DOCKET FILE

Docket No. 50-298 November 29, 1991

Mr. Guy R. Horn
Nuclear Power Group Manager
Nebraska Public Power District
Post Office Box 499
Columbus, Nebraska 68602-0499

Dear Mr. Horn:

SUBJECT: COOPER NUCLEAR STATION - AMENDMENT NO. 151 TO FACILITY OPERATING LICENSE NO. DPR-46 (TAC NO. M81164)

The Commission has issued the enclosed Amendment No.151 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. The amendment consists of changes to the Technical Specifications in response to your application dated July 29, 1991, as supplemented October 3, 1991.

The amendment changes the Technical Specifications to allow plant operation in the plant operating domain on the power/flow map above the rated rod line and to incorporate improvements in the average power range monitor and rod block monitor systems.

A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY
Roby B. Bevan, Project Manager
Project Directorate IV-1
Division of Reactor Projects III, IV, and V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 151 to License No. DPR-46
2. Safety Evaluation

cc w/enclosures:
See next page

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C40097

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

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RBBevan

Roby B. Bevan, Project Manager
Project Directorate IV-1
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 151
License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District (the licensee) dated July 29, 1991, as supplemented October 3, 1991, 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 license 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 Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 151, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John T. Larkins, Director
Project Directorate IV-1
Division of Reactor Projects III, IV, and V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 29, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 151

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 pages are identified by Amendment number and contain vertical lines indicating the area of change.

REMOVE PAGES	INSERT PAGES
6	6
7	7
8	8
18	18
19	19
27	27
28	28
31	31
42	42
43	43
61	61
--	61a
62	62
62a	62a
77	77
86	86
96	96
102	102
104	104
210	210
211	211
212	212
214	214
214a	214a
214b	214b

SAFETY LIMITS

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 ≥800 psia and Core Flow ≥10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.06 for two recirculation loop operation (1.07 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.

LIMITING SAFETY SYSTEM SETTINGS

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)

(1) Flow Referenced Scram Trip Setting

When the Reactor Mode Selector is in the RUN position, the APRM flow referenced flux scram trip setting shall be:

$S \leq 0.58\ W + 62\% - .58\ \Delta W$

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% core flow 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)

ΔW = 0 for two recirculation loop operation.

(2) Fixed APRM Flux Scram Trip Setting

The fixed APRM flux scram trip setting shall not be allowed to exceed 120% of rated thermal power.

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

When the Reactor Mode Selector 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:

$S_{RB} \leq 0.58\ W + 50\% - .58\ \Delta W$

with a maximum of ≤ 108% or rated power.

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.

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

≥ +12.5 in. on vessel level instruments.

2.1 Bases: (Cont'd)

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.

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.

2.1 Bases (Cont'd)

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.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

To assure the operability of the reactor protection system.

Specification:

The setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.1.

4.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

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

Specification:

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

B. Deleted.

C. Deleted.

D. When it is determined that a channel has failed in the unsafe condition, the other RPS channels that monitor the same variable shall be functionally tested immediately before the trip system containing the failure is tripped. The trip system containing the unsafe failure may be placed in the untripped condition during the period in which surveillance testing is being performed on the other RPS channels.

COOPER NUCLEAR STATION
TABLE 3.1.1
REACTOR PROTECTION SYSTEM INSTRUMENTATION REQUIREMENTS

Reactor Protection System Trip Function	Applicability Conditions Mode Switch Position				Trip Level Setting	Minimum Number of Operable Channels Per Trip Systems (1)	Action Required When Equipment Operability is Not Assured (1)
	Shutdown	Startup	Refuel	Run			
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)	≤ 120/125 of indicated scale	3	A
Inoperative		X	X	(5)		3	A
APRM (17) High Flux (Flow biased)				X	≤ (0.58W+62%-0.58ΔW) (14)(19) with a maximum of 120% of rated power	2	C
High Flux (fixed)				X	≤ 120% Rated Power	2	A
High Flux	X(7)	X(9)	X(9)	(16)	≤ 15% Rated Power	2	A
Inoperative		X(9)	X(9)	X	(13)	2	A
Downscale	(12)	(12)	(12)	X(11)	≥ 2.5%	2	A
High Reactor Pressure NBI-PS-55 A,B,C, & D		X(9)	X(10)	X	≤ 1045 psig	2	A
High Drywell Pressure PC-PS-12 A,B,C, & D		X(9)(8)	X(8)	X	≤ 2 psig	2	A or D
Reactor Low Water Level NBI-LIS-101 A,B,C, & D		X	X	X	≥ + 4.5 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	≤ 92 inches	3 (18)	A

11. The APRM downscale trip function is only active when the Reactor Mode Selector Switch is in RUN. When in RUN, this function is automatically bypassed when the companion IRM instrumentation is operable and not upscale.

12. The APRM downscale trip is automatically bypassed when the Reactor Mode Selector 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 when the Reactor Mode Selector Switch is in RUN.

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

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 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.

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COOPER NUCLEAR STATION
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)	$\leq$ (0.58W + 50% - 0.58 ΔW) (2)(13) with a maximum of $\leq$ 108% of rated power	2(1)
APRM Upscale (Startup)	$\leq$ 12%	2(1)
APRM Downscale (9)	$\geq$ 2.5%	2(1)
APRM Inoperative	(10b)	2(1)
RBM Upscale (Power Referenced):	(14) (16)	1(15)
RPM Power Range:	(14)	Not Applicable (15)
Low Power Setpoint (LPSP)	$\leq$ 30% of rated power	
Intermediate Power Setpoint (IPSP)	$\leq$ 65% of rated power	
High Power Setpoint (HPSP)	$\leq$ 85% of rated power	
RBM Downscale (9)	$\geq$ 91/125 (16)	1(15)
RBM Inoperative	(10c) (15)	1(15)
IRM Upscale (8)	$\leq$ 108/125 of Full Scale	3(1)
IRM Downscale (3)(8)	$\geq$ 2.5/125 of Full Scale	3(1)
IRM Detector Not Full In (8)		3(1)

Lowest Rated MCPR Limit	LTSP (LPSP $\leq$ P < IPSP)	ITSP (IPSP $\leq$ P < HPSP)	HTSP (HPSP $\leq$ P)
$\geq$ 1.20	$\leq$ 117.0/125	$\leq$ 111.2/125	$\leq$ 107.4/125
$\geq$ 1.25	$\leq$ 120.0/125	$\leq$ 115.2/125	$\leq$ 110.2/125
$\geq$ 1.30	$\leq$ 123.0/125	$\leq$ 118.0/125	$\leq$ 113.2/125

COOPER NUCLEAR STATION
TABLE 3.2.C (page 2)
CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

Function	Trip Level Setting	Minimum Number Of Operable Instrument Channels/Trip System(5)
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 rod blocks need not be operable in the RUN mode, and the APRM (flow biased) 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 Instrument tion, 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).

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.

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 Reactor Mode Selector Switch is placed in RUN.

9. This function is only active when the Reactor Mode Selector Switch is in RUN.

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
        (4) Loss of negative supply voltage

NOTES FOR TABLE 3.2.C (Continued)

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.

14. One set of power referenced RBM upscale trip settings (LTSP, ITSP, and HTSP) is applied based on the lowest rated MCPR limit given in Specification 3.11.C. The RBM power range setpoints control the enforcement of the appropriate upscale trips over the proper core thermal power range as follows:

a. All RBM trips are automatically bypassed below the Low Power Setpoint (LPSP).

b. The upscale Low Trip Setpoint (LTSP) is applied at the LPSP and up to the Intermediate Power Setpoint (IPSP).

c. The upscale Intermediate Trip Setpoint (ITSP) is applied at the IPSP up to the High Power Setpoint (HPSP).

d. The upscale High Trip Setpoint (HTSP) is applied at and above the HPSP.

15. The RBM is only required when core thermal power is $\geq$ 30% of rated power and a limiting control rod pattern (defined in Specification 3.3.B.5) exists. Requirements for operating with a limiting control rod pattern are specified in Specification 3.3.B.5.b.

16. RBM trip level settings are relative to the initialization reference signal of 100/125 of full scale which takes place upon control rod selection.

TABLE 4.2.C
SURVEILLANCE REQUIREMENTS FOR ROD WITHDRAWAL BLOCK INSTRUMENTATION

Function	Functional Test Freq.	Calibration Freq.	Instrument Check
APRM Upscale (Flow Bias)	(1) (3)	Once/3 Months	Once/Day
APRM Upscale (Startup Mode)	(1) (3)	Once/3 Months	Once/Day
APRM Downscale	(1) (3)	Once/3 Months	Once/Day
APRM Inoperative	(1) (3)	N.A.	Once/Day
RBM Upscale (Power Referenced)	(1) (3)	Once/6 Months	Once/Day
RBM Power Range	(3)	Once/6 Months	N.A.
RBM Downscale	(1) (3)	Once/6 Months	Once/Day
RBM Inoperative	(1) (3)	N.A.	Once/Day
IRM Upscale	(1) (2) (3)	Once/3 Months	Once/Day
IRM Downscale	(1) (2) (3)	Once/3 Months	Once/Day
IRM Detector Not Full In	(2) (Once/operating cycle	Once/Oper. Cycle (10)	Once/Day
IRM Inoperative	(1) (2) (3)	N.A.	N.A.
SRM Upscale	(1) (2) (3)	Once/3 Months	Once/Day
SRM Downscale	(1) (2) (3)	Once/3 Months	Once/Day
SRM Detector Not Full In	(2) (Once/operating cycle	Once/Oper. Cycle (10)	N.A.
SRM Inoperative	(1) (2) (3)	N.A.	N.A.
Flow Bias Comparator	(1) (8)	Once/Oper. Cycle	N.A.
Flow Bias Upscale	(1) (8)	Once/3 Months	N.A.
Rod Block Logic	(9)	N.A.	N.A.
RSCS Bypass	(1)	Once/3 Months	N.A.
SDV High Water Level	Quarterly	Once/Oper. Cycle	N.A.

3.2 BASES (cont'd.)

C. Control Rod Block Actuation

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to the safety limit CPR. The trip logic for this function is 1 out of n: e.g., any trip on one of six APRM's, eight IRM's, or four SRM's will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This time period is only 3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

The APRM rod block function is flow biased 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 CPR.

The RBM rod block function provides local protection of the core; i.e., the prevention of the MCPR reaching the safety limit CPR in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern. Additional details are provided in Bases Section 3.3.B.5.

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.

A downscale indication on 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 2.5 indicated on scale.

The flow comparator and scram discharge volume high level components have only one logic channel and are not required for safety. The SDV high level rod block does provide adequate time to determine the cause of the level increase and take corrective action prior to automatic scram.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

D. Radiation Monitoring Systems - Isolation and Initiation Functions

1. Steam Jet Air Ejector Off-Gas System

Two air ejector off-gas monitors are provided and when their trip point is reached, cause an isolation of the air ejector off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip. There is a fifteen minute delay accounted for by the 30-minute holdup time of the off-gas before it reaches the stack.

LIMITING CONDITION FOR OPERATION

3.3.B.3 (cont'd)

e. If Specifications 3.3.B.3a through d cannot be met, the reactor shall not be started, or if the reactor is in the run or startup modes at less than 20% rated power, it shall be brought to a shutdown condition immediately.

f. The sequence restraints imposed on the control rods may be removed by the use of the individual rod position bypass switches for scram testing only those rods which are fully withdrawn in the 100% to 50% rod density range.

4. Control rods shall not be withdrawn for startup unless at least two source range channels have an observed count rate equal to or greater than three counts per second.

5. Operation with a Limiting Control Rod Pattern (for Rod Withdrawal Error, RWE)

a. A Limiting Control Rod Pattern for RWE exists when either:

1) Core thermal power is $\geq$ 30% and $<$ 90% of rated power and the MCPR is less than 1.70, or

2) Core thermal power is $\geq$ 90% of rated power and the MCPR is less than 1.40.

b. During operation with a Limiting Control Rod Pattern for RWE either:

1) Both rod block monitor (RBM) channels shall be operable, or

2) With one RBM channel inoperable, control rod withdrawal shall be blocked within 24 hours, or

3) With both RBM channels inoperable, control rod withdrawal shall be blocked until operability of at least one channel is restored.

SURVEILLANCE REQUIREMENT

4.3.B.3.b (cont'd)

1) The correctness of the Banked Position Withdrawal Sequence input to the RWM computer shall be verified.

2) The RWM computer on line diagnostic test shall be successfully performed.

3) Proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group shall be verified.

4) The rod block function of the RWM shall be verified by withdrawing the first rod as an out-of-sequence control rod no more than to the block point.

c. When required, the presence of a second licensed operator or other qualified employee to verify the following of the correct rod program shall be verified.

4. Prior to control rod withdrawal for startup, verify that at least two source range channels have an observed count rate of at least three counts per second.

5. Operation with a Limiting Control Rod Pattern (for Rod Withdrawal Error, RWE)

During operation when a Limiting Control Rod Pattern for RWE exists and only one RBM channel is operable, an instrument functional test of the RBM channel shall be performed prior to withdrawal of the control rod(s). A Limiting Control Rod Pattern for RWE is defined by Specification 3.3.B.5.

3.3 and 4.3 BASES: (Cont'd)

5. The RBM provides local protection of the core; i.e., the prevention of boiling transition in a local region of the core, for a single rod withdrawal error from a Limiting Control Rod Pattern as defined in Specification 3.3.B.5. The trip point is referenced to a power signal provided by the APRMs. A statistical analysis (Reference 5) of many single control rod withdrawal errors has been performed. At the 95/95 level the results show that with the specified trip settings, rod withdrawal is blocked at MCPRs which are greater than Safety Limit 1.1.A, thus allowing adequate margin. This analysis assumed operation at steady state operating limit MCPRs (Specification 3.11.C) prior to the postulated rod withdrawal error. The RBM functions are required when core thermal power is greater than 30% and a Limiting Control Rod Pattern (Specification 3.3.B.5) exists. When both RBM channels are operating either channel will assure required withdrawal blocks occur even assuming a single failure of one channel. When a Limiting Control Rod Pattern exists, with one RBM channel inoperable for no more than 24 hours, testing of the RBM before withdrawal of control rods assures that improper control rod withdrawal will be blocked.

   A Limiting Control Rod Pattern for rod withdrawal error (RWE) exists when (a) core thermal power is greater than or equal to 30% of rated and less than 90% of rated ($30\% \leq P < 90\%$) and the MCPR is less than 1.70, or (b) core thermal power is greater than or equal to 90% of rated ($P \geq 90\%$) and MCPR is less than 1.40.

   Under these conditions, complete withdrawal of a control rod could result in MCPR violating the safety limits. Therefore, RBM operation is required to block control rod withdrawal. RBM setpoints have been selected such that the required control rod blocks shall occur even if one of the redundant RBM channels fails. Should one RBM channel become inoperable during the use of such patterns, it is judged that testing of the RBM system before withdrawal of such rods assures the remaining channel operability.

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. The limiting power transient is defined in Reference 2. Analysis of this transient shows that the negative reactivity rates resulting from the scram provide the required protection, and MCPR remains greater than the safety limit.

The surveillance requirement for scram testing of all the control rods after each refueling outage and 10% of the control rods at 16-week intervals is adequate for determining the operability of the control rod system yet is not so frequent as to cause excessive wear on the control rod system components.

The numerical values assigned to the predicted scram performance are based on the analysis of data from other BWR's with control rod drives the same as those on Cooper Nuclear Station.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of a systematic problem with control rod drives.

In the analytical treatment of the transients which are assumed to scram on high neutron flux, 290 milliseconds are allowed between a neutron sensor reaching the scram point and start of motion of the control rods. This is adequate and conservative when compared to the typical time delay of about 210 milliseconds estimated from scram test results. Approximately the first 90 milliseconds of each of these time intervals result from the sensor and circuit delays; at this point, the pilot scram solenoid deenergized. Approximately 120 milliseconds later,

3.3 and 4.3 BASES: (Cont'd)

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. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-(latest approved revision).

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

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

4. General Electric Service Information Letter No. 316. Reduced Notch Worth Procedure, November, 1979.

5. "Extended Load Line Limit and ARTS Improvement Program Analysis for Cooper Nuclear Station Cycle 14, "NEDC-31892P, Revision 1, May 1991.

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, with both recirculation loops operating, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value specified in the Core Operating Limits Report for two recirculation loop operation. For single-loop operation these values are reduced for each fuel type as specified in the Core Operating Limits Report. 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, reduce reactor power to $\leq$ 25% of rated power within the next four (4) hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

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.

LIMITING CONDITIONS FOR OPERATION

3.11.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 specified in the Core Operating Limits Report.

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.

SURVEILLANCE REQUIREMENTS

4.11.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

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 specified in the Core Operating Limits Reports 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, reduce reactor power to $\leq$ 25% of rated power within the next four (4) hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

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

SURVEILLANCE REQUIREMENTS

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at > 25% rated thermal power and following any change in power level or distribution that could cause operation on the operating limit MCPR.

3.11 BASES

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 10CFR50.46.

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 ± 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 10CFR50.46 limit. The limiting value for APLHGR for each fuel type is specified in the Core Operating Limits Report.

The flow dependent correction factor $MAPFAC_F$ specified in the Core Operating Limits Report is applied to the MAPLGHR limits at rated conditions to assure that (1) the 10CFR50.46 limit would not be exceeded during a LOCA initiated from less than rated core flow conditions and (2) the fuel thermal mechanical design criteria would be met during abnormal operating transients initiated from less than rated core flow conditions (Reference 11).

The power dependent correction factor $MAPFAC_P$ specified in the Core Operating Limits Report is applied to the MAPLHGR limits at rated conditions to assure that the fuel thermal mechanical design criteria would be met during abnormal operating transients initiated from less than rated core power conditions (Reference 11).

The APLHGR values are reduced for single loop operation per Reference 10.

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 LHGR as a function of core height shall be checked daily during reactor operation at ≥ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern. Pellet densification power spiking in GE fuel has been accounted for in the safety analysis presented in References 1 and 2; thus no adjustment to the LHGR limit for densification effects is required.

3.11 Bases: (Cont'd)

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 and 11). 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.

Flow-dependent and power-dependent MCPR limits ($MCPR_F$ and $MCRP_P$) are used to define the required Operating Limit MCPR (OLCPR) such that the above Safety Limit MCPR requirement is met for all power/flow conditions. $MCRP_F$ provides the thermal margin required to protect the fuel from transients resulting from inadvertent core flow increases. $MCRP_P$ protects the fuel from the other limiting abnormal operating transients, including localized events such as a rod withdrawal error.

Direct scram on Turbine Stop Valve Closure or Turbine Control Valve fast closure provides the fastest response to an abnormal operating transient such as load rejection, turbine trip, or feedwater controller failure. These direct scrams are bypassed at low power (P_{bypass}), to reduce the frequency of scrams during power ascension. For operation at or above P_{bypass} (30% of rated power), the required OLMCPR is the larger of $MCPR_F$ or $MCRP_P$ at the existing core power/flow state; where $MCPR_F$ and $MCRP_P$ are determined in the Core Operating Limits Report by multiplying the scram time dependent MCPR limit for rated power and flow MCPR(100) by the K_P factor. Below 30% of rated power, when the direct scrams are bypassed, a slightly more severe transient response results. To compensate for the more severe transient response, two power dependent MCPR limits are established, one for high flow (>50% of rated) conditions and one for low flow (≤50% of rated) conditions. These limits are specified in the Core Operating Limits Report. Further information on the MCPR operating limits for off-rated conditions is presented in Reference 11.

References for Bases 3.11

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A. (The approved revision at the time the reload analyses are performed.) The approved revision number shall be identified in the Core Operating Limits Report.

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

3-8. Deleted

9. Letter (with attachment), R. H. Buckholz (GE) to P. S. Check (NRC), "Response to NRC Request for Information on ODYN Computer Model," September 5, 1980.

10. "Cooper Nuclear Station Single-Loop Operation," NEDO 24258.

11. "Extended Load Limit and ARTS Improvement Program Analysis for Cooper Nuclear Station Cycle 14," NEDC-31892P, Revision 1, May 1991.

4.11 Bases:

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 an operating limit MCPR 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

RELATED TO AMENDMENT NO. 151 TO FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By letter dated July 29, 1991 (Ref. 1), the Nebraska Public Power District (the licensee) submitted a proposal to change the Technical Specifications (TS) of the Cooper Nuclear Station (CNS). Additional information provided by the licensee by letter dated October 3, 1991 (Ref. 2) was a clarification of previously provided information and did not change the action noticed or affect the initial proposed no significant hazards considerations determination.

The changes are required to provide for implementation of an extended operating region of the power/flow map and implementation of the average power range monitor (APRM), rod block monitor (RBM), Technical Specification (ARTS) Improvement Program. The ARTS improvements are intended to increase the plant operating efficiency, update the compliance with the thermal margin requirements, improve the accuracy and response of the pertinent instrumentation, and to improve the man/machine interface.

The licensee has provided Extended Load Line Limit (ELLL) and ARTS Improvement analyses in General Electric Topical Report NEDC-31892P (Ref. 3) which is provided as Attachment 2 to Reference 1. The ELLL analysis is provided as a basis for normal reactor operation in the region of the power/flow map above the rated rod line. Operation in the extended region is achieved by changing the slope of the flow bias scram trip algorithm and revising the APRM rod block line. The ARTS improvements involve (1) elimination of APRM trip setdown requirements, (2) power and flow dependent limits on Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Minimum Critical Power Ratio (MCPR), (3) changes from flow to power referenced setpoints for the RBM, (4) an altered rod withdrawal error at power analysis, and (5) changes in normalization procedure and new trip logic in the RBM providing definition of a limiting rod pattern for RBM bypass decisions.

In support of the request the licensee has submitted (1) a description and evaluation of the changes, (2) the proposed new TS, (3) an identification of changes to the Core Operating Limits Report, and (4) a General Electric Company (GE) analysis of the consequences of operation in an extended operating domain (Ref. 3) to justify the proposed changes for CNS. The licensee's submittal included proposed Table changes to the CNS TS relating to APRM reactor protection system instrumentation requirements and control rod withdrawal block instrumentation.

2.0 EVALUATION

2.1 Elimination of Flow-Biased APRM Scram and Rod Block Trip Setpoint Setdown Requirement

In the current CNS Technical Specifications the flow-biased APRM scram and rod block trip setpoints are reduced (set down) when the core Total Peaking Factor (TPF) is greater than the design TPF. This requirement is associated with a now obsolete Hench-Levy Minimum Critical Heat Flux Ratio criterion. GE has since developed the General Electric Thermal Analysis Basis (GETAB/GEXL) critical power correlation, which allows an acceptable alternative to the setdown requirement. The GE analysis (Ref. 3) enclosed with the CNS submittal includes results from analyses made to determine a set of power and flow dependent operating limits (MCPR and MAPLHGR) that would be needed to satisfy the pertinent licensing criteria if APRM setdown was eliminated. The new limits should (1) prevent violation of the MCPR safety limit, (2) keep the fuel thermal-mechanical performance within the design and licensing basis, and (3) keep peak cladding temperature and maximum cladding oxidation within allowable limits. The results of the analyses with approved analytical methods are stated below.

In order to restore safety margins which might be reduced when the APRM setdown is removed, the limiting transients were reanalyzed assuming the absence of this feature. The analyses assumed operation within the proposed extended power/flow domain with flows up to 100 percent of rated flow. Analyses of the transient events were made as a function of initial power and flow and the results used to determine multipliers to be applied to full power-full flow values of MCPR and MAPLHGR. The power dependence was most sensitive at full flow and the feedwater controller failure was the transient showing the largest sensitivity. This event was then used to construct a curve of MCPR multiplier, K_p, and MAPLHGR multiplier, $MAPFAC_p$, as a function of core power. Cycle-specific curves will be placed in the Core Operating Limits Report (COLR) for CNS.

Flow dependence of MCPR and MAPLHGR was determined from analyses of flow runout events in which the core flow is ramped upward to the maximum value permitted by the setting of the recirculation pump scoo; The flow multipliers, MCPR(F) and $MAPFAC_f$, are thus a function of the initial flow and the maximum flow and a family of curves is drawn. The multipliers are chosen so that a flow runout to the maximum flow will not result in a violation of MCPR or linear heat generation rate (LHGR) safety limits. The $MAPFAC_f$ curves are combined with the results of loss of coolant accident (LOCA) analysis described in the following Section 2.2 and the combined family of curves is used in the COLR. In connection with this amendment request, the new flow dependent MCPR factors will replace the previously used MCPR multiplier, K_f, which will no longer be utilized.

The above discussion applies to the power range from 30 to 100 percent of full power. Below 30 percent of full power the turbine stop and control valve scrams are bypassed and the analyses do not apply. Below 25 percent of full

power no MCPR and MAPLHGR limits are defined. In the interval between 25 and 30 percent of full power, flow dependent effects are taken into account by having two power dependent curves - one for flows greater than 50 percent of rated and one for lower flows. Analyses are then performed to obtain limiting MCPR and MAPLHGR values in these domains.

The development of the adjustment factors described above used the ARTS improvement program, which, in part, supports both the implementation of updated operating limits and the elimination of the APRM trip setdown requirements. The transient analysis results discussed in the following Safety Evaluation (SE) Section 2.2 are a part of the ARTS program. The ARTS concept has been used over the past few years successfully and with staff approval by a number of utilities having boiling water reactors (BWRs). Its use in the present amendment request is acceptable to the staff. We also conclude that deletion of the APRM setdown requirement is acceptable when it is replaced by the power and flow dependent operating limits described above.

2.2 Abnormal Operational Transient

All core-wide anticipated operational occurrences (AOO) of the Cooper licensing basis were considered for the ELLL. The transients re-evaluated were generator load reduction without turbine bypass (LRNBP), feedwater controller failure maximum demand (FWCF), and inadvertent high pressure coolant injection (HPCI) events. The potentially limiting LRNBP and FWCF events were re-evaluated at the power/flow conditions corresponding to the ELLL bounding point (100% power/87% flow) and rated conditions (100% power/100% flow). The HPCI event was analyzed at both points with an additional 2% power uncertainty allowance as prescribed by the use of the approved GEMINI methodology. These are presented for each fuel type presently loaded in the CNS core and are compared with the values for standard operating conditions. All transient analyses were done with approved methodology (Ref. 4).

GE has also examined other events and affected system components related to the requested extension. These include overpressure protection, thermal-hydraulic stability, LOCA events, and pressure differentials on reactor internals. The results show that design limits will not be exceeded. The containment LOCA response was analyzed and the results show no significant impact of the ELLL. For thermal-hydraulic stability, Cooper Station is operating under the "Interim Recommendations for Stability Actions" of the NRC Bulletin No. 88-07, Supplement 1. These operations are applicable to the CNS operations with ELLL/ARTS, as is the case for other operating reactors with similar systems.

An evaluation of the standard LOCA methodology (Ref. 5) has established that, when the ARTS improvement program is considered, the full power MAPLHGR limits should be multiplied by a factor of 0.86 at or below 65% core flow to ensure compliance with 10 CFR 50.46 PCT limits. This factor will be incorporated in the new $MAPFAC_p$ curves for CNS. The staff review of these various GE examinations has concluded that suitable analyses were performed and the results are comparable to other reviews and are acceptable for CNS.

2.3 Modification of Flow-Biased APRM Scram and Rod Block Trip Equations

The ELLL proposal changes the APRM flux scram lines on the power/flow map and permits operation up to the new APRM flux scram line (0.58W + 62%) and up to the intersection with the 100 percent power line occurring at a flow of 87 percent. This is a standard change for ELLL previously approved at other facilities and is acceptable. The flow-biased rod block trip equation is changed to 0.58W + 50% with a maximum value of 108%.

2.4 Modification of RBM Trip Setpoints and RBM System Operability Requirements

The RBM system is used to prevent withdrawal of a control rod under conditions which could lead to violation of the fuel integrity safety criteria. A modified RBM system configuration is described in Reference 3 and will be implemented during the upcoming refueling outage for CNS. The process of defining RBM operating requirements entails replacing the present deterministic bounding cycle specific analysis with a statistical analysis that forms the basis for the modified system. This modification is made in coordination with the ARTS program and allows the selection of the RBM setpoint such that the RWE analysis results are bounded by the limiting transient analysis (Section 2.2).

The following summarizes the RBM changes which have been considered in past reviews and the current Cooper proposal.

(1) The RBM is used to block rod motion to prevent exceeding MCPR limits during control rod withdrawal at power. For ARTS there is an optimization in the assignment of LPRMs to RBM channels. This change provides better and more uniform sensitivity to rod motion.

(2) ARTS has a more direct trip logic including a calibration to a fixed reference upon selection of a control rod for movement rather than calibration to the APRM, and an upscale trip level which is a step function of core power rather than a flow biased trip. Upon selecting a control rod for movement, each of two RBM channels automatically computes the average of all assigned local power range monitor channels. The average signal is then processed to produce a signal calibrated to a fixed reference signal. The signal is subsequently compared to an automatically selected preset rod block alarm/trip level which is dependent upon the current reactor power level. The trip level is selected based on the magnitude of a reference APRM. If the local neutron flux level increases to the upscale trip setpoint, further control rod motion is blocked, thus limiting the increase in local power. Thus, the ARTS modification to the RBM trip logic replaces the standard RBM flow biased trip feature with power biased trips.

(3) The changes to the system require a reevaluation of the rod withdrawal event and new analyses to provide setpoints for the system. The previous deterministic analysis is replaced with a statistical analysis using a large number of calculations of various operating states and giving

results valid for all CNS cores using GE fuel through type BP8x8R. Currently approved methods were used in the calculations. The calculations give, as a function of RBM trip setting, values of an initial MCPR which assure that 95 percent of withdrawal errors do not violate the MCPR safety limit (1.07) with a 95 percent confidence level. This was used to select setpoints for the RBM, chosen with respect to the other ARTS APRM limits so that the withdrawal event is not limiting. For a rated MCPR limit the setpoints (as used for analyses) are divided into three intervals with a constant setpoint in each interval. For CNS at a rated MCPR limit of 1.20, the setpoints for power intervals of 30-65, 65-85, and 85-100 percent power are 117, 111.2, and 107.4 percent of the reference signal respectively. Values for other MCPR limits (1.25, 1.30, 1.35) are also provided in Reference 2. The setpoints are designated as Low Trip Setpoint (LTSP), Intermediate Trip Setpoint (ITSP) and High Trip Setpoint (HTSP) and are applicable to the three designated power intervals (that is, the LTSP is applied at 30% power up to 65% power, the ITSP is applied at 65% power up to 85%, and the HTSP is applied at and above 85% power. These values are in the proposed TS for CNS and are acceptable based on the use of approved methodology.

(4) Calculations were done to examine the sensitivity of the results for core periphery rods (with fewer LPRM strings) and for LPRM failures (up to a 30 percent failure rate). The results indicate that the setpoints are suitable.

(5) An analysis of the effects of filters and time delays in the system was made. Results were presented for system setup with and without a RBM filter. Nebraska Public Power District (NPPD), by Reference 2, has proposed to use RBM setpoints for CNS based on the use of system filters. Use of filters will require a reduction of setpoints and values for adjustments required, if used, are given in Reference 3. A value for time delay setting in the system was not used since supporting analyses for CNS have not been made. Values for power referenced RBM upscale trip level settings based on lowest rated MCPR limits and adjusted for the use of filters are in the proposed TS for CNS.

(6) The data base described above has been used to determine operating limit MCPRs such that no rod withdrawal error could lead to exceeding safety limits. Two MCPR limits are defined, for below 90 percent power and for equal to or above 90 percent power, which are 1.70 and 1.40 respectively. When the operating MCPR is below these values the plant is on a limiting control rod pattern and the RBM system must be operable. When above these values bypass is allowed.

We have reviewed these changes and analyses for the RBM and have concluded, as was the case in previous ARTS reviews, that the instrumentation changes, analyses, methods used, criteria and setpoints are acceptable.

2.5 Technical Specification Changes for ELLL and ARTS

The proposed changes to the CNS Technical Specifications are identified in the licensee's submittal. The changes include (1) deletion of the current setdown requirement, (2) new power and flow dependent MCPR and MAPLHGR limits, and (3) new RBM limits and operability requirements. There would normally be a number of TS changes to parameters required to implement these changes. However, most of the significant changes are proposed to be made to the existing CNS COLR rather than directly to the TS. The RBM limits, which are not considered cycle-specific by the licensee, are proposed for the CNS TS. The changes to the MCPR and MAPLHGR limits, i.e., the new multipliers and limits which are, or may be, cycle specific, are proposed to be in the COLR, which is designed for such parameters and has been approved for CNS (Reference 6). The standard MCPR and MAPLHGR values already appear in the COLR. Directions for applying the multiplying factors and new limit curves are also provided in the COLR along with the corresponding limits. NEDC-31892, which describes the changes and methodology, becomes a reference in the COLR and the TS bases.

There are also the following direct changes to the TS supporting the ELLL-ARTS changes.

(1) TS 2.1.A.1.a "APRM Flux Scram Trip Setting (Run Mode)". The flow-biased APRM flux scram trip setting is changed to a new setting to allow operation in the ELLL power-flow operating region. This is changed for both two loop and single loop operation. This change is necessary for ELLL operation and is similar to previously approved ELLL setpoints for other plants. Administrative changes for clarification and consistency are also proposed. The last two paragraphs in the Bases for TS 2.1.A.1.a concerning the APRM setdown requirements will be deleted since they are no longer applicable when the setdown requirement is removed. These changes are acceptable.

(2) TS 2.1.A.1.d "APRM Rod Block Trip Setting". The APRM rod block trip setting is changed to a new setting for operation in ELLL. This is changed for both two loop and single loop operation. A Bases section discussing the setdown requirement for the rod block setting while operating when the MFLPD is greater than the fraction of rated power (FRP) will be deleted since it is no longer applicable. These changes are acceptable.

(3) TS 4.1.B and TS 4.1.C would be deleted. These specifications provide surveillance requirements for the APRM scram and rod block trip settings when operating with a MFLPD greater than the FRP. The proposed deletion is acceptable since the settings would no longer be applicable.

The first and third paragraphs of Bases Section 4.1.B discussing the surveillance requirements will also be deleted.

(4) Table 3.1.1 "Reactor Protection System Instrumentation Requirements". The flow-biased APRM flux scram trip setting discussed in Item (1) above

will be changed in Table 3.1.1, and Note 20 will be deleted. The 120% APRM Fixed High Flux trip level setting will be listed separately from the flow biased trip setting. These changes are consistent with the LCO changes and are acceptable.

(5) Table 3.2.C, "Control Rod Withdrawal Block Instrumentation", has the APRM and RBM rod block trip setpoints changed similarly to that for the APRM scrams in TS 2.1.A.a discussed above. The rod block changes are acceptable, as they were for the scram changes. A current reference to Note 14 is deleted since it was only applicable when a setdown requirement was imposed. The elimination of the setdown requirement will make the current Note 14 no longer applicable. In its place, a new Note 14 is written to discuss the use of a set of power referenced RBM upscale trip settings designated as LTSP, ITSP, and HTSP discussed earlier in this staff evaluation. The setpoints themselves are identified in TS Table 3.2.C as a function of lowest rated MCPR limit. The RBM Downscale trip setting would also be changed from the current value of greater than or equal to 2.5% of rated power to a new value of greater than or equal to 91/125. This change is made to provide a value which is relative to a RBM initialization reference of 100/125 of full scale. This is consistent with other trip level setting usage in Table 3.2.C. This results in a control rod withdrawal block when abnormally low RBM signals are detected. The setpoints were derived using acceptable ARTS methodology (Ref. 3).

A new Note 15 is added to define RBM operability requirements which states: "The RBM is only required when core thermal power is greater than or equal to 30% of rated power and a limiting control rod pattern (defined in Specification 3.3.B.5) exists. Requirements for operating with a limiting control rod pattern are specified in Specification 3.3.B.5.b."

For clarification, a Note 16 is added to Table 3.2.C which states "RBM trip level settings are relative to the initialization reference signal of 100/125 of full scale which takes place upon control rod selection." This change is administrative and is acceptable.

Table 4.2.C, "Surveillance Requirements for the Rod Withdrawal Block Instrumentation", will be updated to identify a change in the name for the RBM Upscale function from "RBM Upscale (Flow Biased)" to "RBM Upscale (Power Biased)." This is a change resulting from the ARTS system hardware modifications and is acceptable.

Additional changes to notes for Table 3.2.C are proposed as follows:

(a) In Note 1, the second sentence is revised to read "The SRM and IRM rod blocks need not be operable in the RUN mode, and the APRM (flow biased) rod blocks need not be operable in the STARTUP mode." These changes are acceptable, based on the new definition of RBM operability in proposed Note 15.

(b) In Note 2, the third sentence concerning the definition for the equation variable N is deleted. This is acceptable since N is no longer used in the RBM upscale trip setting calculation.

(c) In Note 3, the definition of the minimum number of RBM instrument channels is deleted and will be moved to the new Note 15. This is acceptable.

(6) TS 3/4.3.B.5, "Operation with a Limiting Control Rod Pattern (for Rod Withdrawal Error, RWE)" would be changed to reflect the changes to the RBM, rod withdrawal error analysis, and limiting rod pattern definition. The surveillance requirement would be changed to indicate that surveillance would only be required when only one of the two RBM channels is operable. These are consistent with the ARTS program and are acceptable.

(7) TS 3.11.A, "Average Planar Limiting Heat Generation Rate (APLHGR)", would be revised to specify that the APLHGR for each fuel type as a function of average planar exposure shall not exceed the limiting value specified in the COLR for two recirculation loop operation. TS 3.11.A would also specify that "If at any time during steady state operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded action will be initiated within 15 minutes to restore operation within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, reduce reactor power to less than or equal to 25% of rated power within the next four (4) hours."

For single loop operation, a sentence is added to state: "For one recirculation loop operation the MCPR limits are 0.01 higher than the comparable two-loop values."

The Bases Section 3.11.A would be revised to discuss the bases for the flow and power dependent correction factors. These changes are based on approved methodology and are acceptable.

(8) TS 3.11.C, "Minimum Critical Power Ratio (MCPR)", would be revised to eliminate the requirement to place the reactor in a cold shutdown condition when the steady state MCPR is not returned to lower than the limiting value specified in the COLR. Instead, reactor power will be reduced so that MCPR limits are no longer required. This change is a consequence of the application of the ARTS approach and is acceptable. The LCO for TS 3.11.C would include a specification similar to TS 3.11.A above for exceeding the limiting value of MCPR and for single loop operation.

(9) Surveillance Requirement 4.11.C will be revised to state: "MCPR shall be determined daily during reactor power operation at greater than 25% rated thermal power and following any change in power level or distribution that could cause operation on the operating limit MCPR." This is a result of the change from a limiting control rod pattern and is acceptable.

(10) The Bases for the proposed revised TSs have been changed to reflect the relevant ARTS changes and TS changes described above. These Bases changes reasonably reflect the TS changes and are acceptable. Many of the explicit limits are presented in the COLR as well as in NEDC-31892P (now referenced in the CNS TS and COLR). These provide an acceptable complete description of the ARTS program and limits for CNS.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Nebraska State official was notified of the proposed issuance of the amendment. The State official had no comment.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The NRC 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 previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (56 FR 43810). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 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 the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) 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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Date: November 29, 1991