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Ltr notarized 3-12-75...trans the following:

ENCLOSURES:
Amdt to OL/Change to Tech Specs: Consisting of the ddoptation of the GETAB.GEXL heat flux correlation.....(40 cys encl rec'd)

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ACKNOWLEDGED

PLANT NAME: Monticello

FOR ACTION/INFORMATION 3-15-75 ehf

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NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

March 12, 1975

File Cy

Regulatory

RECEIVED
MAR 15 1975
U.S. ATOMIC ENERGY COMMISSION
Regulatory
Mail Section

Mr A Giambusso, Director
Division of Reactor Licensing
U S Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr Giambusso:

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

License Amendment Request Dated March 12, 1975

Attached are three signed originals and 37 conformed copies of a request for a change of Technical Specifications, Appendix A, of the Provisional Operating License for the Monticello Nuclear Generating Plant. The requested changes reflect the adoption of the GETAB/GEXL heat flux correlation in place of the current Hench-Levy correlation. A number of minor changes on those pages under revision have also been included.

Yours very truly,

L. O. Mayer

L O Mayer, PE
Manager of Nuclear Support Services

DOCKETED
USAEC
MAR 15 1975

LOM/MHV/ak

cc: J G Keppler
G Charnoff
Minnesota Pollution Control Agency
Attn: E A Pryzina
MECCA
Attn: L J Vogel
City of Saint Paul
Attn: D L Ficker
S J Gadler

Attachments

2857

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

(License Amendment Request Dated March 12, 1975)

Northern States Power Company, a Minnesota corporation, requests 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.

This request contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By L J Wachter
L J Wachter
Vice President, Power Production & System Operation

On this 12th day of March, 1975, before me a notary public in and for said County, personally appeared L J Wachter, Vice President, Power Production & System Operation, and first being duly sworn acknowledged that he is authorized to execute this document in 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.

John J. Smith
JOHN J. SMITH
Notary Public, Hennepin County, Minnesota
My Commission Expires March 3, 1976

EXHIBIT A

MONTICELLO NUCLEAR GENERATING PLANT
DOCKET NO. 50-263

LICENSE AMENDMENT REQUEST DATED MARCH 12, 1975

PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS
APPENDIX A OF PROVISIONAL OPERATING
LICENSE NO. DPR-22

Pursuant to 10CFR50.59 the holders of Provisional Operating License DPR-22 hereby propose the following changes to Appendix A, Technical Specifications.

The reason for most changes is the adoption of the GETAB/GEXL thermal analysis as a new basis for safety limits. This new, improved correlation replaces the Hench-Levy heat flux correlation which presently serves as the basis for safety limits. The reason for change stated below as "Adoption of GETAB/GEXL" implies this fact. Additional supporting analysis is presented in the following references:

i) General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application; NEDO-10958

ii) General Electric BWR Generic Reload Application for 8x8 Fuel, NEDO-20360, Revision 1

iii) Monticello Nuclear Generating Plant, Third Reload Submittal, December 11, 1974

1. SPECIFICATION 1.0.I (page 2)

PROPOSED CHANGE

Delete the definition of MCHFR and insert the definition of MCPR as indicated.

REASON FOR CHANGE

Adoption of GETAB/GEXL

Note: Our License Amendment Request Dated August 20, 1974, proposed to change this Specification in response to 10CFR50 Appendix K. The change proposed above supercedes both the existing and the August 20, 1974, proposed versions.

2. <u>SPECIFICATIONS 2.1 and 2.3 and BASES 2.1 and 2.3 (pages 6-8, 10 and 13-22A)</u>

PROPOSED CHANGES

Revise the pages to conform to the corrected pages included as Exhibit B. Changes are indicated by sidelines. Page 10 (previously Figure 2.1.1) has been deleted due to the change to Specification 2.1.A.

REASON FOR CHANGE

Adoption of GETAB/GEXL

3. <u>BASES 2.1 (page 17)</u>

PROPOSED CHANGE

Delete the last paragraph of this section, found on page 17 of existing Bases.

REASON FOR CHANGE

This section has become obsolete. It served as the Bases for a reporting requirement on the "Summary Status of Fuel" which has been proposed to be eliminated in License Amendment Request Dated December 16, 1974.

4. <u>BASES 3.2/4.2 (page 67)</u>

PROPOSED CHANGE

Change the third sentence on this page to read, "The trip settings of 200^{o}F and 150% of HPCI and 300% of RCIC design flows and valve closure times are such that the core will not be uncovered and fission product release will not exceed 10CFR100 guidelines."

REASON FOR CHANGE

This change corrects an erroneous statement concerning the RCIC high steam flow setpoint. The trip setting for RCIC High Steam Flow listed on page 51 is less than or equal to 45,000 lb/hr. Design flow is 16,500 lb/hr maximum.

5. <u>BASES 3.2/4.2 (page 67)</u>

PROPOSED CHANGES

In addition to the change in item 4 above, make the remaining changes as indicated in the modified page in Exhibit B. Changes are indicated by sidelines.

REASON FOR CHANGES

Adoption of GETAB/GEXL

6. BASES 3.3/4.3 (page 85)

PROPOSED CHANGES

Revise the page to conform to the corrected page included in Exhibit B. Changes are indicated by sidelines.

REASON FOR CHANGE

Adoption of GETAB/GEXL.

Note: Our Change Request Dated March 1, 1974, proposed to change the same subject matter on page 85 as discussed above. Since that request has not been acted on, we request that the attached changes to page 85 supersede the earlier request and that page 85 be considered withdrawn from Change Request Dated March 1, 1974.

7. SPECIFICATIONS 3.11 AND 4.11 (pages 189B thru 189D and 189K) AND BASES 3.11 AND 4.11 (pages 189E thru 189G)

PROPOSED CHANGES

Revise the pages to conform to the corrected pages included in Exhibit B. Changes are indicated by sidelines. Page 189K (Figure 3.11.2) has been added due to the change in Specification 3.11.C.

REASON FOR CHANGES

Adoption of GETAB/GEXL. The Operating MCPR proposed for Specification 3.11.C is based on the analysis of abnormal operational transients. It is more restrictive than that proposed previously which was the initial condition of LOCA calculations. The required Operating MCPR Limit for 8x8 fuel was erroneously reported in Reference iii to be 1.36. The proposed LCO correctly states the limit to be MCPR of 1.41.

The surveillance requirements have been increased to acknowledge the more restrictive limit. The same surveillance requirements have been applied to all fuel limits addressed in sections 3.11 and 4.11. The increase in surveillance frequency requires that the daily surveillance be augmented by a program specified by the plant technical staff. The necessity for such a program and the frequency at which surveillance should be done is a function of the power level, amount and rate of power change, the associated xenon transient, the control rod sequence and pattern, the core exposure and the margin between operating conditions and limiting values. Because of the interrelationship of all these parameters, a reasonable augmented surveillance can only be determined on a case-by-case basis by the plant technical staff.

Note: Specifications 3.11 and 4.11 were formed by our License Amendment Request Dated August 20, 1974. By order of 10 CFR Part 50.46 the more restrictive of the existing Technical Specifications and those proposed on August 20, 1974, are in effect. The present changes, upon issuance, will supersede the respective portions of the August 20, 1974, submittal. However, since the present changes are not the result of further ECCS evaluations, they are not covered by 10 CFR Part 50.46 and therefore will not become effective until issued by the Nuclear Regulatory Commission.

EXHIBIT B

LICENSE AMENDMENT REQUEST DATED MARCH 12, 1975

Exhibit B, attached, consists of newly prepared pages for the Appendix A Technical Specifications as listed below. These pages incorporate the proposed changes.

Page 2
6
7
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21
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22a
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189B
189C
189D
189E
189F
189G
189K

D. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

E. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the primary sensor to verify the proper instrument channel response, alarm, and/or initiating action.

F. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, accuracy, and response time to a known value (s) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument including actuation, alarm or trip. Response time is not part of the routine instrument calibration but will be checked once per cycle.

G. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

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

I. Minimum Critical Power Ratio (MCPR) - The minimum critical power ratio is the value of critical power ratio associated with the most limiting assembly in the reactor core. Critical power ratio (CPR) is the ratio of that power in a fuel assembly which is calculated by the GEXL correlation to cause some point in the assembly to experience boiling transition to the actual assembly operating power.

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

K. Operable - A system or component shall be considered operable when it is capable of performing its intended function in its required manner.

L. Operating - Operating means that a system or component is performing its required functions in its required manner.

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

2.0 SAFETY LIMITS

2.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variables associated with fuel thermal behavior.

Objective:

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 1.06 shall constitute violation of the fuel cladding integrity safety limit.

LIMITING SAFETY SYSTEM SETTINGS

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.

Objective:

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 as shown in Figure 2.3.1 unless the combination of power and peak heat flux is above the applicable curve in Figure 2.3.2. When the combination of power and peak heat flux is above the curve in Figure 2.3.2, the scram setting (S) is given by:

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.

$$S = \frac{486{,}000\ P}{X} \quad \text{(7x7 fuel)}$$

$$S = \frac{425{,}000\ P}{X} \quad \text{(8x8 fuel)}$$

Where:
P = percent of rated power
X = peak heat flux - ($BTU/HR/FT^2$)
shall be used.

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

B. APRM Rod Block - The APRM rod block setting shall be as shown in Figure 2.3.1 unless the combination of power and peak heat flux is above the applicable curve in Figure 2.3.2. When the combination of power and peak flux is above the curve in Figure 2.3.2, the rod block trip setting (RB) is given by:

$$RB = \frac{437{,}400\ P}{X} \quad \text{(7x7 fuel)}$$

$$RB = \frac{382{,}400\ P}{X} \quad \text{(8x8 fuel)}$$

where:
P = percent of rated power
X = peak heat flux ($BTU/HR/FT^2$)
shall be used.

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

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.

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

E. Turbine Control Valve Fast Closure Scram shall initiate upon loss of pressure at the acceleration relay with turbine first stage pressure $\geq$ 30%.

F. Turbine Stop Valve Scram shall be $\leq$ 10% valve closure from full open with turbine first stage pressure $\geq$ 30%.

G. Main Steamline Isolation Valve Closure Scram shall be $\leq$ 10% valve closure from full open.

This page has been deleted.

Bases:

2.1 The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is no less than 1.06. MCPR > 1.06 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with margin to the conditions which would produce onset of transition boiling. (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation. The concept of MCPR, as used in the GETAB/GEXL critical power analysis, is discussed in Reference 1.

A. Core Thermal Power Limit (Reactor Pressure > 800 psia and Core Flow > 10% of Rated.) Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables. The Safety Limit (T.S.2.1.A) has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from the Operating MCPR Limit (T.S.3.11.C) more than 99.9% of the fuel rods in the core re expected to avoid boiling transition. The margin between MCPR of 1.0 (onset

Bases Continued:

of transition boiling) and the Safety Limit is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1. The uncertainties employed in deriving the Safety Limit are provided at the beginning of each fuel cycle.

Because the boiling transition correlation is based on a large quantity of full scale data, there is a very high confidence that operation of a fuel assembly at the MCPR Safety Limit would not produce boiling transition. Thus, although it is not required to establish the Safety Limit, additional margin exists between the Safety Limit and the actual occurrence of loss of cladding integrity.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to Monticello operated above the boiling transition for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1400 psia during normal power operating (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

In addition to the MCPR Safety Limit, operation is constrained to a maximum LHGR of 17.5 kw/ft for 7x7 fuel and 13.4 kw/ft for 8x8 fuel. At 100% power this limit is reached with a maximum total peaking factor of 3.08 for 7x7 fuel or 3.04 for 8x8 fuel. For the case of the maximum total peaking factor exceeding design, operation is permitted only at less than 100% of rated thermal power and only with reduced APRM scram and rod block settings as required by specifications 2.3.A.1 and 2.3.B.

B. Core Thermal Power Limit (Reactor Pressure $\leq$ 800 psia or Core Flow $\leq$ 10% of Rated) At pressure below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and all core flows, this pressure differential is maintained in the bypass region of the core.

Bases Continued:

Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and all flows will always be greater than 4.56 psi. Analyses show that with a bundle flow of $28x10^3$ lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Therefore, due to the 4.56 psi driving head, the bundle flow will be greater than $28x10^3$ lbs/hr irrespective of total core flow and independent of bundle power for the range of 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 $28x10^3$ lbs/hr 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 initiated by exceeding safety system setting will assure that the Safety Limit of 2.1.A or 2.1.B will not be exceeded. Control rod scram times and safety system settings are checked periodically to assure that a scram will proceed as analyzed. As a further check, the plant process computer will be used as a fast data-acquisition system, when available during a scram, to verify that the scram was initiated by the primary source signal. The computer is normally available for this function. However, it is recognized that the plant may operate without the computer in service, in which event the confirmatory data will not be available and the vertification specified by 2.1.C will not be required. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. For this specification, when a scram is only accomplished by means of a backup feature of the plant design, a specific analysis is required to determine whether or not a Safety Limit has been violated. The concept of not approaching a Safety Limit, providing scram signals are operable, is supported by the extensive plant safety analysis.

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 will be cooled sufficiently to prevent clad melting 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 whenever the recirculation pumps are not operating.

Bases Continued:

References

1. General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, NEDO 10958.

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.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in Reference 1.

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum value expected to occur during the core lifetime. The Doppler reactivity feedback coefficient has conservatively been derated to 90% of the expected value. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion assumed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 20% insertion. The early portion of the scram stroke accomplishes the desired effect by inserting sufficient negative reactivity to turn the transient around. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

Bases Continued:

For analyses of the thermal consequences of the transients, the Operating MCPR Limit (T.S.3.11.C) is conservatively assumed to exist prior to initiation of the transients.

This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

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.

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 operational transients, the thermal power of the fuel will be less than

Bases Continued:

that indicated by the neutron flux at the scram setting. Analyses demonstrate that, with a 120% 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. Therefore, it is intended to ultimately replace (with prior NRC approval) the automatic flow referenced scram with a fixed 120 percent scram setting.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of maximum total peaking factor and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.3.A.1, when the maximum total peaking factor is greater than design. If the APRM scram setting should require a change due to an abnormal peaking condition, it will be done by increasing the APRM gain and thus reducing the slope and intercept point of the flow referenced scram curve by the reciprocal of the APRM gain change. Analyses of the limiting transients show that no scram adjustment is required to assure that the MCPR Safety Limit (T.S.2.1.A) is not exceeded when the transient is initiated from the Operating MCPR Limit (T.S.3.11.C).

For operation in the startup mode while the reactor is at low pressure, the IRM scram setting of 20% of rated power provides adequate thermal margin between the setpoint and the safety limit, 25% 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,

Bases Continued:

temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures 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 open. 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 shown in Figure 2.3.1. However, the actual setpoint can be as much as 3% greater than that shown on Figure 2.3.1 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,

Bases Continued:

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. When the maximum total peaking factor exceeds the design value, the rod block setting is adjusted in accordance with the formula in Specification 2.3.B. If the APRM rod block setting should require a change due to an abnormal peaking condition, it will be done by increasing the APRM gain and thus reducing the slope and intercept point of the flow referenced rod block curve by the reciprocal of the APRM gain change.

The operator will set the APRM rod block trip settings no greater than that shown in Figure 2.3.1. However, the actual setpoint can be as much as 3% greater than that shown on Figure 2.3.1 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.

C. Reactor Low Water Level Scram The reactor low water level scram is set at a point which will assure that the water level used in the bases for the safety limit is maintained.

The operator will set the low water level trip setting no lower than 10'6" above the top of the active fuel. However, the actual setpoint can be as much as 6 inches lower due to the deviations discussed on page 18.

D. Reactor Low Low Water Level ECCS Initiation Trip Point The emergency core cooling subsystems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident and to limit fuel clad temperature to well below the clad melting temperature to assure that core geometry remains intact and to limit any clad metal-water reaction to less than 1%. The design of the ECCS components to meet the above criterion was dependent on three previously set parameters: the maximum break size, the low water level scram setpoint, and the ECCS initiation setpoint. To lower the setpoint for initiation of the ECCS could prevent the ECCS components from

Bases Continued:

meeting their criterion. To raise the ECCS initiation setpoint would be in a safe direction, but it would reduce the margin established to prevent actuation of the ECCS during normal operation or during normally expected transients.

The operator will set the low low water level ECCS initiation trip setting $\geq$ 6'6" $\leq$ 6'10" above the top of the active fuel. However, the actual setpoint can be as much as 3 inches lower than the 6'6" setpoint and 3 inches greater than the 6'10" setpoint due to the deviations discussed on page 18.

E. Turbine Control Valve Fast Closure Scram The turbine control valve fast closure scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass. This transient is less severe than the turbine stop valve closure with bypass failure and therefore adequate margin exists.

F. Turbine Stop Valve Scram The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of $\leq$ 10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the Safety Limit (T.S.2.1.A) even during the worst case transient that assumes the turbine bypass is closed.

G. Main Steam Line Isolation Valve Closure Scram The main steam line isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation closure. With the scram set at 10% valve closure there is no increase in neutron flux.

H. Reactor Coolant Low Pressure Initiates Main Steam Isolation Valve Closure The low pressure isolation of the main steam lines at 850 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires

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 850 psig. However, the actual trip setting can be as much as 10 psi lower due to the deviations discussed on page 18.

References

1. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802, Feb., 1973.

Bases Continued:

3.2 The HPCI and/or RCIC high flow and temperature instrumentation is provided to detect a break in the HPCI and/or RCIC piping. Tripping of this instrumentation results in actuation of HPCI and/or RCIC isolation valves; i.e., Group 4 and/or Group 5 valves. The trip settings of 200°F and 150% of HPCI and 300% of RCIC design flows and valve closure times are such that the core will not be uncovered and fission product release will not exceed 10 CFR 100 guidelines.

The instrumentation which initiates ECCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR remains above the Safety Limit (T.S.2.1.A). The trip logic for this function is 1 out of n; e.g., any trip on one of the six APRM's, eight IRM's, or four SRM's will result in a rod block. The minimum instrument channel requirements for the IRM and RBM may be reduced by one for a short period of time to allow for maintenance, testing, or calibration. See Section 7.3 FSAR.

The APRM rod block trip is referenced to flow and prevents a significant reduction in MCPR especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than the Safety Limit.

The RBM provides local protection of the core; i.e., the prevention of critical power in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern. The trip point is referenced to flow. The worst case single control rod withdrawal error has been analyzed and the results show that with the specified trip settings rod withdrawal is blocked at MCPR greater than the Safety Limit, thus allowing adequate margin. Below 60% power, MCPR remains above the Safety Limit for the worst case withdrawal of a single control rod without rod block action, thus below this level it is not required. This subject is discussed in General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, NEDO-10958. Requiring at least half of the normal LPRM inputs from each level to be operable assures that the RBM response will be adequate to prevent rod withdrawal errors.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCPR approaches the Safety Limit (T.S.2.1.A).

A downscale indication of an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus control rod motion is prevented. The downscale trips are set at 3/125 of full scale.

consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10% of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The consequences of a rod block monitor failure have been evaluated. These evaluations show that during reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCPR's below the Safety Limit (T.S.2.1.A). During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Engineer, Nuclear, to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable rods in other than limiting patterns.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the Safety Limit (T.S.2.1.A). This requires the negative reactivity insertion in any local region of the core and in the overall core to be equivalent to at least the scram reactivity curve used in the transient analysis. The required average scram times for three control rods in all two by two arrays and the required average scram times for all control rods are based on inserting this amount of negative reactivity at the specified rate locally and in the overall core. Under these conditions, the thermal limits are never reached during the transients requiring control rod scram. The limiting operational transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above Specification, provide the required protection, and MCPR remains above the Safety Limit (T.S.2.1.A). In the analytical treatment of the transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods.

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 steady state power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figures 3.11-1. If at any time it is determined that the limiting value for APLHGR is being exceeded, action shall be taken immediately to restore operation to within the prescribed limits.

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)

1. The APLHGR 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.
2. Whenever the plant technical staff determines that more frequent surveillance of APLHGR is necessary, it shall specify an augmented surveillance program commensurate with reactor conditions.

B. Local LHGR

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation:

$$LHGR_{max} \leq LHGR_{d} \left[1 - \left(\frac{\Delta P}{P}\right)_{max} \left(\frac{L}{LT}\right) \right]$$

$LHGR_d$ = Design LHGR

= 17.5 kw/ft for 7x7 fuel

= 13.4 kw/ft for 8x8 fuel

$\left(\frac{\Delta P}{P}\right)_{max}$ = Maximum power spiking penalty

= 0.026 for 7x7 fuel

= 0.021 for 8x8 fuel

LT= Total core length = 12 ft

L= Axial position above bottom core

If at any time it is determined that the limiting value of LHGR is being exceeded, action shall be taken immediately to restore operation to within prescribed limits.

B. Local LHGR

1. The local LHGR as a function of core height shall be checked daily during reactor operation at ≥ 25% of rated thermal power.

2. Whenever the plant technical staff determines that more frequent surveillance of local LHGR is necessary, it shall specify an augmented surveillance program commensurate with reactor conditions.

C. Minimum Critical Power Ratio (MCPR)

During steady state power operation, the Operating MCPR Limit shall be ≥1.41 for 8x8 fuel and ≥ 1.33 for 7x7 fuel at rated power and flow. For core flows other than rated the Operating MCPR Limit shall be the above value multiplied by K_f, where K_f is given by Figure 3.11.2. If at any time it is determined that the limiting value of MCPR is being exceeded, action shall be taken immediately to restore operation to within prescribed limits.

C. Minimum Critical Power Ratio (MCPR)

1. MCPR shall be checked daily during reactor power operation at ≥ 25% rated thermal power.

2. Whenever the plant technical staff determines that more frequent surveillance of MCPR is necessary, it shall specify an augmented surveillance program commensurate with reactor conditions.

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^{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 10CFR50 Appendix K limit. The limiting value for APLHGR is given by this specification.

It is recognized that APLHGR is a calculated parameter that is not continually monitored and alarmed directly during core power distribution changes. If at the time of the calculation it is found that the limits are being exceeded, there is always an action which will return the average planar LHGR to within prescribed limits, namely power reduction. Under most circumstances, this will not be the only alternative. Whenever the limit is exceeded the monitored value will be documented and available for review, audit and inspection of plant operations. The only way to violate the Limiting Condition for Operation is to knowingly allow operation beyond the prescribed limits without taking the necessary action to restore the average planar LHGR to within prescribed limits.

B. Local 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.

It is recognized that LHGR is a calculated parameter that is not continually monitored and alarmed directly during core power-distribution changes. If at the time of the calibration it is found that the limits are being exceeded, there is always an action which will return the LHGR to within prescribed limits, namely power reduction. Under most circumstances, this will not be the only alternative. Whenever the limit is exceeded the monitored value will be documented and available for review, audit and inspection of plant operations. The only way to violate the Limiting Condition for Operation is to knowingly allow operation beyond the prescribed limits without taking the necessary action to restore the LHGR to within prescribed limits.

C. Minimum Critical Power Ratio (MCPR)

The ECCS evaluation presented in Reference 4 assumed the steady state MCPR prior to the postulated loss of coolant accident to be 1.19 for all fuel types. The Operating MCPR Limit of 1.41 for 8x8 fuel and 1.33 for 7x7 fuel 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 of 1.06 (T.S.2.1.A) applicable to all fuel types is maintained in the event of the most limiting abnormal operational transient.

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.

It is recognized that MCPR is a calculated parameter that is not continually monitored and alarmed directly during core power distribution and thermal-hydraulic changes. If at the time of the evaluation it is found that the limits are being exceeded, there is always an action which will return the MCPR to within prescribed limits, namely power reduction. Under most circumstances, this will not be the only alternative. Whenever the limit is exceeded the monitored value will be documented and available for review, audit and inspection of plant operations. The only way to violate the Limiting Condition for Operation is to knowingly allow operation beyond the prescribed limits without taking the necessary action to restore the MCPR to within prescribed limits.

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: V A Moore to I S Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

4. "Monticello Nuclear Generating Plant Loss-Of-Coolant Accident Analysis Conformance with 10 CFR 50 Appendix K, August 1974," L O Mayer (NSP) to J F O'Leary, August 20, 1974.

5. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," NEDO-10958, November, 1973.

Bases 4.11

The APLHGR, LHGR and MCPR shall be checked daily to determine if fuel burnup, or control rod movement has 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.

At certain times during plant startups and power changes the plant technical staff may determine that surveillance of APLHGR, LHGR and/or MCPR is necessary more frequently than daily. Because the necessity for such an augmented surveillance program is a function of a number of interrelated parameters, a reasonable program can only be determined on a case-by-case basis by the plant technical staff. The check of APLHGR, LHGR and MCPR will normally be done using the plant process computer. In the event that the computer is unavailable, the check will consist of either a manual calculation or a comparison of existing core conditions to those existing at the time of a previous check to determine if a significant change has occurred.

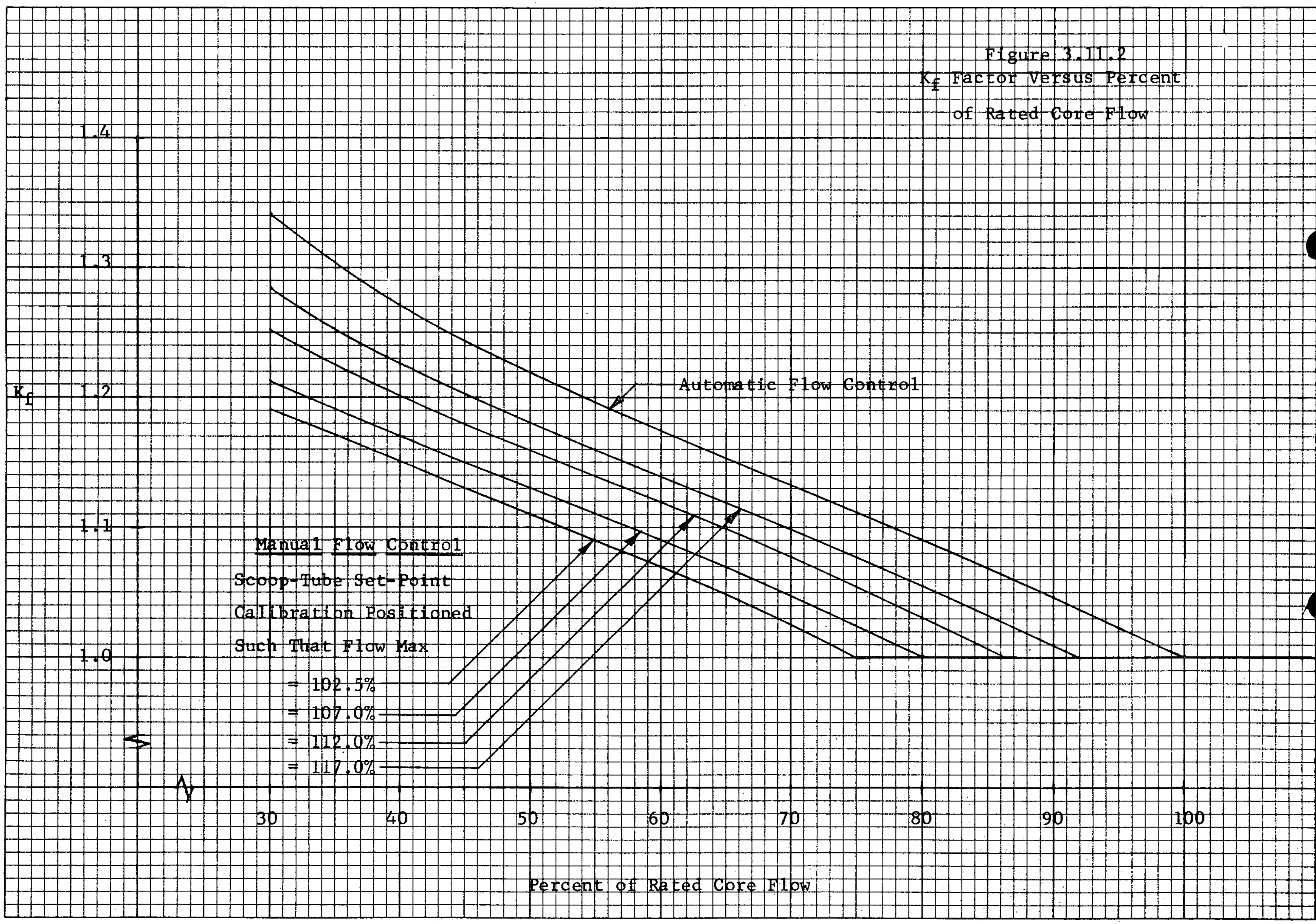
Figure 3.11.2
K_f Factor Versus Percent
of Rated Core Flow
Automatic Flow Control
Manual Flow Control
Scoop-Tube Set-Point
Calibration Positioned
Such That Flow Max
= 102.5%
= 107.0%
= 112.0%
= 117.0%
K_f
1.4
1.3
1.2
1.1
1.0
30
40
50
60
70
80
90
100
Percent of Rated Core Flow

189K
REV