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Docket No. 50-298

Nebraska Public Power District
ATTN: Mr. J. M. Pilant, Director
Licensing & Quality Assurance
P. O. Box 499
Columbus, Nebraska 68601

Gentlemen:

The Commission has issued the enclosed Amendment No. 46 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 request of February 6, 1978 as supplemented March 3, April 11, 14, 26 and 27, 1978.

The amendment modifies the Technical Specifications to: (1) permit operation of the facility during Cycle 4 with 100 improved two water rod 8x8R reload fuel bundles, designed and fabricated by the General Electric Company (GE) and having average enrichments of 2.74 and 2.83 wt/% U-235, and (2) revise limits based on transient and accident analysis for the Cycle 4 core loading.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

B. Grimes for

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 46 to DPR-46
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

*SEE PREVIOUS YELLOW FOR CONCURRENCES

OFFICE	ORB #3	ORB #3	OELD	ORB #3		
SURNAME	*SSheppard	*VRooney:mjf	*	GLear		
DATE	4/28/78	4/28/78	5/1/78	5/2/78		

Docket No. 50-298

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Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

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1. Amendment No. to DPR-46
2. Safety Evaluation
3. Notice

#1 withdrawn per Zeby 5/2/78
BGrimes

cc w/enclosures:
see next page

concurrence is conditional on
1) not issuing amendment until Thurs. as discussed 5/1 p.m.
2) making generic safety evaluation available in some form as discussed in 5/1 meeting am
3.) making corrections discussed (typo in notice + clarifying generic comment about Cooper)

OFFICE	ORB#3	ORB#3	OELD VKH	ORB#3		
SURNAME	SSheppard	VRooney:acr	V Harding	GLear		
DATE	4/28/78	4/28/78	5/1/78	4/ /78		

cc w/enclosures:
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Columbus, Nebraska 68601

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Brownville, Nebraska 68321

Auburn Public Library
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Auburn, Nebraska 68305

Director
Nebraska Dept. of Environmental Control
P. O. Box 94877, State House Station
Lincoln, Nebraska 68509

Mr. William Siebert, Commissioner
Nemaha County Board of Commissioners
Nemaha County Courthouse
Auburn, Nebraska 68305

Chief, Energy Systems Analyses
Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
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Washington, D. C. 20460

U. S. Environmental Protection Agency
Region VII
ATTN: EIS COORDINATOR
1735 Baltimore Avenue
Kansas City, Missouri 64108

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. 46
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 February 6, 1978, as supplemented March 3, April 11, 14, 26 and 27, 1978, 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 License No. DPR-46 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 46, 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Brian Grimes

Brian Grimes, Assistant Director
for Engineering and Projects
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 2, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 46

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following pages of the Technical Specifications contained in Appendix A of the above indicated license with the attached pages. Changed areas on the revised pages are reflected by a marginal line.

Remove	Insert
1	1
6	6
7	7
8	8
12	12
13	13
14	14
15	15
16	16
16a	
17	17
19	19
20	20
21	21
22	22
24	24
25	25
26	26
27	27
28	28
31	31
42	42
43	43
61	61
62	62
85	85
211	211
-	211b
212	212
214b	214b
214c	214c
214d	214d
214e	214e
217	217

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. Thermal Parameters

1. Critical Power Ratio (CPR) - The critical power ratio is the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest as calculated by application of the GEXL correlation. (Reference NEDO-10958)

2. Maximum Fraction of Limiting Power Density - The Maximum Fraction of Limiting Power Density (MFLPD) is the highest value existing in the core of the Fraction of Limiting Power Density (FLPD).

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

4. Fraction of Limiting Power Density - The ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR for that bundle type. Design LHGR's are 18.5 KW/ft for 7x7 bundles and 13.4 KW/ft for 8x8 bundles.

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

B. Alteration of the Reactor Core - The act of moving any component in the region above the core support plate, below the upper grid and within the shroud. Normal control rod movement with the control rod drive hydraulic system is not defined as a core alteration. Normal movement of in-core instrumentation is not defined as a core alteration.

C. Cold Condition - Reactor coolant temperature equal to or less than 212°F.

D. Design Power - Design power means a steady-state power level of 2486 thermal megawatts. This is 105% of Rated Power and the power to which the safety analysis applies.

E. Engineered Safeguard - An engineered safeguard is a safety system the actions of which are essential to a safety action required to maintain the consequences of postulated accidents within acceptable 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.

Specifications

A. Reactor Pressure ≥800 psia and Core Flow ≥10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.07 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:

$$S \leq 0.66\ W + 54\%$$

where:

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

W = 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)

SAFETY LIMITS	LIMITING SAFETY SYSTEM SETTINGS

1.1.D (Cont'd)

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.

2.1.A (Cont'd)

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:

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

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. <u>APRM Flux Scram Trip Setting (Refuel or Start and Hot Standby Mode)</u>

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

c. <u>IRM</u>

The IRM flux scram setting shall be $\leq$120/125 of scale.

Amendment No. ~~16~~, ~~32~~, ~~39~~, 46

SAFETY LIMITS	LIMITING SAFETY SYSTEM SETTINGS

2.1.A (Cont'd)

d. APRM Rod Block Trip Setting

The APRM rod block trip setting shall be:

$S_{RB} \leq 0.66\ W + 42\%$

where:

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

W = 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)

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

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

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.

Amendment No. ~~16~~, ~~32~~, ~~39~~, ~~41~~, 46

1.1 Bases: (Cont'd)

Rod Array	16, 64 Rods in an 8 x 8 array
	49 Rods in a 7 x 7 array

The required input to the statistical model are the uncertainties listed on Table A-1, Reference 3, the nominal values of the core parameters listed in Table A-2, Reference 3, and the relative assembly power distribution shown in Figure A-1, Reference 3. The R factor distributions that are input to the statistical model which is used to establish the safety limit MCPR are given in Table 5-2B of Reference 4. The basis for the uncertainties in the core parameters is given in NEDO-20340[2] and the basis for the uncertainty in the GEXL correlation is given in NEDO-10958[1]. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution in Cooper Nuclear Station during any fuel cycle would not be as severe as the distribution used in the analysis.

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

The use of the GEXL correlation is not valid for the critical power calculations at pressures below 800 psia or core flows less than 10% of rated. Therefore, the fuel cladding integrity safety limit is protected by limiting the core thermal power.

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 28×10^3 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 28×10^3 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

1.1 Bases: (Cont'd)

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

1. General Electric Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, General Electric Co. BWR Systems Department, November 1973 (NEDO-10958).

2. Process Computer Performance Evaluation Accuracy, General Electric Company BWR Systems Department, June 1974 (NEDO-20340).

3. Supplemental Reload Licensing Submittal for Cooper Nuclear Station Reload 3, January 1978 (NEDO-24093).

4. "Licensing Topical Report GE-BWR Generic Reload Fuel Application," NEDE-24011-P, May 1977, Supplement 2, NEDE-24011-P-2, Feb. 1978.

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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 the thermal power condition of 2381 MWt. The analyses were based upon plant operation in accordance with the operating map given in Figure III-7-1 of the FSAR. 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.

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

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greater significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 25% insertion. By the time the rods are 60% inserted, approximately four dollars of negative reactivity have been inserted which strongly turns the transient, and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

For analyses of the Thermal consequences of the transients, a MCPR of 1.31 for 7x7 fuel and 8x8 fuel and 1.23 for 8x8R fuel is conservatively assumed to exist prior to initiation of the transients.

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

Steady-state operation without forced recirculation will not be permitted, except during startup testing. The analysis to support operation at various

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.

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 > 1.07 when the transient is initiated from MCPR > 1.31 for 7x7 bundles, 1.31 for 8x8 bundles, and 1.23 for 8x8R buncles.

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 850 psig.

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 a 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 1.07. 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 1.07. 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.

2.1 Bases: (Cont'd)

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 FSAR Subsection 14.5 show that scram at this level adequately protects the fuel and the pressure barrier, because MCPR remains well above 1.07 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 $\leq$ 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above 1.07 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 Final 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 at 850 psig was provided to protect against rapid reactor depressurization.

B. Reactor Water Level Trip Settings Which Initiate Core Standby Cooling Systems (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 Final Safety Analyses Report demonstrate that these conditions result in adequate safety margins for the fuel.

C. References

1. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10801, Feb., 1973.

2. Station Safety Analysis Report (Section XIV).

3. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Reload 3", January 1978 (NEDO-24093).

1.2. BASES

The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The safety limits for reactor coolant system pressure are derived directly from unacceptable safety results 1-3, 2-3, and 3-3 of the Station Nuclear Safety Operational Analysis (Appendix G). This unacceptable results require that applicable code limits for the nuclear system not be exceeded. Thus, the safety limits are direct measures of the unacceptable safety results.

The safety limits for the reactor coolant system pressure have been selected so that they are below pressures at which it can be shown that the integrity of the system is not endangered. However, the pressure safety limits are set high enough that no foreseeable circumstances can cause the system pressure to rise over these limits. The pressure safety limits are arbitrarily selected to be the lowest transient overpressures allowed by the applicable codes. ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

The reactor vessel steam dome pressure of 1337 psig is equivalent to a pressure of 1375 psig at the vessel bottom. The design pressure (1250 psig) of the reactor vessel is established so that, when the 10 percent allowance (125 psi) allowed by the ASME Boiler and Pressure Vessel Code, Section III, for pressure transients, is added to the design pressure, a transient pressure limit of 1375 psig at the vessel bottom is established. Correspondingly, the suction and discharge design pressures (1148 and 1274 psig) of the reactor coolant system piping are set so that, when the 20 percent allowance (230 and 254 psi) allowed by the USAS Piping Code, Section B31.1 for pressure transients, are added to the design pressures, transient pressure limits of 1378 and 1528 psig are established. Thus, the pressure safety limit for power operation is established at 1375 psig, the lowest transient overpressure allowed by the pertinent codes, ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

Reference 6 states that the load rejection from high power without bypass is the most severe abnormal operational transient resulting directly in a reactor coolant system pressure increase. The reactor vessel pressure code limit of 1375 psig, given in Subsection IV2 of the Safety Analysis Report, is well above the peak pressure produced by the overpressure transient described above. Thus, the pressure safety limit applicable to power operation is well above the peak pressure that can result from reasonably expected overpressure transients.

Higher design pressures have been established for piping within the reactor coolant system than for the reactor vessel. These pressures create a consistent design with assurance that, if the pressure within the reactor vessel does not exceed 1375 psig, the pressures within the piping cannot exceed their respective transient pressure limits because of static and pump heads.

A safety limit is applied to the Residual Heat Removal System (RHRS) when it is operating in the shutdown cooling mode. When operating in the shutdown cooling mode, the RHRS is included in the reactor coolant system.

REFERENCES

1. Station Safety Analysis (Section XIV)

2. ASME Boiler and Pressure Vessel Code, Section III

3. USAS Piping Code, Section B31.1

4. Reactor Vessel and Appurtenances Mechanical Design (Subsection IV-2)

5. Station Nuclear Safety Operational Analysis (Appendix G)

6. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Reload 3", January 1978 (NEDO-24093).

2.2 BASES

The 8 relief valves and 3 safety valves are sized and set pressures are established in accordance with the requirements of Section III of the ASME Code. A turbine trip without bypass is assumed. Relief valves are taken to operate normally, and credit is taken for a high pressure scram at 1045 psig. This analysis is discussed in Subsection IV-4 and Question 4.20 of Amendment 11 to the Safety Analysis Report.

The relief valve settings satisfy the Code requirements that the lowest valve set point be at or below the vessel design pressure of 1250 psig. These settings are also sufficiently above the normal operating pressure range to prevent unnecessary cycling caused by minor transients. The results of postulated transients where inherent relief valve actuation is required are given in Section XIV of the Safety Analysis Report.

Reanalysis in Reference 6 for the case of MSIV-Closure with flux scram transient results in the peak pressure of 1288 psig at the vessel bottom. This represents a 95 psi margin below the maximum of 110 percent of design pressure allowed by the Code. This is adequate margin to ensure that the 1375 psig pressure safety limit is not exceeded. A sensitivity study on peak vessel pressure to the failure to open of one of the lowest set-point safety valves was performed for a typical high power density BWR (Reference 7). The study is applicable to the Cooper reactor and shows that the sensitivity of a high power density plant to the failure of a safety valve is approximately 20 psi. A plant specific analysis for the Cooper Reload 3 overpressure transient would show results equal to or less than this value.

The design pressure of the shutdown cooling piping of the Residual Heat Removal System is not exceeded with the reactor vessel steam dome less than 75 psig.

REFERENCES

1. Topical Report, "Summary of Results Obtained from a Typical Startup and Power Test Program for a General Electric Boiling Water Reactor", General Electric Company, Atomic Power Equipment Department (APED-5698)

2. Station Nuclear Safety Operational Analysis (Appendix G)

3. Station Safety Analysis (Section XIV)

4. Control and Instrumentation (Section VII)

5. Summary Technical Report of Reactor Vessel Overpressure Protection (Question 4.20, Amendment 11 to SAR).

6. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Reload 3", January 1978 (NEDO-24093)

7. Letter from I. F. Stewart (GE) to v. Stello (NRC) dated December 23, 1975.

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. The designed system response times from the opening of the sensor contact up to and including the opening of the trip actuator contacts shall not exceed 100 milliseconds.

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. Daily during reactor power operation, the peak heat flux and maximum fraction of limiting power density shall be checked and the SCRAM and APRM Rod Block settings given by equations in Specification 2.1.A.1 and 2.1.B shall be calculated if maximum fraction of limiting power density exceeds the fraction of rated power.

C. During reactor power operation with MFLPD > FRP, MCPR shall be calculated at least daily and following any change in power level or distribution that would cause operation with a limiting control rod pattern as defined in Specification 3.3.B.5 and associated bases.

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 continuing 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: 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	$\leq (0.66W+54\%) \left[\frac{FRP}{MFLPD}\right]$ (14)	2	A or C
High Flux	X(7)	X(9)	X(9)	(16)	$\leq$ 15% Rated Power		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)	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 Volume High Water Level CRD-LS-231 A,B,C, & D	X(2)(7)	X	X(2)	X	$\leq$ 36 gallons	2	A

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 recirculation flow in percent of rated flow.

15. The mode switch shall be placed in refuel whenever core alterations are being made.

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.

LIMITING CONDITIONS FOR 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.

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SURVEILLANCE REQUIREMENTS

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 1.07. Usually with power densities of this magnitude the peak occurs low in the core in a low quality region where the initial heat

3.1 BASES

4.1 BASES (Cont'd)

flux is very high. Therefore, with MFLPD $\leq$ FRP there are no technical specification requirements for calculating MCPR. With MFLPD greater than FRP, a daily calculation of MCPR 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 insures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

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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)	$\leq (0.66W + 42\%) \left[\frac{FRP}{MFLPD}\right]$ (2)	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 + 40%) (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 (0.3 counts/second prior to achieving burnup of 3500 MWD/T on first core)	1(1) (6)
RSCS Rod Group C Bypass	$\geq$ 20% Core Thermal Power	(11)

NOTES FOR TABLE 3.2.C

1. For the startup and run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems 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. If the third column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system with the inoperable channel shall be tripped. If the first column cannot be met for both trip systems, both trip systems shall be tripped.

   The minimum number of operable instrument channels may be reduced by one in one of the trip systems for maintenance and/or testing provided that this condition does not last longer than 24 hours in any thirty day period.

2. W is the recirculation loop flow 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. One instrument channel; i.e., one APRM or IRM or RBM, per trip system may be bypassed except only one of four SRM may be bypassed.

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.

break in the HPCI steam piping including the RHR Condensing Mode Steam. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1 out of 2 logic.

Temperature is monitored at twelve (12) locations with four (4) temperature sensors at each location. Two (2) sensors at each location are powered by "A" direct current control bus and two (2) by "B" direct current control bus. Each pair of sensors, e.g., "A" or "B", at each location are physically separated and the tripping of either "A" or "B" bus sensor will actuate HPCI isolation valves.

The trip settings of $\leq$ 300% of design flow for high flow and $\leq$ 200°F for high temperature are such that core uncovery is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of $\leq$ 300% for high flow and $\leq$ 200°F for temperature are based on the same criteria as the HPCI.

The Reactor Water Cleanup System high flow and temperature instrumentation are arranged similar to that for the HPCI. The trip settings are such that core uncovery is prevented and fission product release is within limits.

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

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

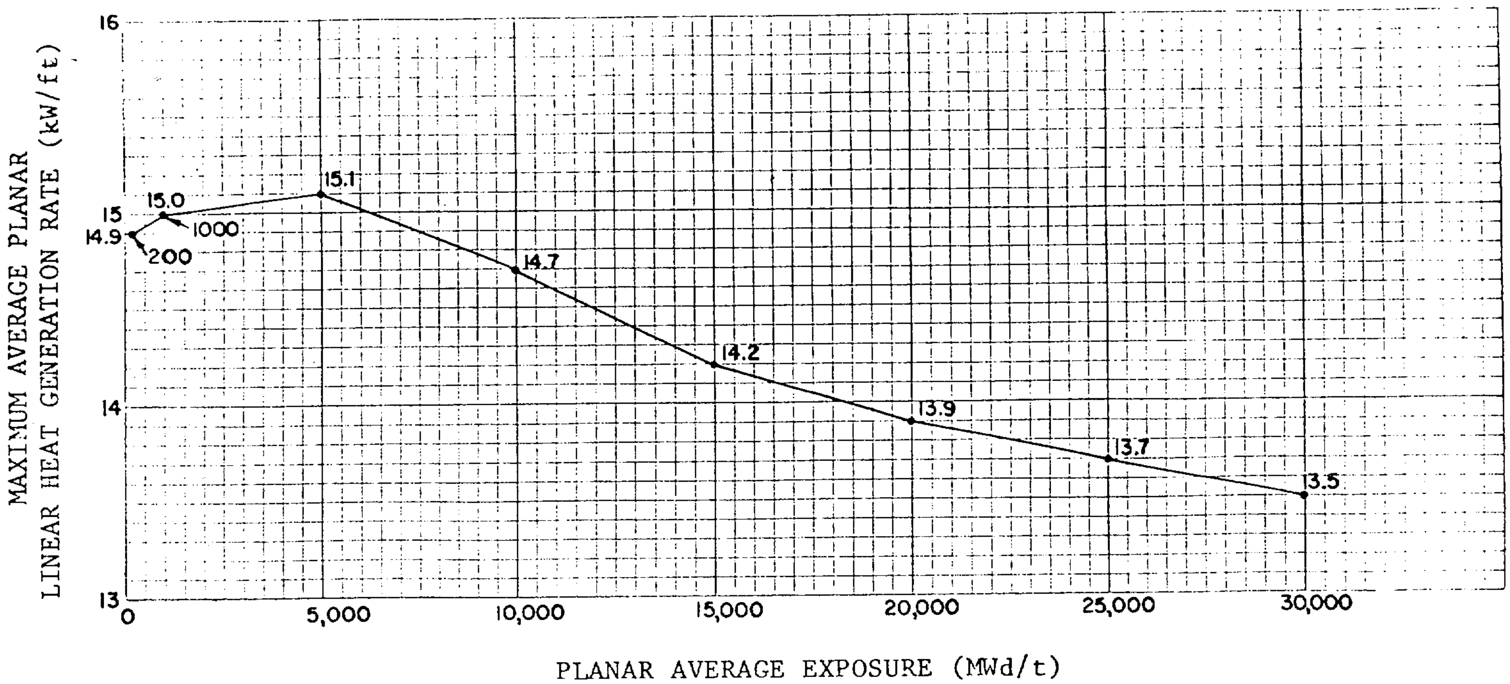

Figure 3.11-1.1 Maximum Average Planar Linear Heat Generation Rate versus Exposure with LPCI Modification and Bypass Flow Holes Plugged, Initial Core Fuel Type 3.

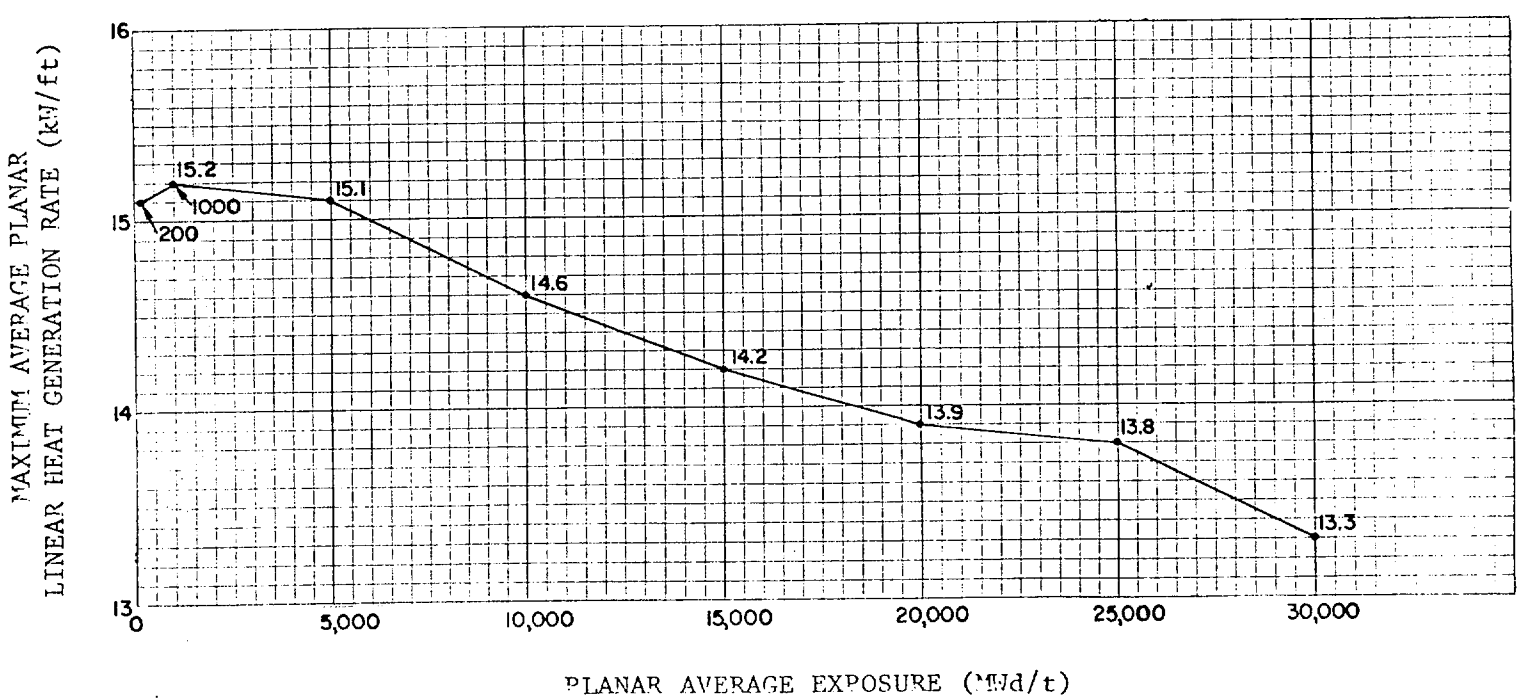

Figure 3.11-1.2 Maximum Average Planar Linear Heat Generation Rate versus Exposure with LPCI Modification and Bypass Holes Plugged, Initial Core Fuel Type 2

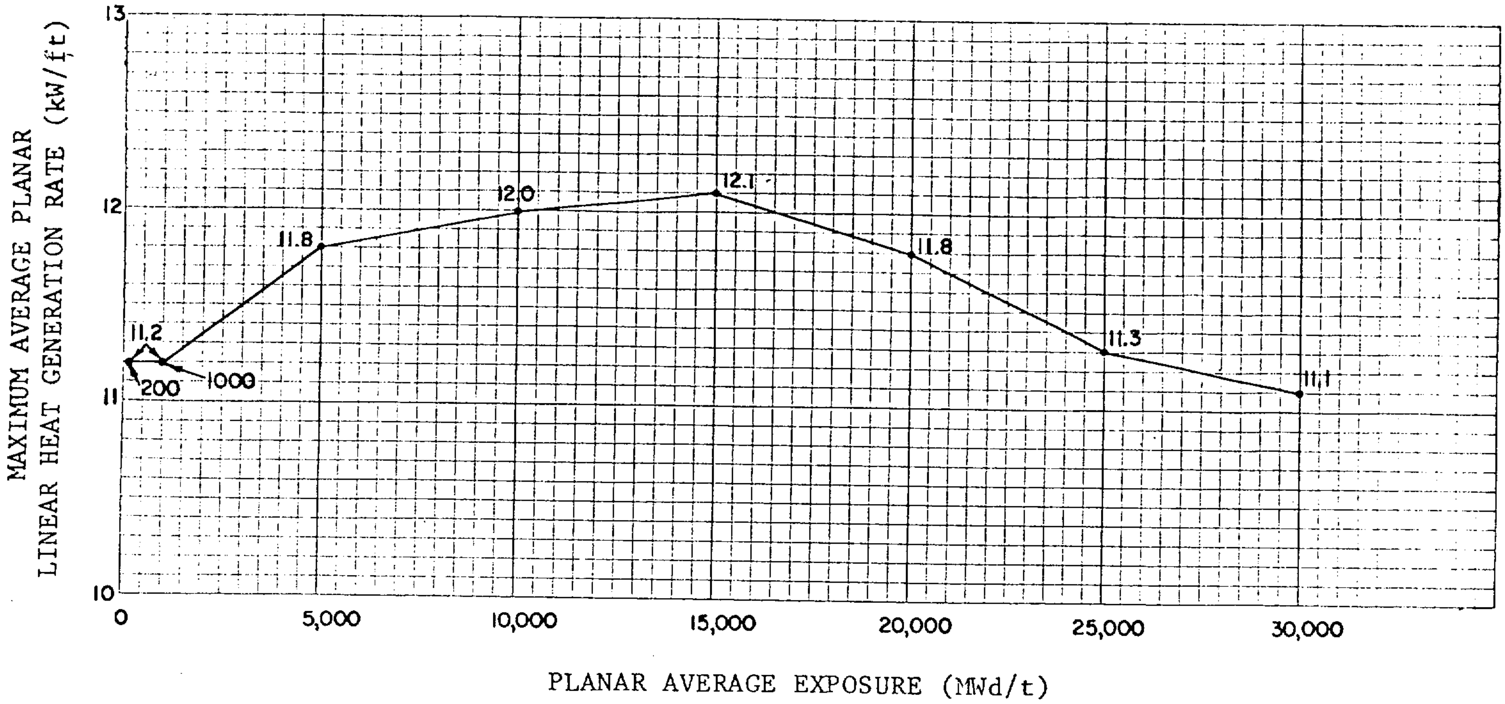

Figure 3.11-1.5 Maximum Average Planar Linear Heat Generation Rate versus Exposure with LPCI Modification and Bypass Flow Holes Plugged, 8DRB283 Fuel.

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
= 0.022 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 than 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 MCPR shall be $\geq$ 1.31 for 7x7 bundles, $\geq$ 1.31 for 8x8 bundles, and $\geq$ 1.23 for 8x8R bundles, at rated power and flow. 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 3.11-2.

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.

REFERENCES

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566 (Draft) Submitted August 1974.

2. General Electric Refill Reflood Calculation (Supplement to SAFE Code Description) transmitted to USAEC by letter, G. L. Gyorey to V. Stello, Jr., dated December 20, 1974.

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 2 and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 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.

C. Minimum Critical Power Ratio (MCPR)
Operating Limit 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 MCPR of 1.07, and an analysis of abnormal operational transients (Reference 5). 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 type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

3.11 Bases (Cont'd)

The limiting transient which determines the required steady state MCPR limit is the loss of feedwater heating for 8x8 and 8x8R bundles and the rod withdrawal error for 7x7 bundles. The transients yield the largest ΔCPRs. When added to the safety limit MCPR of 1.07 the required minimum operating limit MCPR of specification 3.11C are obtained.

Prior to the analysis of abnormal operational transients an initial fuel bundle MCPR was determined. This parameter is based on the bundle flow calculated by a GE multi-channel steady state flow distribution model as described in Section 4 of NEDO-24011[(2)] and on core parameters shown in Table A-2 of Reference 5.

The evaluation of a given transient begins with the system initial parameters shown in Table A-2 of Reference 5 that are input to a GE core dynamic behavior transient computer program described in NEDO-10802[(3)]. Also, the void reactivity coefficients that were input to the transient calculational procedure are based on a new method of calculation termed NEV which provides a better agreement between the calculated and plant instrument power distributions. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic SCAT code described in NEDE-20566[(4)]. The principal result of this evaluation is the reduction in MCPR caused by the transient.

D. MCPR Limits for Core Flows Other than Rated

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

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR of 1.31 for 7x7 and 8x8 and 1.23 for 8x8R 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.

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The K_f factor curves shown in Figure 3.11-2 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.

For the manual flow control mode, the K_f factors were calculated such that at the maximum flow state (as limited by the pump scoop tube set point) and the corresponding core power (along the rated flow control line), the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPR's were calculated at different points along the rated flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR determines the K_f.

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at rated power and flow.

The K_f factors shown in Figure 3.11-2, are conservative for Cooper operation because the operating limit MCPR of 1.31 for 7x7 and 8x8 and 1.23 for 8x8R are greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

References

1. "Cooper Nuclear Station Channel Inspection and Safety Analyses with Bypass Holes Plugged," NEDO-21072, October 1975.

2. Licensing Topical Report, General Electric Boiling Water Reactor, Generic Reload Fuel Application, (NEDE-24011-P) May 1977, Supplement 1 (NEDE-24011-P-1), January 1978.

3. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

4. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566 (Draft), August 1974.

5. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Reload 3," January 1978 (NEDO-24093).

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 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 will be made at 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be 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.

D. Core Stability

The calculations, regarding reactor core stability, presented in "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Reload 3," January 1978 (NEDO-24093), show that the reactor is in compliance with the ultimate performance criteria, including the most responsive condition at natural circulation and rod block power. However, to preclude the possibility of operation under conditions which could result in reactor core instability, the NRC requested the incorporation of a specification limit.

The power level specified results in a decay ratio (X_2/X_0) which is significantly less than the ultimate stability limit of 1.0.

5.0 MAJOR DESIGN FEATURES

5.1 Site Features

The Cooper Nuclear Station site is located in Nemaha County, Nebraska, on the west bank of the Missouri River, at river mile 532.5. This part of the river is referred to by the Corps of Engineers as the Lower Brownville Bend. Site coordinates are approximately 40^{o} 21' north latitude and 95^{o} 38' west longitude. The site consists of 1351 acres of land owned by Nebraska Public Power District. About 205 acres of this property is located in Atchison County, Missouri, opposite the Nebraska portion of the station site. The land area upon which the station is being constructed is crossed by the Missouri River on the east and is bounded by privately owned property on the north, south, and west. At the west site boundary, a county road and Burlington Northern Railroad spur pass the site.

The reactor (center line) is located approximately 3600 feet from the nearest property boundary. No part of the present property shall be sold or leased by the applicant which would reduce the minimum distance from the reactor to the nearest site boundary to less than 3600 feet without prior NRC approval.

The protected area is formed by a seven foot chain link fence which surrounds the site buildings.

5.2 Reactor

A. The core shall consist of not more than 548 fuel assemblies of 7x7 (49 fuel rods) and 8x8 (63 fuel rods) and 8x8R (62 fuel rods).

B. The core shall contain 137 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C) compacted to approximately 70% theoretical density.

5.3 Reactor Vessel

The reactor vessel shall be as described in Section IV-20 of the SAR. The applicable design shall be as described in this section of the SAR.

5.4 Containment

A. The principal design parameters for the primary containment shall be as given in Table V-2-1 of the SAR. The applicable design shall be as described in Section XII-2.3 of the SAR.

B. The secondary containment shall be as described in Section V-3.0 of the SAR.

C. Penetrations to the primary containment and piping passing through such

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 Introduction

Nebraska Public Power District (the licensee) has proposed changes to the Technical Specifications of Facility Operating License DPR-46 for Cooper Nuclear Station (CNS). The proposed changes permit operation of the CNS after the replacement of 100 fuel assemblies constituting refueling of the core for fourth cycle operation.

In support of the reload application, the licensee has provided the GE BWR Reload 3 licensing submittal for CNS (Reference 1), proposed Technical Specification changes (Reference 2), information on the CNS Loss of Coolant Accident (LOCA) analysis (References 3 and 4), and responses to NRC requests for additional information (Reference 5). This licensing action was noticed in the FEDERAL REGISTER on March 31, 1978 (43 F.R. 13650).

This reload is the first for CNS which involves loading of GE 8 x 8 Retrofit (8 x 8 R) fuel. The description of the nuclear and mechanical design of the 8 x 8 R fuel and the older design 8 x 8 fuel is contained in GE's licensing topical report for BWR reloads (Reference 6). Reference 6 also contains a complete set of references to topical reports which describe GE's analytical methods for nuclear, thermal-hydraulic, transient and accident calculations, and information regarding the applicability of these methods to cores containing 7 x 7, 8 x 8, and 8 x 8 R fuel. Portions of the plant-specific data such as operating conditions and design parameters which are used in transient and accident calculations have also been included in Reference 6.

The staff's safety evaluation (Reference 8) of the GE generic reload licensing topical report has concluded that the nuclear and mechanical design of the 8 x 8 R fuel, and GE's analytical methods for nuclear, thermal-hydraulic, and transient and accident calculations as applied to mixed cores containing 7 x 7, 8 x 8, and 8 x 8 R fuel are acceptable. Approval of the nuclear and mechanical design of 8 x 8 fuel was originally based on information in Reference 7 and expressed in the staff's evaluation (Reference 9) of that document.

Based on the staff's review, the plant-specific input data for transient and accident analyses presented in Reference 6 are acceptable (Reference 8). Additional plant and cycle-dependent data and information are provided in Reference 1, which closely follows the outline of Appendix A of Reference 6.

Because of the staff's review of a large number of generic considerations related to use of 8 x 8 R fuel in mixed loadings with 8 x 8 and 7 x 7 fuel, and on the basis of the evaluations which have been presented in Reference 8, only a limited number of additional areas of review have been included in this safety evaluation report. These include the plant and cycle-specific input data and results presented in Reference 1, the physics startup test program described in Reference 5, the application of a new GE method for analyzing fuel loading errors (References 10 and 11), and those items identified in Reference 8 as requiring special attention during reload reviews.

For evaluations of areas not specifically addressed in this safety evaluation report, the reader is referred to Reference 8.

2.0 Evaluation

2.1 Nuclear Characteristics

For Cycle 4 operation of CNS, twenty-four fresh 8 x 8 fuel bundles of type 8D274L and seventy-six fresh 8 x 8 R bundles of type 8DRB283 will be loaded into the core (Reference 1). The remainder of the 548 fuel bundles in the core will be 7 x 7 and 8 x 8 fuel exposed during the first three cycles.

The fresh fuel will be loaded in an eighth-core symmetric pattern (Figure 1 of Reference 1) which is acceptable.

Based on the data presented in sections 4 and 5 of Reference 1, both the control rod system and the standby liquid control system will have acceptable shutdown capability during Cycle 4.

The Cooper station presently has the GE type A spent storage racks and these racks will meet the fuel storage subcriticality requirement for the uranium 235 enrichments in this reload.

2.2 Thermal Hydraulics

2.2.1 Fuel Cladding Integrity Safety Limit MCPR

As stated in Reference 6, the minimum critical power ratio (CPR) which may be allowed to result from core-wide or localized transients or from undetected fuel loading errors in 1.07. This limit has been imposed to assure that during transients 99.9% of the fuel rods will avoid transition boiling, and that transition boiling will not occur during steady state operation as the result of the worst possible fuel loading error.

The safety limit MCPR for CNS is being raised from 1.06 to 1.07 because the distribution of fuel rod power within the 8x8R fuel bundles is different from that of the 8 x 8 fuel. The reason for the difference is the presence of two rather than one water rods in 8 x 8 R fuel. The issue has been addressed in Reference 8 and the 1.07 limit has been found acceptable for BWRs with uncertainties in flux monitoring and operational parameters no greater than those listed in Table 5-1 of Reference 6, and for which the CPR distribution is within the bounds of Figures 5.2 and 5.2a of Reference 6. It has been shown in Table A-1 and Figure A-2 of Reference 1 that these conditions are met for CNS Cycle 4.

In addition, the most important of the plant/cycle-specific uncertainties, the TIP uncertainty, will be evaluated during the physics startup tests to confirm that the TIP uncertainty is within acceptable bounds (Reference 5).

2.2.2. Operating Limit MCPR

Various transients or perturbations to the CPR distribution could reduce* the MCPR below the intended operating limit during Cycle 4 operation of CNS. The most limiting operational transients and the fuel loading error have been analyzed by the licensee to determine which could potentially induce the largest reduction in MCPR (References 1 and 5).

*The reduction in CPR has sometimes been called ΔCPR.

The transients evaluated were the generator load rejection without bypass, feedwater controller failure at maximum demand, the turbine trip with failure of the bypass valves, loss of a 100° F. feedwater heating, and the control rod withdrawal error. Initial conditions and transient input parameters as specified in Tables 6 and 7 and Figure 2 of Reference 1 were assumed.

The input data for the transient calculations have been reviewed and will provide adequate conservatism for determination of transient reductions in CPR.

Calculated system responses and reductions in CPR during each of the operational transients have been given in Reference 1. For 7 x 7 fuel, the reduction in CPR for the rod withdrawal error is the largest, having the value 0.17. For the 8 x 8 and 8 x 8 R fuel, the reductions in CPR for the 100° loss of feedwater heating are the largest, having the value 0.14.

A fuel loading error will cause the calculated CPR to be based on an erroneous bundle type or orientation and thus will result in a CPR error.

Fuel loading errors have also been taken into account, and as indicated in Reference 5, it has been found that if an 8 x 8 R bundle is misoriented by 180°, the error in CPR is 0.16; if a fresh 8 x 8 bundle is substituted for an exposed 7 x 7, the CPR error is 0.24; and if a fresh 8 x 8 is substituted for an exposed 8 x 8, the CPR error is no greater than 0.24.

Addition of the most severe CPR errors or reductions to the safety limit (1.07) gives the appropriate operating limit MCPR for each fuel type. This results in MCPRs of 1.23 for 8 x 8 R fuel, and 1.31 for 8 x 8 and 7 x 7 fuel. These operating limit MCPRs will assure that the safety limit MCPR is not violated due to transients or fuel loading errors and are acceptable.

2.3 Accident Analysis

2.3.1 ECCS Appendix K Analysis

Input data and results for the CNS ECCS analysis have been given in References 1, 3, and 4. The information presented fulfills the requirements for such analyses outlined in Reference 8.

We have reviewed the analyses and information submitted for the reload and conclude that the CNS plant will be in conformance with all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 when: (1) it is operated in accordance with the "MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE" values given in Figures 3.11.1.1 through 3.11.1.5 of Reference 2, and (2) when it is operated at a Minimum Critical Power Ratio (MCPR) equal to or greater than 1.20 (more restrictive MCPR limits are currently required for reasons not connected with the Loss-of-Coolant-Accident, as described in Section 2.2).

2.3.2 Control Rod Drop Accident

For the worst case control rod drop accident (CRDA) under hot startup conditions, the characteristic parameters for the accident meet the requirements for bounding analyses described in Reference 6. As stated in Reference 8, this is adequate to show that the design basis of 280 cal/gm peak fuel enthalpy for a hot startup CRDA is met.

Because the characteristic accident parameters for the worst cold startup CRDA do not satisfy the requirements for bounding analyses, it was necessary to perform a plant specific analysis. The resulting calculated peak fuel enthalpy for the postulated CRDA would be 256 cal/gm which is acceptable.

2.3.3 Fuel Loading Error

As discussed in Section 2.3, potential fuel loading errors involving misoriented bundles and bundles loaded into incorrect positions have been evaluated. The new GE method for analysis of misoriented bundles (References 10 and 11) has been used for the 8x8R analysis. This method has been reviewed and approved by the staff (Reference 8). All other fuel loading error induced CPR errors were evaluated on the basis of the older standard GE methods.

As recommended in Reference 8, the CPR error calculated using GE's new method for the 180° misoriented 8x8R bundle has been increased by 0.02 to take into account uncertainties related to axially varying R-factors.

The analyses which have been performed for potential CNS fuel loading errors are acceptable for assuring that CPR's will not be below the safety limit MCPR of 1.07.

2.4 Overpressure Analysis

The CNS overpressure analysis for the MSIV closure with high flux scram, which is the limiting overpressure event, has been performed in accordance with the requirements of Reference 8. As specified in Reference 8, the sensitivity of peak vessel pressure to failure of one safety valve has been evaluated. There is sufficient margin between the peak calculated vessel pressure and the design limit pressure to allow for the failure of the valve..

2.5 Thermal Hydraulic Stability

The results of the CNS thermal hydraulic stability analysis (Reference 1) show that the channel hydrodynamic and reactor core decay ratios at the Natural Circulation - 105% Rod Line intersection (which is the least stable physically attainable point of operation) are below the stability limit.

Because operation in the natural circulation mode at greater than 25% power will be prohibited by Technical Specifications, there will be added margin to the stability limit and this is acceptable.

2.6 Physics Startup Testing

The licensee will perform a series of physics startup tests and procedures to provide assurance that the conditions assumed for the transient and accident analysis calculations will be met during Cycle 4. The tests will check that the core is loaded as intended, that the incore monitoring system is functioning as expected, and that the process computer has been reprogrammed to properly reflect changes associated with the reload.

Methods and criteria for the tests have been described in Reference 5 and are acceptable. A written report of the startup tests will be provided to NRC within approximately 45 days.

2.7 Technical Specification Modifications

The only proposed changes in the CNS Technical Specifications which have not yet been discussed are those related to the prescribed method for calculating APRM flow-biased trip settings (Reference 2). This modification does not involve a quantitative change in the setpoints or a reduction in margins to the trip point. The change prescribes a more direct use of limits monitoring data from the plant process computer than has been the case with the previous procedure.

For these reasons the modification is acceptable.

As a result of the licensee's proposal and our review, modification to the licensee's proposed Technical Specifications were necessary. These modifications were discussed with and agreed to by the licensee.

3.0 Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

4.0 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

References

1. "Supplemental Reload Licensing Submittal" for Cooper Nuclear Station, Reload 3, NEDO-24093, January 1978.

2. Letter, Jay Pilant, Northeast Nuclear Energy Company to George Lear, NRC, March 3, 1978.

3. Letter, Jay Pilant, NNEC to George Lear, NRC, April 11, 1978.

4. "Loss of Coolant Accident Analysis Report," NEDO-24045, August 1977.

5. Letters from Jay Pilant, NNEC to George Lear, NRC, dated April 14, 26 and 27, 1978.

6. General Electric Boiling Water Reactor Generic Reload Fuel Application, NEDE-24011-P, April 1977 - Draft, Transmitted by May 1, 1978 Memo, Vernon L. Rooney to George E. Lear.

7. General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel, NEDO-20360, Rev. 1, Supplement 4, April 1, 1976.

8. Safety Evaluation of the GE Generic Reload Fuel Application (NEDE-24011-P), April 1978.

9. Status Report on the Licensing Topical Report "General Electric Boiling Water Generic Reload Application for 8x8 Fuel," NEDO-20360, Revision 1 and Supplement 1 by Division of Technical Review, Office of Nuclear Reactor Regulation, United States Nuclear Regulatory Commission, April 1975.

10. Letter, Ronald Engel, GE to Darrell Eisenhut, NRC, Fuel Assembly Loading Error, June 1, 1977.

11. Letter, Ronald Engel, GE to Darrell Eisenhut, NRC, November 30, 1977.

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-298

NEBRASKA PUBLIC POWER DISTRICT

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 46 to Facility Operating License No. DPR-46, issued to the Nebraska Public Power District (the Licensee), which revised the Technical Specifications for operation of the Cooper Nuclear Station (the facility) located in Nemaha County, Nebraska. The amendment is effective as of the date of issuance.

The amendment modifies the Technical Specifications to: (1) permit operation of the facility during Cycle 4 with 100 improved two water rod 8x8R reload fuel bundles, designed and farbicated by the General Electric Company (GE) and having average enrichments of 2.74 and 2.83 wt/% U-235, and (2) revise limits based on transient and accident analysis for the Cycle 4 core loading.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on March 31, 1978 (43 F.R. 13650). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated February 6, 1978, as supplemented March 3, April 11, 14, 26, and 27, 1978, (2) Amendment No. 46 to License No. DPR-46, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Auburn Public Library, 118 - 15th Street, Auburn, Nebraska 68305. A single copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulation Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 2nd day of May 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Brian Grimes

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