

The IRM system consists of 6 chambers, 3 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods that the heat flux is in equilibrium with the neutron flux, and an IRM scram would result in a reactor shutdown well before any Safety Limit is exceeded.

In order to ensure that the IRM provides adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents has been analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is by-passed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above safety limit. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

B. Scram and Isolation on Reactor Low Water Level

The setpoint for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. Analyses show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than safety limit in all cases, and system pressure does not reach the safety valve settings. The scram setting is approximately 21 inches below the normal operating range and is thus adequate to avoid spurious scrams.

C. Scram - Turbine Stop Valve Closure

The turbine stop-valve closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from rapid closure of the turbine stop valves.

With a scram setting at 10 percent of valve closure, the resultant increase in surface heat flux is such that MCPR remains above safety limit even during the worst case transient that assumes the turbine bypass is closed. This scram is by-passed when turbine steam flow is below 30 percent of rated, as measured by the turbine first stage pressure.

D. Turbine Control Valve Fast Closure (Loss of Control Oil Pressure) Scram

The control valve fast closure scram is provided to limit the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection. It prevents MCPR from becoming less than safety limit for this transient.

E. F. and J. Main Steam Line Isolation on Low Pressure, Low Condenser Vacuum, and Main Steam Line Isolation Scram

The low pressure isolation of the main steam lines at 880 psig has been provided to protect against rapid reactor depressurization. To protect the main condenser against over-pressure, a loss of condenser vacuum initiates automatic closure of the main steam isolation valves.

G. H. and I. Reactor Low Water Level Setpoint for Initiation of HPCI and RCIC, Closing Main Steam Isolation Valves, and Starting LPCI and Core Spray Pumps

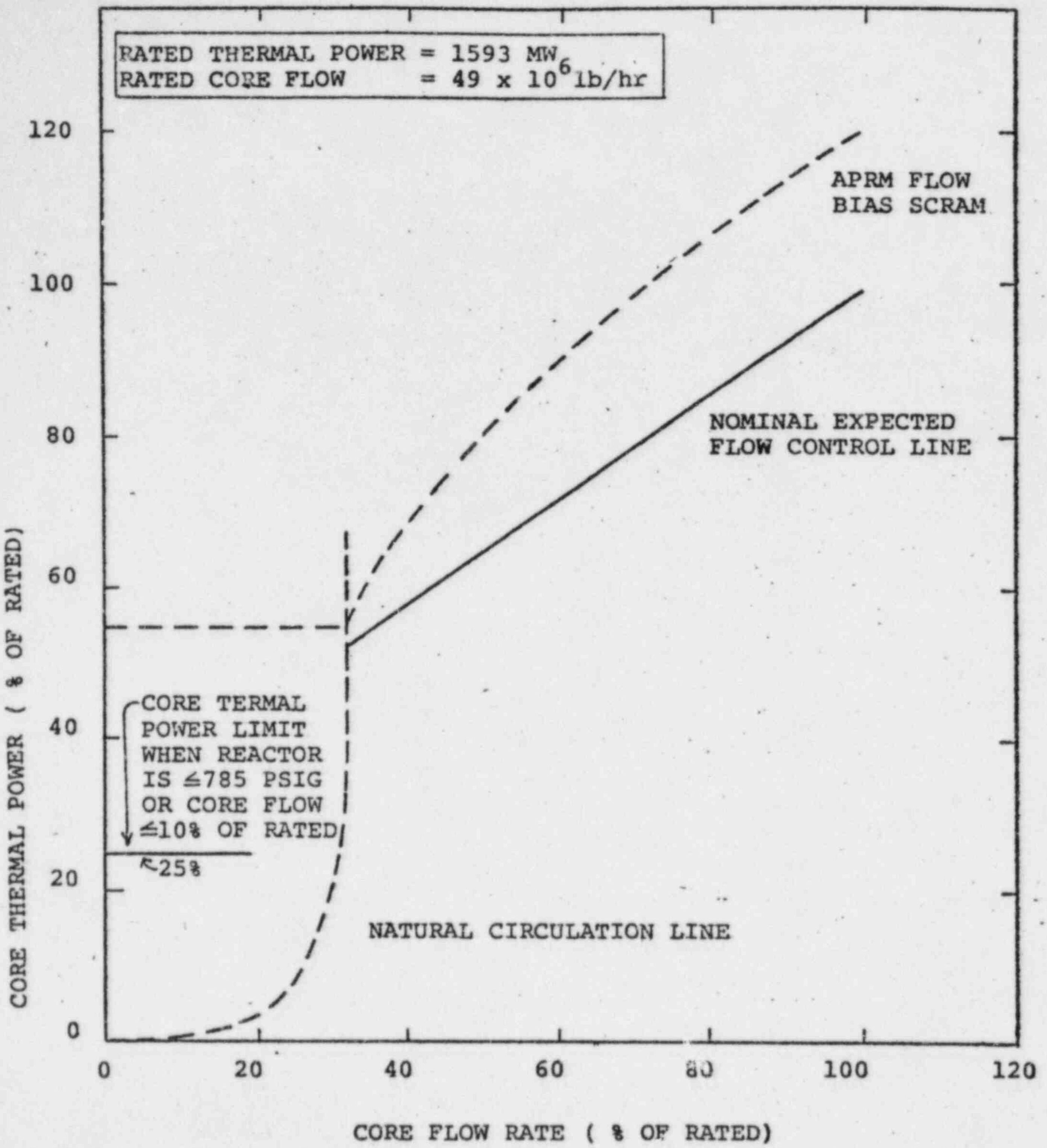
These systems maintain adequate coolant inventory and provide core cooling with the objective of preventing excessive clad temperatures. The design of these systems to adequately perform the intended function is based on the specified low level scram setpoint and initiation setpoints. Transient analyses demonstrate that these conditions result in adequate safety margins for both the fuel and the system pressure.

2.1 REFERENCES

1. "Generic Reload Fuel Application," NEDE-24011-P-A* or NEDO-24011-A.

2. "Duane Arnold Energy Center Single-Loop Operation," NEDO-24272 July 1980.

*Approved revision number at time analyses are performed.



DUANE ARNOLD ENERGY CENTER IOWA ELECTRIC LIGHT & POWER COMPANY TECHNICAL SPECIFICATIONS
APRM FLOW BIAS SCRAM RELATIONSHIP TO NORMAL OPERATING CONDITIONS FIGURE 1.1-1

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Table 3.2-C

Minimum No. of Operable instrument channels Per Trip System	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
2	APRM Upscale (Flow Biased)	for 2 recirc loop operation $\leq(0.66W + 42)\frac{FRP}{MFLPD}$ (2) for 1 recirc loop operation $\leq(0.66W + 38.7)\frac{FRP}{MFLPD}$ (2)	6 Inst. Channels	(1)
2	APRM Upscale (Not in Run Mode)	$\leq$12 indicated on scale	6 Inst. Channels	(1)
2	APRM Downscale	$\geq$5 indicated on scale	6 Inst. Channels	(1)
1 (7)	Rod Block Monitor (Flow Biased)	for 2 recirc loop operation $\leq(0.66W + 39)\frac{FRP}{MFLPD}$ (2) for 1 recirc loop operation $\leq(0.66W + 35.7)\frac{FRP}{MFLPD}$ (2)	2 Inst. Channels	(1)
1 (7)	Rod Block Monitor Downscale	$\geq$5 indicated on scale	2 Inst. Channels	(1)
2	IRM Downscale (3)	$\geq$5/125 full scale	6 Inst. Channels	(1)
2	IRM Detector not in Startup Position	(8)	6 Inst. Channels	(1)
2	IRM Upscale	$\leq$108/125	6 Inst. Channels	(1)
2 (5)	SRM Detector not in Startup Position	(4)	4 Inst. Channels	(1)
2 (5) (6)	SRM Upscale	$\leq 10^5$ counts/sec.	4 Inst. Channels	(1)

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.
	c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.
	2. Whenever there is recirculation flow with the reactor in the Startup or Run mode, and one recirculation pump is operating, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.
F. Jet Pump Flow Mismatch	F. Jet Pump Flow Mismatch
1. When both recirculation pumps are in steady state operation, the speed of the faster pump may not exceed 122% of the speed of the slower pump when core power is 80% or more of rated power or 135% of the speed of the slower pump when core power is below 80% of rated power.	1. Recirculation pump speeds shall be checked and logged at least once per day.
2. If specification 3.6.F.1 cannot be met, one recirculation pump shall be tripped. The reactor may be started and operated with one recirculation loop out of service provided that: a. MAPLHGR multipliers as indicated in section 3.12A are applied. b. The power level is limited to maximum of 82% of licensed power. c. The idle loop is isolated prior to startup, or if disabled during reactor operation, within 24 hours (suction valve closed and electrically disconnected).	

80% power cases, respectively. If the reactor is operating on one pump, the loop select logic trips that pump before making the loop selection.

An evaluation has been provided for ECCS performance during reactor operation with one recirculation loop out of service (Sec. 3.12, Ref. 4). Therefore, continuous operation under such conditions is appropriate. The reactor may in any case be operated up to 24 hours with one recirculation loop out of service without isolating the idle loop. This short period of time permits corrective action to be taken to re-activate the idle loop or to implement the changes for continuous operation with one recirculation loop out of service.

Requiring the discharge valve of the lower speed loop to remain closed until the speed of faster pump is below 50% of its rated speed provides assurance when going from one to two pump operation that excessive vibration of the jet pump risers will not occur.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.12 CORE THERMAL LIMITS

Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A.. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

During reactor power operation, the actual MAPLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figs. 3.12-2, 3, 4, 5, 6, and 7. For single-loop operation, the values in these curves are reduced by multiplying by 0.86, 0.87 and 0.87 for 7x7, 8x8 and 8x8R fuel, respectively. If at any time during reactor power operation it is determined by normal surveillance that the limiting value for MAPLHGR (LAPLHGR) is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the MAPLHGR (LAPLHGR) is not returned to within the prescribed limits within two hours, the reactor shall be brought to the cold shutdown condition within 36 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

4.12 CORE THERMAL LIMITS

Applicability

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

The MAPLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
C. Minimum Critical Power Ratio (MCPR)	C. Minimum Critical Power Ratio (MCPR)
During reactor power operations, MCPR for two recirculation loop operation shall be ≥ values as indicated in Table 3.12-2 at rated power and flow. If at any time during reactor power operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the operating MCPR is not returned to within the prescribed limits within two hours, the reactor shall be brought to the cold shutdown condition within 36 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.	MCPR shall be determined daily during reactor power operation at ≥ 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 3.3.2.
For core flows other than rated the MCPR shall be ≥ values as indicated in Table 3.12-2 times K_f, where K_f is shown in Figure 3.12-1.	
For one recirculation loop operation the MCPR limits at rated flow are 0.01 higher than the comparable two-loop values.	
D. Reporting Requirements	
If any of the limiting values identified in Specifications 3.12.A, B or C are exceeded, a Reportable Occurrence report shall be submitted. If the corrective action is taken, as described, a thirty-day written report will meet the requirements of this specification.	

3.12 BASES: CORE THERMAL LIMITS

A. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR Part 50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm$ 20^{o}F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR Part 50, Appendix K limit.

For two recirculation loop operation the calculational procedure used to establish the MAPLHGR's shown on Figures 3.12-2 to 3.12-6, are documented in Reference 2.

Reduction factors for one recirculation loop operation were derived in Reference 4.

B. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation rate and that the fuel cladding 1% plastic diametral strain linear heat generation rate is not exceeded during any abnormal operating transient if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Reference 2 and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at ≥ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

C. Minimum Critical Power Ratio (MCPR)

1. Operating Limit MCPR

The required operating limit MCPR's at steady state operating conditions as specified in Specification 3.12.C are derived from the established fuel cladding integrity Safety Limit MCPR value, and an analysis of abnormal operational transients[1]. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip settings given in Specification 2.1.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in critical power ratio (CPR).

2. MCPR Limits for Core Flows Other than Rated Flow

The purpose of the K_f factor is to define operating limits at other than rated flow conditions. At less than 100% flow the required MCPR is the product of the operating limit MCPR and the K_f factor. Specifically, the K_f factor provides the required thermal margin to protect against a flow increase transient. The most limiting transient initiated from less than rated flow conditions is the recirculation pump speed up caused by a motor-generator speed control failure.

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR of values as indicated in Table 3.12-2 will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated for the same postulated transient event.

The K_f factor curves shown in Figure 3.12-1 were developed generically and are applicable to all BWR/2, BWR/3 and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to rated thermal power at rated core flow, as described in Reference 2.

The K_f factors shown in Figure 3.12-1 are conservative for Duane Arnold operation because the operating limit MCPR of values as indicated in Table 3.12-2 is greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

D. Reporting Requirements

The Limiting Conditions for Operation associated with monitoring the fuel rod operating conditions are required to be met at all times, i.e., there is no allowable time in which the plant can knowingly exceed the limiting values of MAPLHGR, LHGR and MCPR. It is a requirement, as stated in Specifications 3.12.A, B and C that if at any time during reactor power operation, it is determined that the limiting values for MAPLHGR, LHGR or MCPR are exceeded, action is then initiated to restore operation to within the prescribed limits. This action is initiated as soon as normal surveillance indicates that an operating limit has been reached. Each event involving operation beyond a specified limit shall be reported as a Reportable Occurrence. If the specified corrective action described in the LCO's was taken , a thirty-day written report is acceptable.

TABLE 3.12-2

MCPR LIMITS

Fuel Type	
7 x 7	1.25
8 x 8	1.24
8 x 8R	1.26

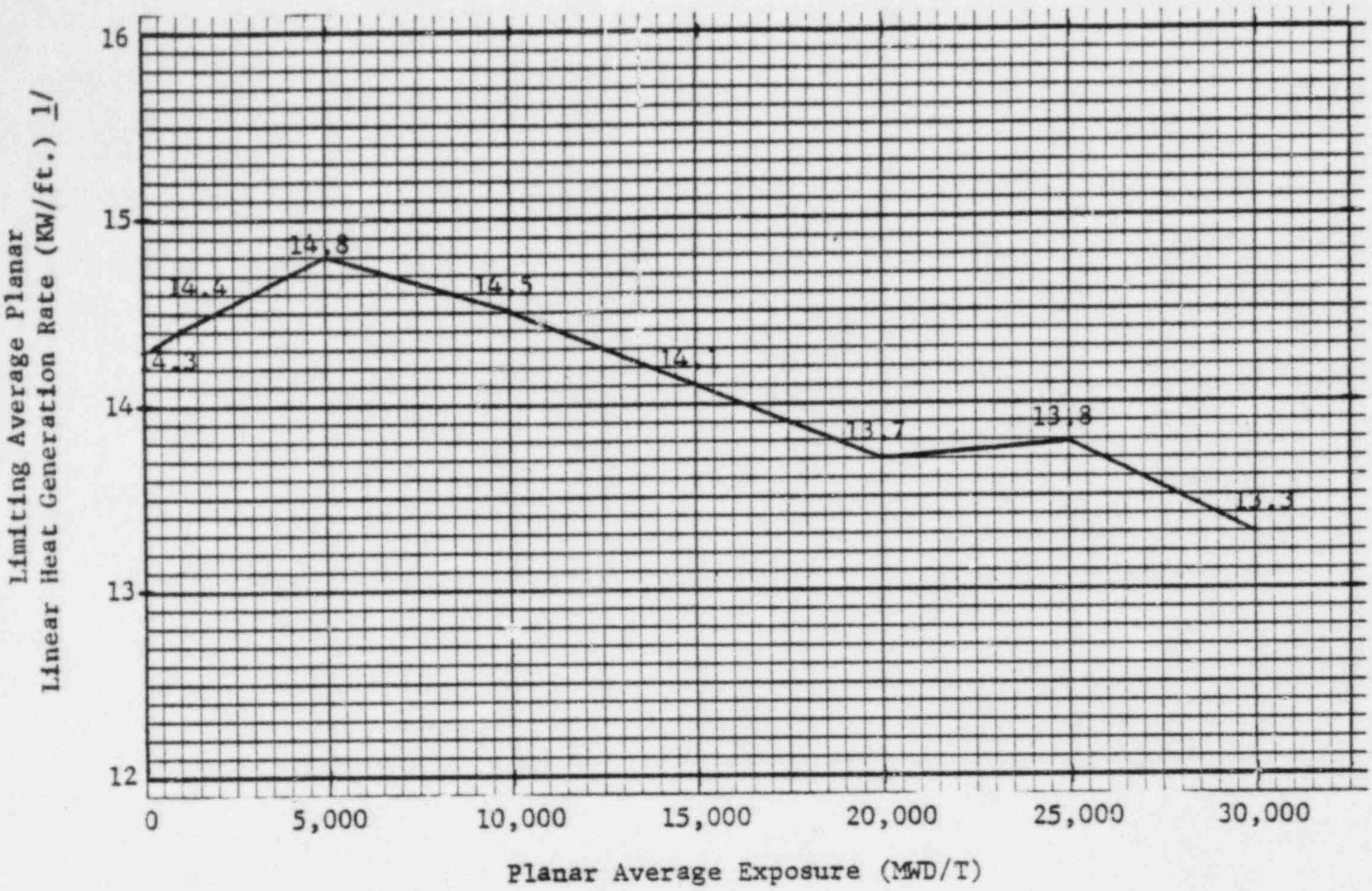
3.12 REFERENCES

1. Duane Arnold Energy Center Loss-of-Coolant Accident Analysis Report, NEDO-21082-02-1A, Class I, July 1977, Appendix A.

2. "Generic Reload Fuel Application," NEDE-24011-P-A**.

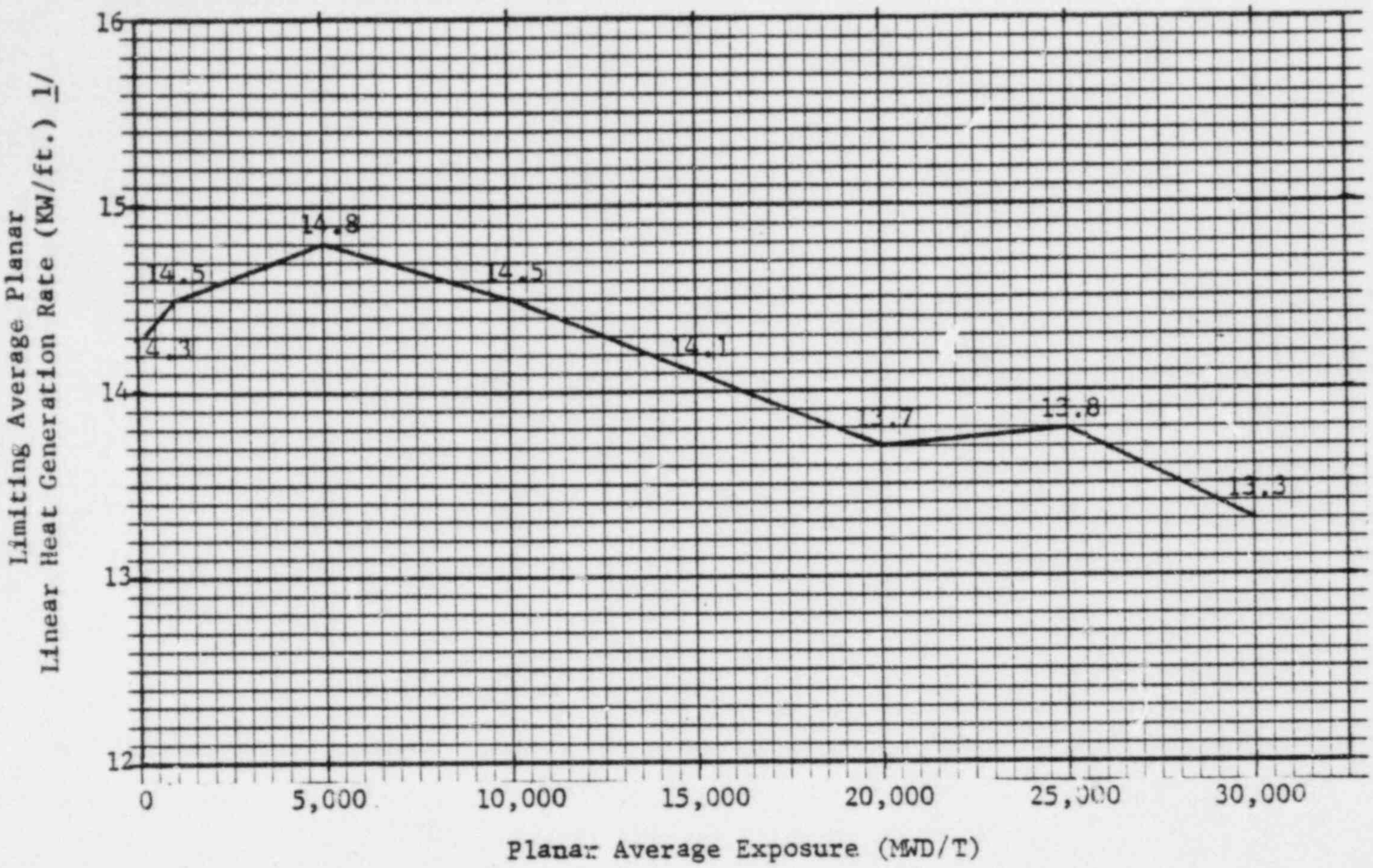
3. Current Reload Submittal for Duane Arnold Energy Center.

4. "Duane Arnold Energy Center Single Loop Operation", NEDO-24272 July 1980.



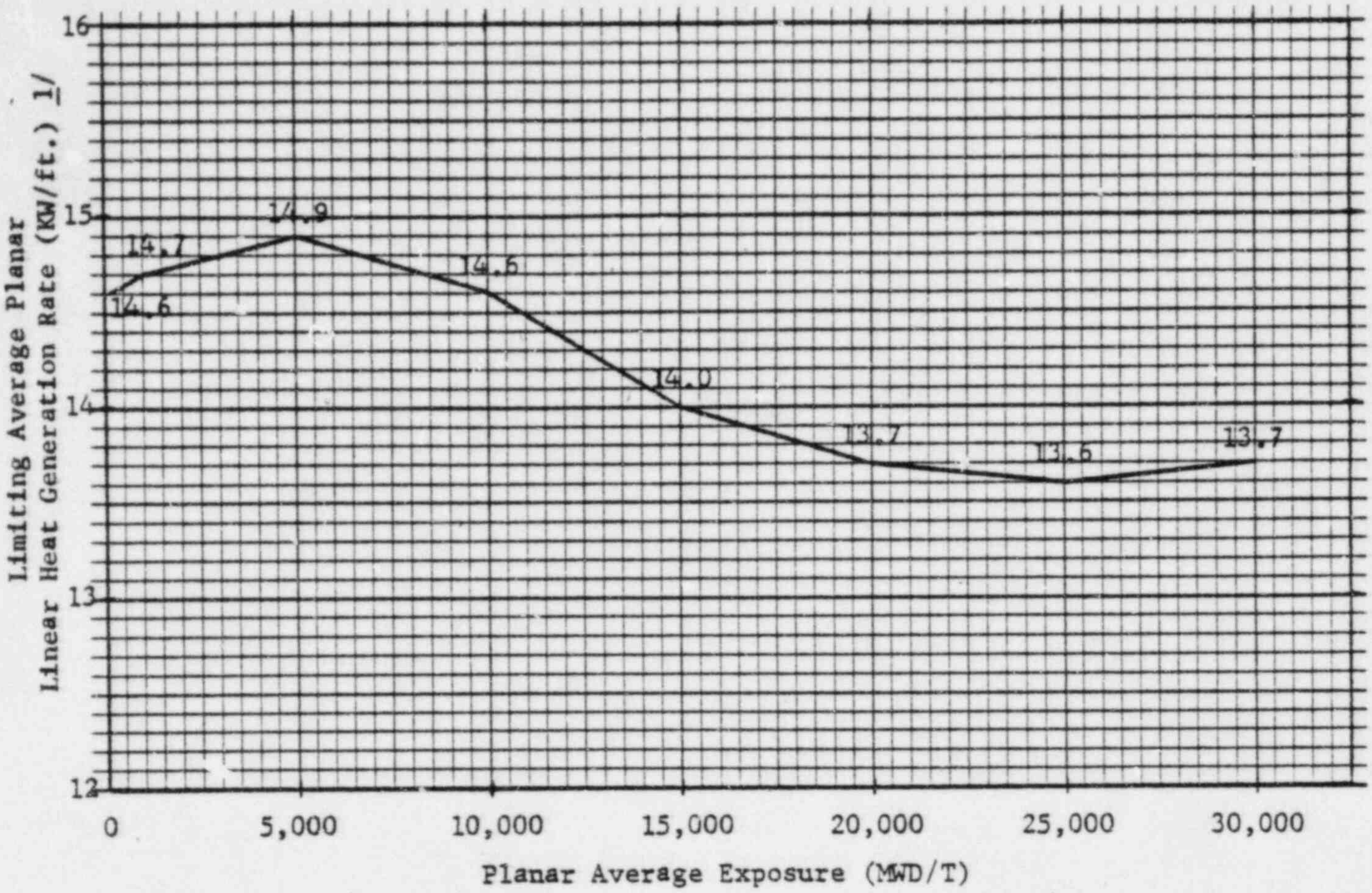
1/ When core flow is equal to or less than 70% of rated, the MAPLHGR shall not exceed 95% of the limiting values shown. Values shown are for two recirculation loops. Reduction factors for one recirculation loop were derived in Reference 4.

DUANE ARNOLD ENERGY CENTER IOWA ELECTRIC LIGHT AND POWER COMPANY TECHNICAL SPECIFICATIONS
LIMITING AVERAGE PLANAR LINEAR HEAT GENERATION RATE AS A FUNCTION OF PLANAR AVERAGE EXPOSURE FUEL TYPE: INITIAL CORE TYPE 2 FIGURE 3.12-2



1/ When core flow is equal to or less than 70% of rated, the MAPLHGR shall not exceed 95% of the limiting values shown. Values shown are for two recirculation loops. Reduction factors for one recirculation loop were derived in Reference 4.

DUANE ARNOLD ENERGY CENTER IOWA ELECTRIC LIGHT AND POWER COMPANY TECHNICAL SPECIFICATIONS
LIMITING AVERAGE PLANAR LINEAR HEAT GENERATION RATE AS A FUNCTION OF PLANAR AVERAGE EXPOSURE FUEL TYPE: INITIAL CORE TYPE 3 FIGURE 3.12-3



1/ When core flow is equal to or less than 70% of rated, the MAPLHGR shall not exceed 95% of the limiting values shown. Values shown are for two recirculation loops. Reduction factors for one recirculation loop were derived in Reference 4.

DUANE ARNOLD ENERGY CENTER

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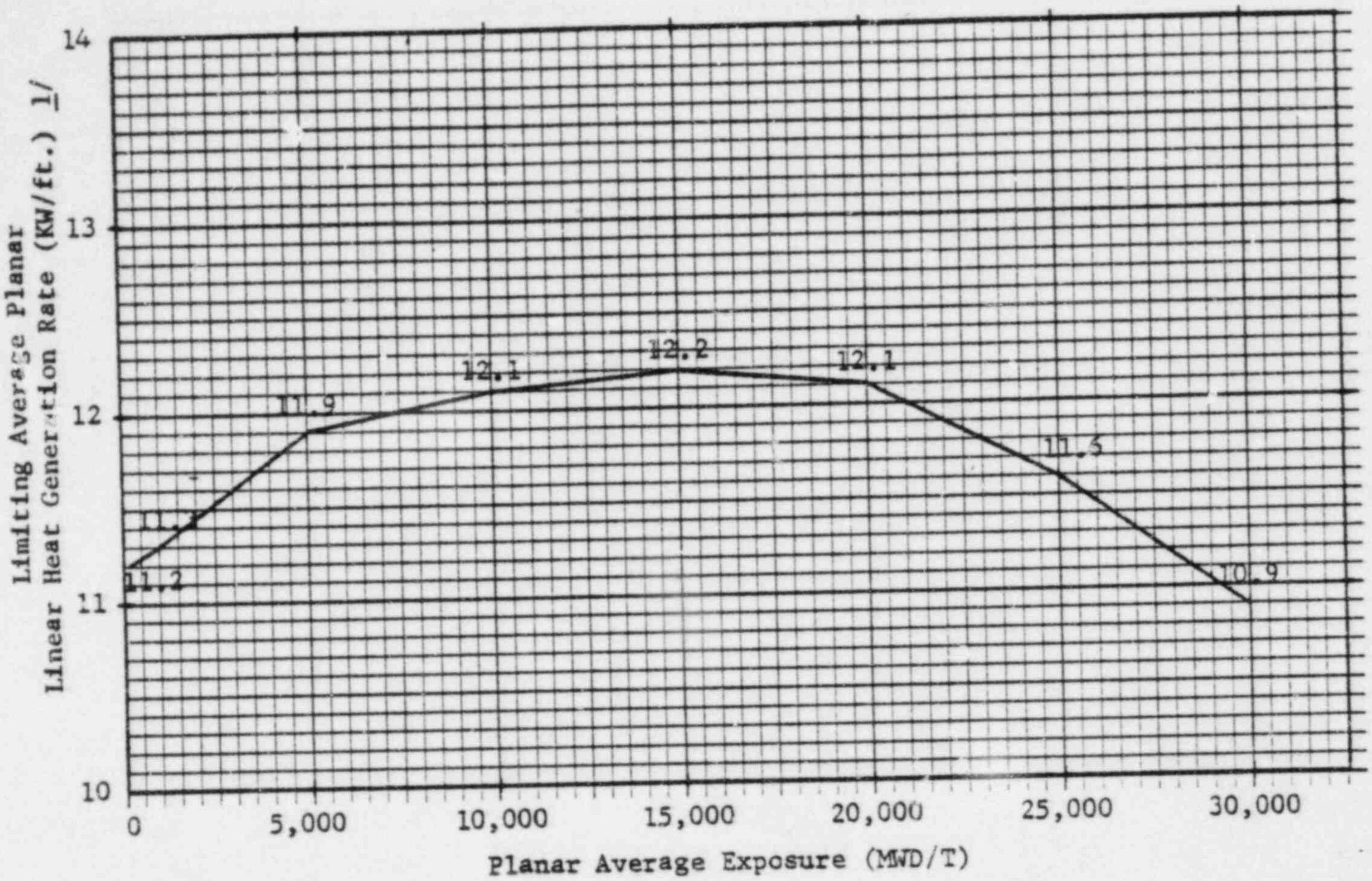
TECHNICAL SPECIFICATIONS

LIMITING AVERAGE PLANAR LINEAR HEAT GENERATION RATE AS A FUNCTION OF PLANAR AVERAGE EXPOSURE

FUEL TYPE: 7D230 TYPE 4

FIGURE 3.12-4



1/ When core flow is equal to or less than 70% of rated, the MAPLHGR shall not exceed 95% of the limiting values shown. Values shown are for two recirculation loops. Reduction factors for one recirculation loop were derived in Reference 4.

DUANE ARNOLD ENERGY CENTER

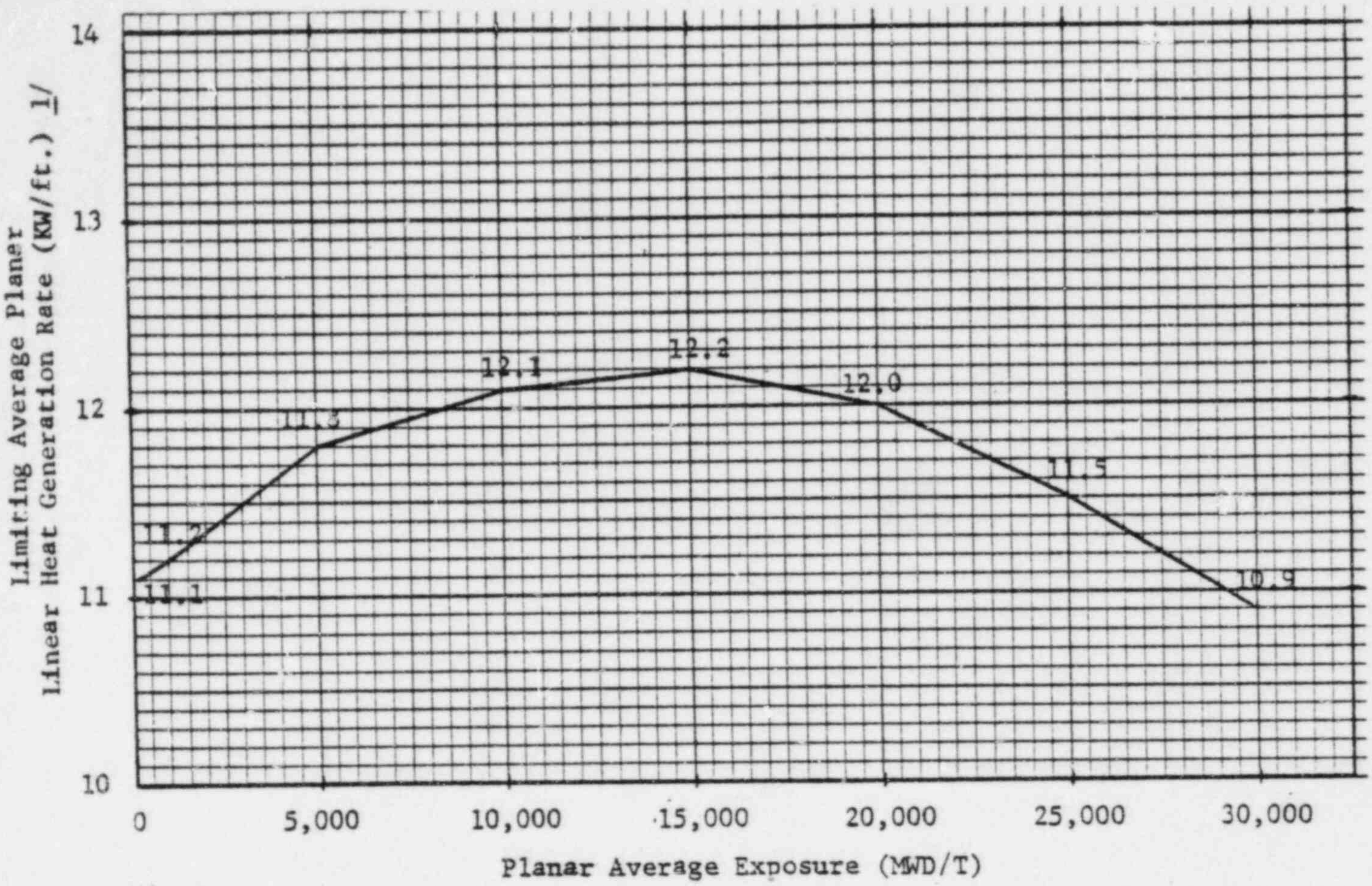
IOWA ELECTRIC LIGHT AND POWER COMPANY

TECHNICAL SPECIFICATIONS

LIMITING AVERAGE PLANAR LINEAR HEAT GENERATION RATE AS A FUNCTION OF PLANAR AVERAGE EXPOSURE

FUEL TYPE: 8D274L

FIGURE 3.12-5



1/ When core flow is equal to or less than 70% of rated, the MAPLHGR shall not exceed 95% of the limiting values shown. Values shown are for two recirculation loops. Reduction factors for one recirculation loop were derived in Reference 4.

DUANE ARNOLD ENERGY CENTER

IOWA ELECTRIC LIGHT AND POWER COMPANY

TECHNICAL SPECIFICATIONS

LIMITING AVERAGE PLANAR LINEAR HEAT GENERATION RATE AS A FUNCTION OF PLANAR AVERAGE EXPOSURE

FUEL TYPE: 8D274H

FIGURE 3.12-6

(38) 50-331

Iowa Electric Light and Power Company

December 18, 1981
LDR-81-262

LARRY D. ROOT
ASSISTANT VICE PRESIDENT
OF NUCLEAR DIVISION

Mr. Harold Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Denton:

In accordance with the requirements of 10CFR50.59 and 50.90, we transmitted our proposed Technical Specification change regarding single recirculation loop operation on October 17, 1980. We hereby amend that application with the enclosed Technical Specification page changes.

This amended application limits single loop rated power operation to 50% maximum. The October 17, 1980 submittal is bounding for 50% power operation.

It is our understanding this is a Class III amendment, therefore, a check for $4,000 is enclosed. This amendment has been reviewed by the Duane Arnold Energy Center Operations Committee and the Safety Committee.

Three signed and 37 additional copies of this application are transmitted herewith. This application consisting of the foregoing letter and enclosures hereto is true and accurate to the best of my knowledge and belief.

IOWA ELECTRIC LIGHT AND POWER COMPANY

BY Larry D. Root
Larry D. Root

Subscribed and sworn to Before Me on
this 18th day of December 1981.

Kathleen M. Herber
Notary Public in and for the State of Iowa

LDR/RFS/kmh*
Enclosure

A001
1/1
w/check!
$4000.00

cc: R. Salmon
K. Eccleston (NRC)
J. Keppler (NRC)
NRC Resident Inspector

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RTS-124A

The following are page changes which represent this amendment to the application which was made regarding single recirculation loop operation on October 17, 1980.

AFFECTED PAGES

1.1-1
1.1-2
1.1-3
1.1-5
3.2-16
3.6-6*
3.6-7
3.12-1
3.12-3
3.12-9
3.12-9a**

* Now contains paragraph E.1.b which was previously on the following page

** Deleted. Information contained on Page 3.12-9a was moved to page 3.12-9.

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the inter-related variables associated with fuel thermal behavior.

Objective:

To establish limits which ensure the integrity of the fuel cladding.

Specifications:

A. Reactor Pressure > 785 psig and Core Flow > 10% of Rated.

The existance of a minimum critical power ratio (MCPR) less than 1.07 for two recirculation loop operation (1.10 for single loop operation) shall constitute violation of the fuel cladding integrity safety limit.

B. Core Thermal Power Limit (Reactor Pressure < 785 psig or Core Flow < 10% of Rated

When the reactor pressure is $\leq$ 785 psig or core flow is less than 10% of rated, the core thermal power shall not exceed 25 percent of rated thermal power.

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

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:

The limiting saftey system settings shall be as specified below:

A. Neutron Flux Trips

1. APRM High Flux Scram When in Run Mode.

For operation with the fraction of rated power (FRP) greater than or equal to the maximum fraction of limiting power density (MFLPD), the APRM scram trip setpoint shall be as shown on Fig. 2.1-1 and shall be:

$$S \leq (0.66W + 54)$$

with a maximum setpoint of 120% rated power at 100% rated recirculation flow or greater.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

C. Power Transient

To ensure that the Safety Limits established in Specification 1.1.A and 1.1.B are not exceeded, each required scram shall be initiated by its primary source signal. A Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the Primary Source Signal.

D. With irradiated fuel in the reactor vessel, the water level shall not be less than 12 in. above the top of the normal active fuel zone. Top of the active fuel zone is defined to be 344.5 inches above vessel zero (see Bases 3.2).

Where: S = Setting in percent of rated power (1,593 MWt)

W = Recirculation loop flow in percent of rated flow. Rated recirculation loop flow is that recirculation loop flow which corresponds to $49x10^6$ lb/hr core flow.

For a MFLPD greater than FRP, the APRM scram setpoint shall be:

$S \leq (0.66\ W + 54)\ \frac{FRP}{MFLPD}$ for two recirculation loop operation, and

$S \leq (0.66\ W + 50.5)\ \frac{FRP}{MFLPD}$

for one recirculation loop operation.

NOTE: These settings assume operation within the basic thermal design criteria. These criteria are LHGR $\leq$ 18.5 KW/ft (7x7 array) or 13.4 KW/ft (8x8 array) and MCPR $\geq$ values as indicated in Table 3.12-2 times K_f, where K_f is defined by Figure 3.12-1. Therefore, at full power, operation is not allowed with MFLPD greater than unity even if the scram setting is reduced. If it is determined that either of these design criteria is being violated during operation, action must be taken immediately to return to operation within these criteria.

2. APRM High Flux Scram

When in the REFUEL or STARTUP and HOT STANDBY MODE, the APRM scram shall be set at less than or equal to 15 percent of rated power.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

2a. For one recirculation loop operation APRM flux noise will be measured once per shift and the recirculation pump speed will be reduced if the flux noise averaged over 1/2 hour exceeds 8% peak to peak, as measured on the APRM chart recorder.

3. APRM Rod Block when in Run Mode.

For operation with MFLPD less than or equal to FRP the APRM Control Rod Block setpoint shall be as shown on Fig. 2.1-1 and shall be:

$S \leq (0.66\ W + 42)$

The definitions used above for the APRM scram trip apply.

For a MFLPD greater than FRP, the APRM Control Rod Block setpoint shall be:

$S \leq (0.66\ W + 42)\ \frac{FRP}{MFLPD}$ for two recirculation loop operation, and

$S \leq (0.66\ W + 38.5)\ \frac{FRP}{MFLPD}$

for one recirculation loop operation.

4. IRM - The IRM scram shall be set at less than or equal to 120/125 of full scale.

B.	Scram and Isolation on reactor low water level	$\geq$ 514.5 inches above vessel zone (+12" on level instruments) zero
C.	Scram - turbine stop valve closure	$\leq$ 10 percent valve closure

D. Turbine control valve fast closure shall occur within 30 milliseconds of the start of turbine control valve fast closure.

1.1 BASES: FUEL CLADDING INTEGRITY

A. Fuel Cladding Integrity Limit at Reactor Pressure $\geq$ 785 psig and Core Flow $\geq$ 10% of Rated

The fuel cladding integrity safety limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedure used to calculate the critical power result in an uncertainity in the value of the cirtical power. Therefore the fuel cladding integrity safety limit is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is generically determined in Reference 1, for two recirculation loop operation. This safety limit MCPR is increased by 0.03 for single-loop operation.

Instrumentation That Initiates Control Rod Blocks

TABLE 3.2-C

Minimum No. of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
2	APRM Upscale (Flow Biased)	for 2 recirc loop operation $\leq (0.66\ W + 42)\ (\frac{FRP}{MFLPD})^{(2)}$ for 1 recirc loop operation $\leq (0.66\ W + 38.5)(\frac{FRP}{MFLPD})^{(2)}$	6 Inst. Channels	(1)
2	APRM Upscale (Not in Run Mode)	$\leq$ 12 indicated on scale	6 Inst. Channels	(1)
2	APRM Downscale	$\geq$ 5 indicated on scale	6 Inst. Channels	(1)
1 (7)	Rod Block Monitor (Flow Biased)	for 2 recirc loop operation $\leq (0.66\ W + 39)(\frac{FRP}{MFLPD})^{(2)}$ for 1 recirc loop operation $\leq (0.66\ W + 35.5)(\frac{FRP}{MFLPD})^{(2)}$	2 Inst. Channels	(1)
1 (7)	Rod Block Monitor Downscale	$\geq$ 5 indicated on scale	2 Inst. Channels	(1)
2	IRM Downscale (3)	$\geq$ 5/125 full scale	6 Inst. Channels	(1)
2	IRM Detector not in Startup Position	(8)	6 Inst. Channels	(1)
2	IRM Upscale	$\leq$ 108/125	6 Inst. Channels	(1)
2 (5)	SRM Detector not in Startup Position	(4)	4 Inst. Channels	(1)
2 (5)(6)	SRM Upscale	$\leq 10^5$ counts/sec.	4 Inst. Channels	(1)

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
2.	2. At least one of the relief valves shall be disassembled and inspected each refueling outage.
a. From and after the date that the safety valve function of one relief valve is made or found to be inoperable, continued reactor operation is permissible only during the succeeding thirty days unless such valve function is sooner made operable.	
b. From and after the date that the safety valve function of two relief valves is made or found to be inoperable, continued reactor operation is permissible only during the succeeding seven days unless such valve function is sooner made operable.	
3. If Specification 3.6.D.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure shall be reduced to atmospheric within 24 hours.	3. With the reactor pressure $\geq$ 100 psig and turbine bypass flow to the main condenser, each relief valve shall be manually opened and verified open by turbine bypass valve position decrease and pressure switches and thermocouple readings downstream of the relief valve to indicate steam flow from the valve once per operating cycle.
E. Jet Pumps	E. Jet Pumps
1. Whenever the reactor is in the startup or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.	1. Whenever there is recirculation flow with the reactor in the startup or run modes, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:
	a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.
	b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.
	2. Whenever there is recirculation flow from the reactor in the Startup or Run mode, and one recirculation pump is operating, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.
F. Jet Pump Flow Mismatch	F. Jet Pump Flow Mismatch
1. When both recirculation pumps are in steady state operation, the speed of the faster pump may not exceed 122% of the speed of the slower pump when core power is 80% or more of rated power or 135% of the speed of the slower pump when core power is below 80% of rated power.	1. Recirculation pump speeds shall be checked and logged at least once per day.
2. If specification 3.6.F.1 cannot be met, one recirculation pump shall be tripped. The reactor may be started and operated, or operated with one recirculation loop out of service provided that: a. MAPLHGR multipliers as indicated in section 3.12A are applied. b. The power level is limited to maximum of 50% of rated power. c. The idle loop is isolated prior to startup, or if disabled during reactor operation, within 24 hours (suction valve closed and electrically disconnected). Refer to specification 3.6.A for startup of the idle recirculation loop.	2. For one recirculation loop out of service the core plate delta p noise will be measured once per shift and the recirculation pump speed will be reduced if the noise exceeds 1 psi peak to peak.

LIMITING CONDITION FOR OPERATION

3.12 CORE THERMAL LIMITS

Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

During reactor power operation, the actual MAPLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figs. 3.12-2, -3, -4, -5, -6 and 7. For single-loop operation, the values in these curves are reduced by multiplying by 0.7. If at any time during reactor power operation (one or two loop) it is determined by normal surveillance that the limiting value for MAPLHGR (LAPLHGR) is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the MAPLHGR (LAPLHGR) is not returned to within the prescribed limits within 2 hours, reduce reactor power to < 25% of rated thermal power within the next 4 hours.

If the reactor is being operated with one recirculation loop out of service and cannot be returned to within prescribed limits within this 4 hour period, the reactor shall be brought to the cold shutdown condition within 36 hours.

For either the one or two loop operating condition surveillance and corresponding action shall continue until the prescribed limits are again being met.

SURVEILLANCE REQUIREMENT

4.12 CORE THERMAL LIMITS

Applicability

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

The MAPLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power and any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for specification 3.3.2. During operation with a limiting control rod pattern, the MAPLHGR (LAPLHGR) shall be determined at least once per 12 hours.

LIMITING CONDITIONS FOR OPERATION

C. Minimum Critical Power Ratio (MCPR)

During reactor power operation MCPR for one or two recirculation loop operation shall be $\geq$ values as indicated in Table 3.12-2. These values are multiplied by K_f which is shown in figure 3.12-1. Note that for one recirculation loop operation the MCPR limits at rated flow are 0.03 higher than the comparable two-loop values. If at any time during reactor power operation (one or two loop) it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the operating MCPR is not returned to within the prescribed limits within two hours, reduce reactor power to $\leq$ 25% of rated thermal power within the next 4 hours.

If the reactor is being operated with one recirculation loop out of service, and cannot be returned to within prescribed limits within this 4 hour period the reactor shall be brought to cold shutdown condition within 36 hours.

For either the one or two loop operating condition surveillance and corresponding action shall continue until the prescribed limits are again being met.

D. Reporting Requirements

If any of the limiting values identified in Specifications 3.12.A, B or C are exceeded, a Reportable Occurrence report shall be submitted. If the corrective action is taken, as described, a thirty-day written report will meet the requirements of this specification.

SURVEILLANCE REQUIREMENTS

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at $\geq$ 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 3.3.2. During operation with a limiting control rod pattern, the MCPR shall be determined at least once per 12 hours.

DAEC - 1

TABLE 3.12-2

MCPR LIMITS

Fuel Type	For two recirculation loop operation	For one recirculation loop operation
7 x 7	1.25	1.28
8 x 8	1.24	1.27
8 x 8R	1.26	1.29