September 24, 1985

Docket No. 50-298

Mr. J. M. Pilant, Technical
Staff Manager
Nuclear Power Group
Nebraska Public Power District
Post Office Box 499
Columbus, Nebraska 68601

Dear Mr. Pilant:

The Commission has issued the enclosed Amendment No. 94 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. This amendment consists of changes to the Technical Specifications in response to your application dated September 20, 1985, as supplemented by letter dated September 23, 1985.

The amendment revises the Technical Specifications to incorporate changes to (1) permit reactor operation with one recirculation loop out of service, (2) provide for detection and suppression of thermal-hydraulic instabilities during both dual loop and single loop operation, and (3) update some references and delete some blank pages.

A copy of the Safety Evaluation is also enclosed.

Sincerely,

Original signed by/

Ernest D. Sylvester, Project Manager
Operating Reactors Branch #2
Division of Licensing

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Enclosures:
1. Amendment No. 94 to License No. DPR-46
2. Safety Evaluation

cc w/enclosures:
See next page

9/24/85

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Mr. J. M. Pilant
Nebraska Public Power District

Cooper Nuclear Station

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NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 94
License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District dated September 20, 1985, as supplemented September 23, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility Operating License No. DPR-46 is hereby amended to read as follows:

(2) Technical Specification

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

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

D B Vassallo

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 24, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 94

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Pages

i
ii
6
7
8
11
12
13, 14, 15, 16 (Blank)
17
18
19
20
21
22 (Blank)
28
31
42
61
62
62a
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104
105, 106 (Blank)
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Cooper Nuclear Station Technical Specifications

1.1 FUEL CLADDING INTEGRITY

Applicability

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

Objective

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

Action

If a Safety Limit is exceeded, the reactor shall be in at least hot shutdown within 2 hours.

Specifications

A. Reactor Pressure $\geq$800 psia and Core Flow $\geq$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.

B. Core Thermal Power Limit (Reactor Pressure <800 psia and/or Core Flow <10%)

When the reactor pressure is <800 psia or core flow is less than 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

C. Power Transient

To ensure that the Safety Limit established in Specification 1.1.A and 1.1.B is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal.

2.1 FUEL CLADDING INTEGRITY

Applicability

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

Objective

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

Specifications

A. Trip Settings

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

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

$S \leq 0.66\ W + 54\% - .66\ \Delta W$

where:

S = Setting in percent of rated thermal power (2381 MWt)

W = Two-loop recirculation flow rate in percent of rated (rated loop recirculation flow rate is that recirculation flow rate which provides 100% coreflow at 100% power)

ΔW = Difference between two-loop and single-loop effective drive flow at the same core flow.

1.1 (Cont'd)

D. Cold Shutdown

Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 18 in. above the top of the normal active fuel zone (top of active fuel is defined in Figure 2.1.1).

2.1.A.1 (Cont'd)

$\Delta W = 0$ for two recirculation loop operation.

a. In the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (0.66\ W + 54\% - 0.66\ \Delta W)\left[\frac{FRP}{MFLPD}\right]$$

where,

FRP = fraction of rated thermal power (2381 MWt)

MFLPD = maximum fraction of limiting power density where the limiting power density is 18.5 KW/ft for 7x7 fuel and 13.4 KW/ft for 8x8 fuel.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

b. APRM Flux Scram Trip Setting (Refuel or Start and Hot Standby Mode)

When the reactor mode switch is in the REFUEL or STARTUP position, the APRM scram shall be set at less than or equal to 15% of rated power.

c. IRM

The IRM flux scram setting shall be ≤120/125 of scale.

2.1.A.1 (Cont'd)

d. APRM Rod Block Trip Setting

The APRM rod block trip setting shall be:

$S_{RB} \leq 0.66\ W + 42\% - .66\ \Delta W$

where:

S_{RB} = Rod block setting in percent of rated thermal power (2381 MWt)

W and ΔW are defined in Specification 2.1.A.1.a.

In the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S_{RB} \leq (0.66\ W + 42\% - 0.66\ \Delta W)\left[\frac{FRP}{MFLPD}\right]$$

where,

FRP = fraction of rated thermal power (2381 MWt)

MFLPD - maximum fraction of limiting power density where the limiting power density is 18.5 KW/ft for 7x7 fuel and 13.4 KW/ft for 8x8 fuel.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

2. Reactor Water Low Level Scram and Isolation Trip Setting (except MSIV)

$\geq$ +12.5 in. on vessel level instruments.

Fuel Cladding Integrity

A. Fuel Cladding Integrity Limit at Reactor Pressure ≥800 psia 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 < 800 psia and/or Core Flow < 10% of Rated)

At pressures below 800 psia, the core elevation 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 10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28 x 10^3 lbs/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 psi 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 1.1A or 1.1B 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 closure 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 Cooper 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 shutdown, 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 18 inches above the top of the fuel provides adequate margin.

References for 1.1 Bases

1. "Generic Reload Fuel Application," NEDE-24011-P (most current approved submittal).

2. "Cooper Nuclear Station Single-Loop Operation," NEDO-24258, May, 1980.

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2.1 Bases:

The abnormal operational transients applicable to operation of the CNS Unit have been analyzed throughout the spectrum of planned operating conditions up to 105% of rated steam flow. The analyses were based upon plant operation in accordance with Reference 3. In addition, 2381 MWt is the licensed maximum power level of CNS, and this represents the maximum steady-state power which shall not knowingly be exceeded.

The 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.

A. Trip Settings

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

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (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 (2381 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% 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 maximum fraction of limiting power density (MFLPD) and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1.a, when the MFLPD is greater than the fraction of rated power (FRP). This adjustment may be accomplished by increasing the APRM gain and thus reducing the slope and intercept point of the flow referenced APRM High Flux Scram Curve by the reciprocal of the APRM gain change.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR above the safety limit when the transient is initiated from the operating MCPR limit.

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

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accomodate 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 procedure backed up by the rod worth minimizer, and the rod sequences control system. 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 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 change can occur when reactor pressure is greater than Specification 2.1.A.6.

c. IRM Flux Scram Trip Setting

The IRM system consists of 8 chambers, 4 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 1, the scram setting would be 120 divisions for that range; likewise, 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 in-sequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that 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 provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was 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 the MCPR fuel cladding integrity 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.

d. APRM Rod Block Trip Setting

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block which is dependent on recirculation flow rate to limit rod withdrawal, thus protecting against a MCPR of less than the MCPR fuel cladding integrity safety limit. 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.

2. Reactor Water Low Level Scram and Isolation Trip Setting (except MSIV)

The set point for low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in SAR Subsection 14.5 show that scram at this level adequately protects the fuel and the pressure barrier, because MCPR remains well above the MCPR fuel cladding integrity limit in all cases, and system pressure does not reach the safety valve settings. The scram setting is approximately 25 in. below the normal operating range and is thus adequate to avoid spurious scrams.

3. Turbine Stop Valve Closure Scram Trip Setting

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 ≤10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the MCPR fuel cladding integrity limit even during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure.

4. Turbine Control Valve Fast Closure Scram Trip Setting

The turbine control valve fast closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection exceeding the capability of the bypass valves. The reactor protection system initiates a scram when fast closure of the control valves is initiated by the loss of turbine control oil pressure as sensed by pressure switches. This setting and the fact that control valve closure time is approximately twice as long as that for the stop valves means that resulting transients, while similar, are less severe than for stop valve closure. No significant change in MCPR occurs. Relevant transient analyses are presented in Paragraph 14.5.1.1 of the Safety Analysis Report.

2.1 Bases: (Cont'd)

5. Main Steam Line Isolation Valve Closure on Low Pressure

The low pressure isolation of the main steam lines (Specification 2.1.A.6) was provided to protect against rapid reactor depressurization.

B. Reactor Water Level Trip Settings Which Initiate Core Standby Cooling System (CSCS)

The core standby 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 assure that core geometry remains intact and to limit any clad metal-water reaction to less than 1%. To accomplish their intended function, the capacity of each Core Standby Cooling System component was established based on the reactor low water level scram set point. To lower the set point of the low water level scram would increase the capacity requirement for each of the CSCS components. Thus, the reactor vessel low water level scram was set low enough to permit margin for operation, yet will not be set lower because of CSCS capacity requirements.

The design for the CSCS components to meet the above guidelines was dependent upon three previously set parameters: The maximum break size, low water level scram set point and the CSCS initiation set point. To lower the set point for initiation of the CSCS may lead to a decrease in effective core cooling. To raise the CSCS initiation set point would be in a safe direction, but it would reduce the margin established to prevent actuation of the CSCS during normal operation or during normally expected transients.

Transient and accident analyses reported in Section 14 of the Safety Analyses Report demonstrate that these conditions result in adequate safety margins for the fuel.

C. References for 2.1 Bases

1. "Generic Reload Fuel Application," NEDE-24011-P, (most current approved submittal).

2. "Cooper Nuclear Station Single-Loop Operation," NEDO-24258, May 1980.

3. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1," (applicable reload document).

4. Safety Analysis Report (Section XIV).

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TABLE 3.1.1
REACTOR PROTECTION SYSTEM INSTRUMENTATION REQUIREMENTS

Reactor Protection System Trip Function	Applicability Conditions Mode Switch Position: Shutdown	Startup	Refuel	Run	Trip Level Setting	Minimum Number of Operable Channels Per Trip Systems (1)	Action Required When Equipment Operability is Not Assured (1)
Mode Switch in Shutdown	X(7)	X	X	X		1	A
Manual Scram	X(7)	X	X	X		1	A
IRM (17) High Flux	X(7)	X	X	(5)	$\leq$ 120/125 of indicated scale	3	A
Inoperative		X	X	(5)		3	A
APRM (17) High Flux (Flow biased)				X	$\leq (0.66W+54\%-0.66\Delta W) \left[\frac{FRP}{MFLPD}\right]$ (14)(19)	2	A or C
High Flux	X(7)	X(9)	X(9)	(16)	$\leq$ 15% Rated Power	2	A or C
Inoperative		X(9)	X(9)	X	(13)	2	A or C
Downscale		(11)		X(12)	$\geq$ 2.5% of indicated scale	2	A or C
High Reactor Pressure NBI-PS-55 A,B,C, & D		X(9)	X(10)	X	$\leq$ 1045 psig	2	A
High Drywell Pressure PC-PS-12 A,B,C, & D		X(9)(8)	X(8)	X	$\leq$ 2 psig	2	A or D
Reactor Low Water Level NBI-LIS-101 A,B,C, & D		X	X	X	$\geq$ + 12.5 in. indicated level	2	A or D
Scram Discharge Instrument Volume High Water Level CRD-LS-231 A & B CRD-LS-234 A & B CRD-LT-231 C & D CRD-LT-234 C & D		X	X(2)	X	$\leq$ 92 inches	3 (18)	A

XXXXXXX

11. The APRM downscale trip function is only active when the reactor mode switch is in RUN.

12. The APRM downscale trip is automatically bypassed when the mode switch is not in RUN.

13. An APRM will be considered inoperable if there are less than 2 LPRM inputs per level or there is less than 11 operable LPRM detectors to an APRM.

14. W is the two-loop recirculation flow in percent of rated flow.

15. This note deleted.

16. The 15% APRM scram is bypassed in the RUN mode.

17. The APRM and IRM instrument channels function in both the Reactor Protection System and Reactor Manual Control System (Control Rod Withdraw Block, Section 3.2.C.). A failure of one channel will affect both of these systems.

18. The minimum number operable associated with the Scram Discharge Instrument Volume are three instruments per Scram Discharge Instrument Volume and three level devices per RPS channel.

19. ΔW is the difference between two-loop and single-loop effective drive flow and is used for single recirculation loop operation. ΔW=0 for two recirculation loop operation.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.1 BASES (Cont'd.)

there is proper overlap in the neutron monitoring system functions and thus, that adequate coverage is provided for all ranges of reactor operation.

4.1 BASES (Cont'd.)

For the APRM system, drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1.1 and 4.1.2 indicates that two instrument channels have not been included in the latter table. These are: mode switch in shutdown and manual scram. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration during operation is not applicable.

B. The MFLPD is checked once per day to determine if the APRM scram requires adjustment. This will normally be done by checking the LPRM readings. Only a small number of control rods are moved daily and thus the MFLPD is not expected to change significantly and thus a daily check of the MFLPD is adequate.

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. This is compensated for in the APRM system by calibrating once a week using a heat balance data and by calibrating individual LPRM's every six weeks of power operation above 20% of rated power.

It is highly improbable that in actual operation with MFLPD $\leq$ FRP that MCPR will be as low as the MCPR fuel cladding integrity safety limit. Usually with power densities of this magnitude the peak occurs low in the core in a low quality region where the initial heat

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TABLE 3.2.C
CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

Function	Trip Level Setting	Minimum Number Of Operable Instrument Channels/Trip System(5)
APRM Upscale (Flow Bias)	$\leq (0.66W + 42\% - 0.66\ \Delta W)\left[\frac{FRP}{MFLPD}\right]$ (2)(13)	2(1)
APRM Upscale (Startup)	$\leq$ 12%	2(1)
APRM Downscale (9)	$\geq$ 2.5%	2(1)
APRM Inoperative	(10b)	2(1)
RBM Upscale (Flow Bias)	$\leq$ 0.66W + (N - 66) (2)	1
RBM Downscale (9)	$\geq$ 2.5%	1
RBM Inoperative	(10c)	1
IRM Upscale (8)	$\leq$ 108/125 of Full Scale	3(1)
IRM Downscale (3)(8)	$\geq$ 2.5%	3(1)
IRM Detector Not Full In (8)		3(1)
IRM Inoperative (8)	(10a)	3(1)
SRM Upscale (8)	$\leq 1 \times 10^5$ Counts/Second	1(1)(6)
SRM Detector Not Full In (4)(8)	($\geq$ 100 cps)	1(1)(6)
SRM Inoperative (8)	(10a)	1(1)(6)
Flow Bias Comparator	$\leq$ 10% Difference In Recirc. Flows	1
Flow Bias Upscale/Inop.	$\leq$ 110% Recirc. Flow	1
SRM Downscale (8)(7)	$\geq$ 3 Counts/Second (11)	1(1)(6)
SDV Water Level High CRD-231E, 234E	$\leq$ 46 inches	1(12)

NOTES FOR TABLE 3.2.C

1. For the startup and run positions of the Reactor Mode Selector Switch, the Control Rod Withdrawal Block Instrumentation trip system shall be operable for each function. The SRM and IRM blocks need not be operable in "Run" mode, and the APRM (flow biased) and RBM rod blocks need not be operable in "Startup" mode. The Control Rod Withdrawal Block Instrumentation trip system is a one out of "n" trip system, and as such requires that only one instrument channel specified in the function column must exceed the Trip Level Setting to cause a rod block. By utilizing the RPS bypass logic (see note 5 below and note 1 of Table 3.1.1) for the Control Rod Withdrawal Block Instrumentation, a sufficient number of instrument channels will always be operable to provide redundant rod withdrawal block protection.

2. W is the two-loop recirculation flow rate in percent of rated. Trip level setting is in percent of rated power (2381 MWt). N is the RBM setpoint selected (in percent) and is calculated in accordance with the methodology of the latest NRC approved version of NEDE-24011-P-A.

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

4. This function is bypassed when the count is $\geq$ 100 cps and IRM above range 2.

5. By design one instrument channel; i.e., one APRM or IRM per RPS trip system may be bypassed. For the APRM's and IRM's, the minimum number of channels specified is that minimum number required in each RPS channel and does not refer to a minimum number required by the control rod block instrumentation trip function. By design only one of two RBM's or one of four SRM's may be bypassed. For the SRM's, the minimum number of channels specified is the minimum number required in each of the two circuit loops of the Control Rod Block Instrumentation Trip System. For the RBM's, the minimum number of channels specified is the minimum number required by the Control Rod Block Instrumentation Trip System as a whole (except when a limiting control rod pattern exists and the requirements of Specification 3.3.B.5 apply).

6. IRM channels A,E,C,G all in range 8 or higher bypasses SRM channels A&C functions. IRM channels B,F,D,H all in range 8 or higher bypasses SRM channels B&D functions.

7. This function is bypassed when IRM is above range 2.

8. This function is bypassed when the mode switch is placed in Run.

9. This function is only active when the mode switch is in Run. This function is automatically bypassed when the IRM instrumentation is operable and not high.

10. The inoperative trips are produced by the following functions:

    a. SRM and IRM

        (1) Mode switch not in operate
        (2) Power supply voltage low
        (3) Circuit boards not in circuit

b. APRM

(1) Mode switch not in operate
(2) Less than 11 LPRM inputs
(3) Circuit boards not in circuit

c. RBM

(1) Mode switch not in operate
(2) Circuit boards not in circuit
(3) RBM fails to null
(4) Less than required number of LPRM inputs for rod selected

11. During spiral unloading/reloading, the SRM count rate will be below 3 cps for some period of time. See Specification 3.10.B.

12. With the number of OPERABLE channels less than required by the Minimum Number of Operable Instrument Channels/Trip System requirements, place the inoperable channel in the tripped condition within one hour.

13. ΔW is the difference between two-loop and single-loop effective drive flow and is used for single recirculation loop operation. $\Delta W=0$ for two recirculation loop operation.

3.3.C (Cont'd.)

3. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

D. Reactivity Anomalies

At a specific steady state base condition of the reactor actual control rod inventory will be periodically compared to a normalized computer prediction of the inventory. If the difference between observed and predicted rod inventory reaches the equivalent of 1% Δk reactivity, the reactor will be shut down until the cause has been determined and corrective actions have been taken as appropriate.

E. Restrictions

If Specifications 3.3.A through D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Shutdown condition within 24 hours.

F. Recirculation Pumps

1. A recirculation pump shall not be started while the reactor is in natural circulation flow and reactor power is greater than 1% of rated thermal power.

2. With two recirculation pumps in operation and with core thermal power greater than the limit specified in Figure 3.3.1 and total core flow less than 45% of rated, the APRM and LPRM* neutron flux noise levels shall be determined within 2 hours, and:

   a) if the APRM and LPRM* neutron flux noise levels are less than or equal to three times their established baseline levels, continue to determine the noise levels at least once per 8 hours and also within 30 minutes after the completion of a core thermal power increase of at least 5% of rated core thermal power while operating in this region of the power/flow map, or

* Detector levels A and C of one LPRM string per core octant plus detector levels A and C of one LPRM string in the center of the core shall be monitored.

4.3.C (Cont'd.)

D. Reactivity Anomalies

During the startup test program and startup following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

F. Recirculation Pumps

1. With two recirculation pumps in operation and with core thermal power greater than the limit specified in Figure 3.3.1 and total core flow less than 45% of rated, establish baseline APRM and LPRM* neutron flux noise levels within 2 hours, provided that baseline values have not been previously established since the last core refueling.

2. a) Prior to operation with one recirculation pump not in operation and core thermal power greater than the limit specified in Figure 3.3.1 establish baseline APRM and LPRM* neutron flux noise levels, provided that baseline values have not been previously established since the last core refueling. Baseline values shall be established with one recirculation pump not in operation and core thermal power less than or equal to the limit specified in Figure 3.3.1.

   b) Prior to operation with one recirculation pump not in operation and core flow greater than 45% of rated, establish baseline core plate ΔP noise levels with core flow less than or equal to 45% of rated, provided that baseline values have not been previously established with one recirculation pump not in operation since the last core refueling.

3.3.F (Cont'd.)

b) if the APRM and/or LPRM* neutron flux noise levels are greater than three times their established baseline levels, immediately initiate corrective action and restore the noise levels to within the required limits within 2 hours by increasing core flow, and/or by initiating an orderly reduction of core thermal power by inserting control rods.

3. The reactor may be started and operated, or operation may continue with one recirculation loop not in operation provided that;

a. with one recirculation pump not in operation and core thermal power greater than the limit specified in Figure 3.3.1, core flow must be greater than or equal to 45% of rated, and

(i) the Surveillance Requirements of 4.3.F.2.a have not been satisfied, immediately initiate action to reduce core thermal power to less than or equal to the limit specified in Figure 3.3.1 within 4 hours, or

(ii) the Surveillance Requirements of 4.3.F.2.a have been satisfied, continue to determine the APRM and LPRM neutron flux levels at least once per 8 hours and also within 30 minutes after the completion of a core thermal power increase of at least 5% of rated core thermal power while operating in this region of the power/flow map. If the APRM and/or LPRM* neutron flux noise levels are greater than three times their established baseline values, immediately initiate corrective action and restore the noise levels to within the required limits within 2 hours by

* Detector levels A and C of one LPRM string per core octant plus detector levels A and C of one LPRM string in the center of the core shall be monitored.

4.3 (Cont'd.)

G. Scram Discharge Volume

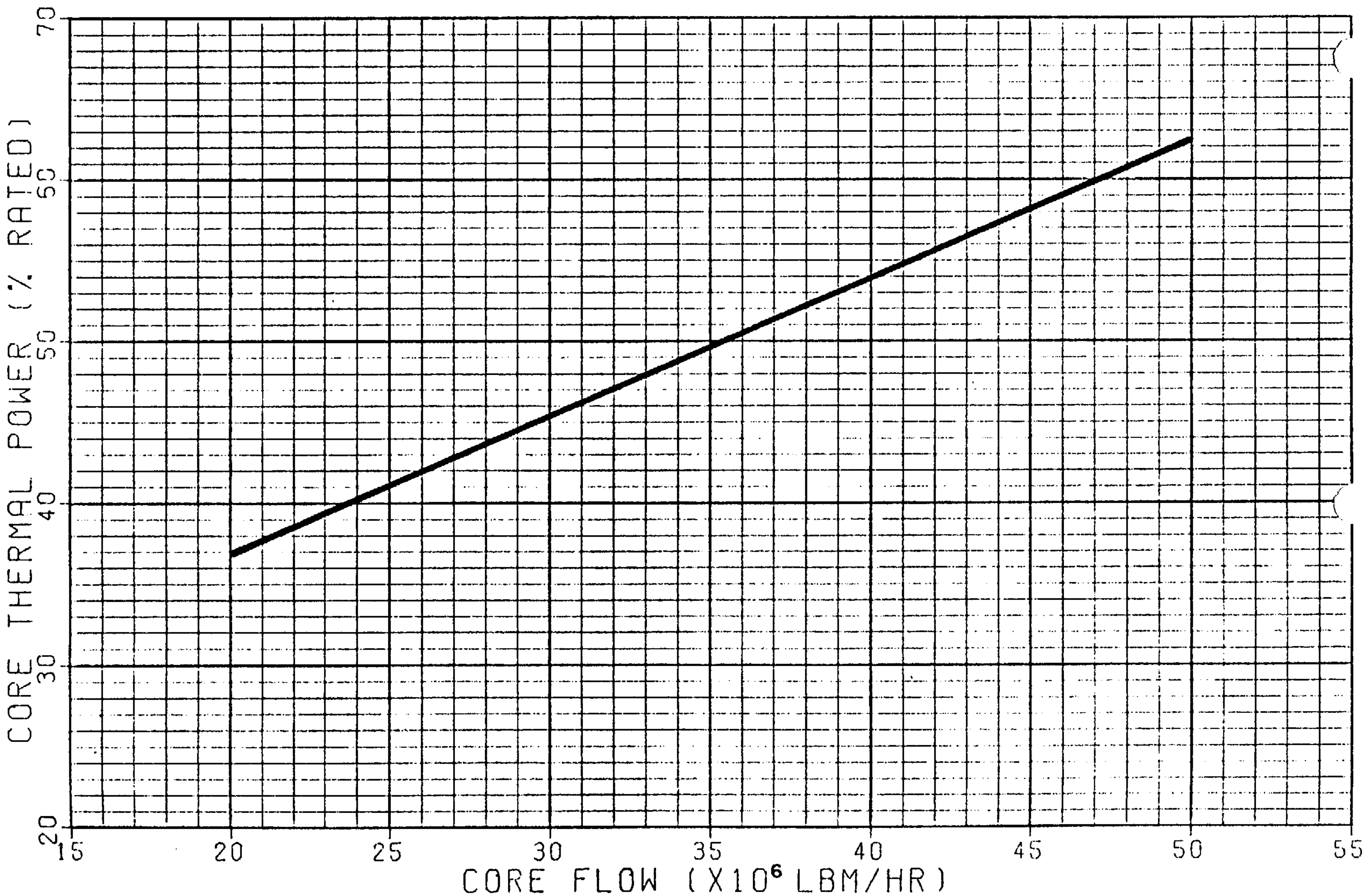
1. The scram discharge volume (SDV) vent and drain valves shall be cycled and verified open at least once every 31 days and prior to reactor start-up.

2. The SDV vent and drain valves shall be verified to close within 30 seconds after receipt of a signal for control rod scram once per refueling cycle.

3. SDV vent and drain valve operability shall be verified following any maintenance or modification to any portion (electrical or mechanical) of the SDV which may affect the operation of the vent and drain vavles.

FIGURE 3.3.1.

THERMAL POWER LIMITATIONS DURING OPERATION WITH LESS THAN TWO REACTOR COOLANT SYSTEM RECIRCULATION LOOPS IN OPERATION



LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.3.F (Cont'd.)

increasing core flow and/or initiating an orderly reduction of core thermal power by inserting control rods.

b. With one recirculation pump not in operation and core flow greater than 45% of rated, and

   (i) the Surveillance Requirements of 4.3.F.2.b have not been satisfied, immediately initiate action to reduce core flow to less than or equal to 45% of rated within 4 hours, or

   (ii) the Surveillance Requirements of 4.3.F.2.b have been satisfied, continue to determine core plate ΔP noise at least once per 8 hours and also within 30 minutes after the completion of a core thermal power increase of at least 5% of rated thermal power. If the core plate ΔP noise level is greater than 1.0 psi and 2 times its established baseline value, immediately initiate corrective action and restore the noise levels to within the required limits within 2 hours by decreasing core flow and/or initiating an orderly reduction of core thermal power by inserting control rods.

c. The idle loop is isolated electrically by disconnecting the breaker to the recirculation pump motor generator (M/G) set drive motor prior to startup, or if disabled during reactor operation, within 24 hours.

d. The recirculation system controls will be placed in the manual flow control mode.

the control rod motion is estimated to actually begin. However, 200 milliseconds is conservatively assumed for this time interval in the transient analyses and this is also included in the allowable scram insertion times of Specification 3.3.C. The time to deenergize the pilot valve scram solenoid is measured during the calibration tests required by Specification 4.1.

D. Reactivity Anomalies

During each fuel cycle excess operative reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern at selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons.

Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% Δk. Deviations in core reactivity greater than 1% Δk are not expected and require thorough evaluation. One percent reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

F. Recirculation Pumps

Until analyses are submitted for review and approval by the NRC which prove that recirculation pump startup from natural circulation does not cause a reactivity insertion transient in excess of the most severe coolant flow increase currently analyzed, Specification 3.3.F.1 prevents starting recirculation pumps while the reactor is in natural circulation above 1% of rated thermal power. Specifications 3.3.F.2 and 3 are based upon providing assurance that neutron flux limit cycle oscillations, which have a small probability of occurring in the high power/low flow corner of the operating domain, are detected and suppressed. BWR cores typically operate with neutron flux noise levels of 1%-12% of rated power (peak to peak) due to random boiling and flow noise. These flux noise levels are considered in the thermal/mechanical design of GE BWR fuel, occur in a stable mode, and are found to be of negligible consequence. However, under certain high power/low flow conditions that could occur during a recirculation pump trip and subsequent Single Loop Operation (SLO) where reverse flow occurs in inactive jet pumps, a hydraulic/reactor kinetic feedback mechanism can be enhanced such that sustained limit cycle oscillations of flow noise with peak to peak levels several times normal values are exhibited. Although large margins to safety limits are maintained when these limit cycle oscillations occur, they are to be monitored for, and suppressed when flux noise exceeds the three time baseline value by inserting rods and/or increasing coolant flow. The line in Figure 3.3.1 is based on the 80% rod line below which the probability of limit cycle oscillations occurring is negligible. The thermal power, core flow, and neutron flux noise level limitations are prescribed in accordance with Reference 3.

G. Scram Discharge Volume

To ensure the Scram Discharge Volume (SDV) does not fill with water, the vent and drain valves shall be verified open at least once every 31 days. This is to preclude establishing a water inventory, which if sufficiently large, could result in slow scram times or only a partial control rod insertion.

The vent and drain valves shut on a scram signal thus providing a contained volume (SDV) capable of receiving the full volume of water discharged by the control rod drives at any reactor vessel pressure. Following a scram the SDV is discharged into the reactor building drain system.

REFERENCES

1. Licensing Topical Report GE-BWR Generic Reload Fuel Application, NEDE-24011-P, (most current approved submittal).

2. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1," (applicable reload document).

3. General Electric Service Information Letter No. 380, Revision 1, dated February 10, 1984.

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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.6.E Jet Pumps

1. Whenever the reactor is in the start-up or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instruments failures occur and cannot be corrected within 24 hours, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.

F. Jet Pump Flow Mismatch

1. Deleted.

2. Following one-pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is equal to or less than 50% of its rated speed.

G. Inservice Inspection

To be considered operable, components shall satisfy the requirements contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for continued service of ASME Code Class 1, 2, and 3 components except where relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

4.6.E. Jet Pumps

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 recirculation pump flow differs by more than 15% from the established speed flow characteristics.

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%.

F. Jet Pump Flow Mismatch

1. Deleted.

G. Inservice Inspection

Inservice inspection shall be performed in accordance with the requirements for ASME Code Class 1, 2, and 3 components contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

jet pump body; however, the converse is not true. The lack of any substantial stress in the jet pump body makes failure impossible without an initial nozzle riser system failure.

F. Jet Pump Flow Mismatch

Requiring the discharge valve of the lower speed loop to remain closed until the speed of faster pump is equal to or less than 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.

G. Inservice Inspection

The inservice inspection program conforms to the requirements of 10 CFR 50, Section 50.55a(g). Where practical, the inspection of components conforms to the requirements of ASME Code Class 1, 2, and 3 components contained in Section XI of the ASME Boiler and Pressure Vessel Code. If a Code required inspection is impractical, a request for a deviation from that requirement is submitted to the Commission in accordance with 10 CFR 50, Section 50.55a(g)(6)(i).

Deviations which are needed from the procedures prescribed in Section XI of the ASME Code and applicable Addenda will be reported to the Commission prior to the beginning of each 10-year inspection period if they are known to be required at that time. Deviations which are identified during the course of inspection will be reported quarterly throughout the inspection period.

3.11 FUEL RODS

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 Figure 3.11-1 for two recirculation loop operation. For single-loop operation the values in these curves are reduced by 0.84 for 7x7 fuel, 0.86 for 8x8 fuel, 0.77 for 8x8R fuel and 0.77 for P8x8R fuel. If at any time during steady state operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

B. Linear Heat Generation Rate (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:

$$LHGR_{max} \leq LHGR_d \, [1 - \{(\Delta P/P)_{max} (L/LT)\}]$$

$LHGR_d$ = Design LHGR = G KW/ft.

$(\Delta P/P)_{max}$ = Maximum power spiking penalty = N

4.11 FUEL RODS

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 $\geq$ 25% rated thermal power.

B. Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% rated thermal power.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

LT = Total core length - 12 feet

L = Axial position above bottom of core

G = 18.5 kW/ft for 7x7 fuel bundles
= 13.4 kW/ft for 8x8 fuel bundles

N = 0.038 for 7x7 fuel bundles
= 0.0 for 8x8 fuel bundles

If at any time during steady state operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded action shall then be initiated to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

C. Minimum Critical Power Ratio (MCPR)

During steady state power operation the MCPR for each type of fuel at rated power and flow shall not be lower than the limiting value shown in Figure 3.11-2 for two recirculation loop operation. If, at any time during steady state 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 steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

For core flows other than rated the MCPR shall be the operating limit at rated flow times 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.

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 that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

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 clad temperature by less than $\pm$ 20°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 shown in Figure 3.11-1.

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 if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 5 of Reference 1 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 $\geq$ 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. Pellet densification power spiking in 8x8 fuel has been accounted for in the safety analysis presented in Reference 2; thus no adjustment to the LHGR limit for densification effects is required for 8x8 fuels.

C. Minimum Critical Power Ratio (MCPR)

The required operating limit MCPR's at steady state operating conditions as specified in Specification 3.11C are derived from the established fuel cladding integrity Safety Limit and an analysis of abnormal operational transients (Reference 2). 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 setting given in Specification 2.1.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the more limiting transients have been analyzed to determine which result in the largest reduction in critical power ratio (CPR). The models used in the transient analyses are discussed in Reference 1.

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 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.

3.11 Bases: (Cont'd)

The K_f factor curves shown in Figure 3.11-3 were developed generically which 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 1.

The K_f factors shown in Figure 3.11-3, are conservative for Cooper operation because the operating limit MCPR's are greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

References for Bases 3.11

1. Licensing Topical Report, General Electric Boiling Water Reactor, Generic Reload Fuel Application, (NEDE-24011-P), (most current approved submittal).

2. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1," (applicable reload document).

A&B. Average and Local LHGR

The LHGR 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 moved daily, a daily check of power distribution is adequate.

C. Minimum Critical Power Ratio (MCPR) - (Surveillance Requirement)

At core thermal power levels less than or equal to 25%, the reactor will be operating at less than or equal to minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation was made at 25% thermal power level with minimum recirculation pump speed. The MCPR margin was thus demonstrated such that subsequent MCPR evaluation below this power level was shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 94 TO FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By letters dated September 20 and 23, 1985, the Nebraska Public Power District (the licensee) requested emergency changes to the Cooper Nuclear Station (CNS) Technical Specifications to (1) permit reactor operation with one recirculation loop out of service, (2) to include General Electric Company's (GE) Service Information Letter (SIL) No. 380, Revision 1 recommendations regarding thermal-hydraulic stability concerns for dual loop and single loop operations, and (3) to incorporate administrative changes dealing with updating references and deleting blank pages. Presently, the CNS operating license requires the reactor to be in cold shutdown within the succeeding 24 hours if a recirculation loop becomes inoperable and cannot be returned to service. The licensee previously requested authorization for unlimited single loop operation of CNS. Subsequently, Tennessee Valley Authority's operation of Browns Ferry Unit 1 (a boiling water reactor similar in design to CNS) in the single loop mode of operation at 59% power lead to concerns related to thermal-hydraulic instability. GE, in SIL No. 380, Revision 1, addressed these concerns by providing the boiling water reactor licensees generic guidance for actions which suppress thermal-hydraulic instability induced neutron flux oscillations. The licensee has proposed Technical Specifications in accordance with the guidance provided by GE in SIL No. 380, Revision 1.

Specifically, the proposed changes requested by the licensee consist of (1) deletion of the license condition restricting the single loop operation; (2) for single and dual loop operation, incorporating requirements in the Technical Specifications to detect thermal-hydraulic instabilities induced neutron flux oscillations and specifying operator response to the detected instabilities; and (3) updating of some references and deletion of some blank pages. The change noted in (1) above involves a revision of the Technical Specifications 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 a revision to the allowable Average Planar Linear Heat Generation Rate (APLHGR) values.

2.0 EVALUATION

We have evaluated the licensee's proposal to permit unlimited operation of the CNS with one recirculation loop out of service, incorporate the GE SIL No. 380, Revision 1 guidance regarding thermal-hydraulic instabilities and implement some administrative changes in the CNS Technical Specifications.

2.1 Single Loop Operation

We have reviewed the licensee's analysis of accidents and transients which are judged to be affected by operation of the CNS in the single operating loop mode. The accidents and transients which are of concern relate to inadvertent variations in the coolant flow through the core and design bases of the fuel performance safety limits. The events evaluated include One Pump Seizure Accident, Idle Loop Startup Event, Rod Withdrawal Error Event, and Loss-of-Coolant Accident.

One Pump Seizure Accident

The licensee states that the one pump seizure accident is a relatively mild event during two recirculation pump operation. Similar analyses were performed to determine the impact this accident would have on one recirculation pump operation. These analyses were performed using NRC approved models for a large core BWR/4 plant. The analyses were conducted from steady-state operation at the following initial conditions, with the added condition of one inactive recirculation loop. Two sets of initial conditions assumed were:

a. Thermal Power = 75% and core flow = 58% of rated
b. Thermal Power = 82% and core flow = 56% of rated

These conditions were chosen because they represent reasonable upper limits of single loop operation within existing Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Minimum Critical Power Ratio 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:

a. The recirculation loop flow in the loop in which the pump seizure occurs drops instantaneously to zero.
b. Core voids increase which result in a negative reactivity insertion and a sharp decrease in neutron flux.
c. Heat flux drops more slowly because of the fuel time constant.
d. 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 licensee concludes that MCPR 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 CNS , and were obtained with the staff approved methodology. We, therefore, agree with the licensee's conclusion that fuel cladding integrity safety margins will not be reduced.

Idle Loop Startup

The idle loop startup transient was analyzed, in the CNS Final Safety Analysis Report (FSAR) for dual loop operation. For single loop operation, the licensee proposed to increase the rated condition steady-state MCPR limit by 0.01 to account for increased uncertainties in the core total flow and Traversing In-core Probe (TIP) readings. The staff found the MCPR increase of 0.01 to be acceptable. The MCPR will also vary depending on flow conditions. This leads to the possibility of a large inadvertent flow increase which could cause the MCPR to decrease below the Safety Limit for a low initial MCPR at reduced flow conditions. Therefore, the required MCPR must be increased at reduced core flow by a flow factor, K_f derived by assuming both recirculation loops increase speed to the maximum permitted by the scoop tube position set screws. This condition maximizes the power increase and hence the MCPR for transients initiated from less than rated conditions. When operating on one loop the flow and power increase will be less than associated with two pumps increasing speed, therefore, the K_f factors derived from the two-pump assumption are conservative for single loop operation.

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 transients, the rod block system will stop rod withdrawal at a minimum critical power ratio which is higher than the fuel cladding integrity safety limit. The proposed correction of the rod block equation and lower initial power for single loop operation will assure that the MCPR safety limit will not be violated.

One pump operation results in backflow through 10 of the 20 jet pumps while flow is being supplied to the lower plenum from the active jet pumps. Because of this backflow through the inactive jet pumps the present rod-block equation and APRM settings must be modified. The licensee has proposed modified rod block equation and APRM settings in the Technical Specification for one pump operation, and the staff has found them acceptable.

Loss-of-Coolant Accident (LOCA)

The licensee has contracted General Electric Company (GE) to perform single loop operation analysis for CNS LOCA. The licensee states that evaluation of these calculations (that are performed according to the procedure outlined in NEDO-20566-2, Rev. 1) indicates that a multiplier of 0.84 (7 X 7 fuel), 0.84 (8 X 8 fuel), 0.77 (8 X 8R fuel) (Ref.: NEDE-24258, May 1980)) should be applied to the MAPLHGR limits for single loop operation of the CNS.

We find the use of MAPLHGR multipliers as indicated will be adequate to offset LOCA consequences in the single loop operation mode. The MAPLHGR factors are, therefore, acceptable.

2.2 Thermal-Hydraulic Stability in Dual and Single Loop Operation

We have evaluated the licensee's proposed Technical Specification changes to assure that the changes provide adequate detection and suppression of potential thermal-hydraulic instabilities.

GE recently presented the staff with stability test data which demonstrated the occurrence of limit cycle neutron flux oscillations at natural circulation and several percent above the rated rod line. The oscillations were observable on the APRMs and were suppressed with control rod insertion. It was predicted that limit cycle oscillations would occur at the operating condition tested; however, the characteristics of the observed oscillations were different from those previously observed during other stability tests. Namely, the test data showed that some LPRM indications oscillated out of phase with the APRM signal and at amplitude as great as six times the core average. GE has prepared and released a service information letter, SIL No. 380, to alert the BWR owners of these new data and to recommend actions to avoid and control abnormal neutron flux oscillations.

The General Electric recommendations were reviewed by the staff and found to be prudent recommendations which provide adequate detection and suppression of potential thermal-hydraulic instabilities as required by General Design Criteria (GDC) 10 and 12. The staff compared these recommendations with the CNS Technical Specifications for operation with a recirculation loop out of service and found that the proposed changes are in conformance with the SIL No. 380, Revision 1 recommendations and are acceptable to the staff.

In addition, on February 9, 1985 a single loop test was performed by Tennessee Valley Authority (TVA) on its Browns Ferry Unit 1 reactor during which thermal-hydraulic stability decay ratios were measured. The main findings of the test were that the observed increase in neutron noise during single loop operation (SLO) is solely due to an increase in flow noise because the inlet flow to power transfer functions during two loop operation (TLO) and SLO are not significantly different when test plateaus with similar power and flow conditions are compared. The Browns Ferry Unit 1 reactor has been found to be stable in all modes of operation attained during the present tests. The most unstable test plateaus corresponded to minimum recirculation pump speed in SLO, which has the minimum flow and maximum power to flow ratio. The estimated decay ratio at this plateau was 0.53. The decay ratio decreased as the flow was increased during SLO (down to 0.34). This implies that the core-wide reactor stability follows the same trends in SLO as it does in TLO. Finally, no local or higher mode instabilities were found in the data taken from local power range monitors (LPRMs). The decay ratios estimated from LPRMs were not significantly different than the ones estimated from the average power range monitors.

In conclusion, the measured decay ratios at Browns Ferry Unit 1 showed the plant to have adequate stability margin over a range of power/flow conditions which are of concern during single loop operation. Since the Cooper Nuclear Station maximum calculated decay ratio (.86) is similar to Browns Ferry Unit 1 (.87), and since it was shown that the stability characteristics of SLO are similar to TLO this test provides additional justification to allow single loop operation at CNS.

2.3 Administrative Changes

The licensee has proposed to update some references and delete blank pages in the Technical Specifications. The staff finds the proposed administrative changes acceptable.

3.0 EMERGENCY CIRCUMSTANCES

At approximately 0900 CDT on September 20, 1985, the Recirculation Pump B was tripped and Cooper Nuclear Station began single loop operation. The recirculation pump was tripped after it was determined that a low insulation reading to ground existed in the recirculation pump motor-generator set windings. The Cooper Nuclear Station Technical Specifications contain a limiting condition for operation (LCO) which requires the reactor to be shutdown if a recirculation loop is out of service for 24 hours. The licensee determined that approximately two weeks would be required to restore Recirculation Pump B to service. On September 20, 1985, the licensee informed the staff of the conditions at CNS and its decision to file an expedited license amendment request which would permit single loop operation for an indefinite period and thus avoid reactor shutdown as a result of the LCO.

By letter dated September 20, 1985, the licensee proposed an expedited Technical Specification change which would remove the 24-hour LCO. The licensee also proposed to add surveillance requirements relative to thermal-hydraulic instability which would justify deletion of the LCO. After discussion with the staff the licensee provided revisions to the original amendment application and documented these changes by letter dated September 23, 1985.

3.1 No Significant Hazards Consideration Determination

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations if operation of the facility in accordance with the amendment would not:

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

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

The information in Section 2.0 above provides the basis for evaluating this license amendment against these criteria. Since the requested operational mode is acceptable and the plant operating conditions, the physical status of the plant, and dose consequences of potential accidents are the same as without the requested change, the staff concludes that:

(1) Operation of the facility in accordance with the amendment would not significantly increase the probability or consequences of an accident previously evaluated because the types of accidents most likely to occur with single loop operation have been evaluated and formed to satisfy the Commissions regulations . In addition, the amendment would add more restrictive limits and surveillance requirements to ensure that the consequences and probabilities would not be increased.

(2) Operation of the facility in accordance with the amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated because all abnormal operating transients which could be initiated with single loop operation, such as an inadvertent startup of an idle recirculation pump or pump trip have already been analyzed in the FSAR and reviewed and accepted by the staff. The additions of thermal-hydraulic instability surveillance requirements involve normal plant operating practices and, therefore, are not expected to create a new or different kind of accident.

(3) Operation of the facility in accordance with the amendment would not involve a significant reduction in a margin of safety because any decrease in margin resulting from single loop operation would be offset be the more stringent operating limits and surveillance requirements that are also added by the amendment.

Accordingly, we conclude the amendment to Facility Operating License No. DPR-46, permitting single loop operation for greater than 24 hours, involves no significant hazards consideration.

3.2 State Consultation

In accordance with the Commission's regulations, consultation was held with the State of Nebraska by telephone. The State expressed no concern either from the standpoint of safety or of no significant hazards consideration determination, in view of the interim nature of the amendment and the compensatory measures.

4.0 ENVIRONMENTAL CONSIDERATIONS

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has made a final no significant hazards consideration finding with respect to this amendment. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Dated: September 24, 1985