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DBrinkman
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OPA (CMiles)
RDiggs
HDenton

APRIL 27 1979

Docket No. 50-298

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

Dear Mr. Pilant:

The Commission has issued the enclosed Amendment No. 55 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 January 31, 1979 as supplemented March 22, 1979 and April 16, 1979.

The amendment modifies the Technical Specifications to: (1) permit operation of the facility during Cycle 5 with 112 exposed 7x7 fuel assemblies loaded with the initial core replaced with an equivalent number of fresh 8x8R fuel assemblies, designed and fabricated by the General Electric Company (GE) and (2) revise limits based on transient and accident analysis for the Cycle 5 core loading.

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

Sincerely,

Original Signed By
T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 55 to DPR-46
2. Safety Evaluation
3. Notice

7905110001

cc w/enclosures:
See next page

concurs in amendment and Fed. Reg. notice only

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SURNAME	SSheppard	VRooney:mjf	BGrimes	STREBY	TIppolito
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NRC FORM 318 (9-76) NRCM 0240 U.S. GOVERNMENT PRINTING OFFICE: 1978 - 265 - 769

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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. 55
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 January 31, 1979, as supplemented March 22 and April 16, 1979 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. 55, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 27, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 55

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Appendix A of the Technical Specifications is revised by removing the pages listed below and replacing with identically numbered pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

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2.1.A (cont'd)

3. <u>Turbine Stop Valve Closure Scram Trip Setting</u>

   $\leq$ 10 percent valve closure when above 30% turbine first stage pressure.

4. <u>Turbine Control Valve Fast Closure Scram Trip Setting</u>

   Turbine control fluid pressure $\geq$1000 psi when above 30% turbine first stage pressure.

5. <u>Main Steam Line Isolation Valve Closure Scram Trip Setting</u>

   $\leq$10 percent valve closure when above 1000 psig reactor pressure, in 3 out of 4 main steam lines.

6. <u>Main Steam Line Isolation Valve Closure on Low Pressure</u>

   $\geq$825 psig when mode switch is in "Run".

Relationship of instrument water level indications to core and reactor vessel levels is illustrated in Figure 2.1-1

B. <u>Reactor Water Level Trip Settings Which Initiate Core Standby Cooling Systems (CSCS)</u>

Reactor low-low water level initiation of CSCS systems setting shall be at or above -145.5 in. indicated level.

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 in Table 5-1, Reference 3, the nominal values of the core parameters listed in Table 5-2, Reference 3, and the relative assembly power distribution shown in Figure 5-1a, 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 3. 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. "Licensing Topical Report GE-BWR Generic Reload Fuel Application," NEDE-24011-P, May 1977, Supplement 2, NEDE-24011-P-2, Feb. 1978.

2.1 Bases:

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

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 1.20 for 7x7 fuel and 1.22 for 8x8 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)

power and flow relationships has considered operation with either one or two recirculation pumps.

In summary:

i. The abnormal operational transients were analyzed to 105% of rated steam flow.

ii. The licensed maximum power level is 2381 MWt.

iii. Analyses of transients employ adequately conservative values of the controlling reactor parameters.

iv. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.

A. Trip Settings

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

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (2381 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120% scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin from fuel damage. Therefore, the use of flow referenced scram trip provides even additional margin.

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 > 1.23 for 7x7 bundles, and 1.23 for 8x8 bundles.

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)

5. Main Steam Line Isolation Valve Closure on Low Pressure

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

B. Reactor Water Level Trip Settings Which Initiate Core Standby Cooling 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 Unit 1, Reload 4", December 1978 (NEDO-24170).

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 Unit 1, Reload 4", December 1978 (NEDO-24170).

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 1276 psig at the vessel bottom. This represents a 99 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 Unit 1, Reload 4", December 1978 (NEDO-24170)

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

COOPER NUCLEAR STATION

TABLE 3.2.A (Page 1)

PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION INSTRUMENTATION

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required When Component Operability is Not Assured (2)
Main Steam Line High Rad.	RMP-RM-251, A,B,C,& D	≤3 Times Full Power	2	A or B
Reactor Low Water Level	NBI-LIS-101 A,B,C,& D	≥+12.5" Indicated Level	2(4)	A or B
Reactor Low Low Water Level	NBI-LIS-57 A & B NBI-LIS-58 A & B	≥-37" Indicated Level	2	A or B
Main Steam Line Leak Detection	MS-TS-121, A,B,C, & D 122, 123, 124, 143, 144, 145, 146, 147, 148, 149, 150	≤200°F	2(6)	B
Main Steam Line High Flow	MS-dPIS-116 A,B,C,& D 117, 118, 119	≤140% of Rated Steam Flow	2(3)	B
Main Steam Line Low Pressure	MS-PS-134 A,B,C,& D	≥825 psig	2(5)	B
High Drywell Pressure	PC-PS-12, A,B,C,& D	≤2 psig	2(4)	A or B
High Reactor Pressure	RR-PS-128 A & B	≤75 psig	1	D
Reactor Water Cleanup System High Temperature	RWCU-TIS-99	≤140°F	1	C
Main Condenser Low Vacuum	MS-PS-103 A,B,C,& D	≥7" Hg (7)	2	A or B
Reactor Water Cleanup System High Flow	RWCU-dPIS-170 A & B	≤200% of System Flow	1	C

Amendment No. 55

NOTES FOR TABLE 3.2.A

1. Whenever Primary Containment integrity is required there shall be two operable or tripped trip systems for each function.

2. If the minimum number of operable instrument channels per trip system requirement cannot be met by a trip system, that trip system shall be tripped. If the requirements cannot be met by both trip systems, the appropriate action listed below shall be taken.

   A. Initiate an orderly shutdown and have the reactor in a cold shutdown condition in 24 hours.

   B. Initiate an orderly load reduction and have the Main Steam Isolation Valves shut within 8 hours.

   C. Isolate the Reactor Water Cleanup System.

   D. Isolate the Shutdown Cooling System.

3. Two required for each steam line.

4. These signals also start the Standby Gas Treatment System and initiate Secondary Containment isolation.

5. Not required in the refuel, shutdown, and startup/hot standby modes (interlocked with the mode switch).

6. Requires one channel from each physical location for each trip system.

7. Low vacuum isolation is bypassed when the turbine stop is not full open, reactor pressure is $\leq$ 1000 psig and manual bypass switches are in bypass.

8. The instruments on this table produce primary containment and system isolations. The following listing groups the system signals and the system isolated.

   Group 1

   Isolation Signals:

   1. Reactor Low Low Water Level (-37 in.)
   2. Main Steam Line High Radiation (3 times full power background)
   3. Main Steam Line Low Pressure ($\geq$ 825 psig in the RUN mode)
   4. Main Steam Line Leak Detection ($\leq$ 200°F)
   5. Condenser Low Vacuum (7" Hg vacuum)
   6. Main Steam Line High Flow (140% of rated flow)

   Isolations:

   1. MSIV's
   2. Main steam line drains

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)	$\leq (0.66W + 42\%) \left[\frac{FRP}{MFLPD}\right]$ (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 + 39%) (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)

Amendment No. 55

3.2 BASES (cont'd.)

and the guidelines of 10 CFR 100 will not be exceeded. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference paragraph VI.5.3.1 FSAR.

The high drywell pressure instrumentation is a diverse signal for malfunctions to the water level instrumentation and in addition to initiating CSCS, it causes isolation of Group 2 and 6 isolation valves. For the breaks discussed above, this instrumentation will generally initiate CSCS operation before the low-low-low water level instrumentation; thus the results given above are applicable here also. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes isolation of all isolation valves except Groups 4 and 5.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case of accident, main steam line break outside the drywell, a trip setting of 140% of rated steam flow in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperatures peak at approximately 1000°F and release of radioactivity to the environs is below 10CFR100 guidelines. Reference Section SIV.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steam tunnel and along the steam line in the turbine building to detect leaks in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves. See Spec. 3.7 for Valve Group. The setting is 200°F for the main steam leak detection system. For large breaks, the high steam flow instrumentation is a backup to the temp. instrumentation.

High radiation monitors in the main steam tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established setting of 6 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Reference Section XIV.6.2 FSAR.

Pressure instrumentation is provided to close the main steam isolation valves in RUN Mode when the main steam line pressure drops below Specification 2.1.A.6. The Reactor Pressure Vessel thermal transient due to an inadvertent opening of the turbine bypass valves when not in the RUN Mode is less severe than the loss of feedwater analyzed in section XIV.5 of the FSAR, therefore, closure of the Main Steam Isolation valves for thermal transient protection when not in RUN mode is not required.

The HPCI high flow and temperature instrumentation are provided to detect a flux. The requirements of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of $10^{-8}\%$ of rated power used in the analyses of transients cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws control rods according to written sequences. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

   A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit (i.e., MCPR = 1.07, and LHGR = as defined in 1.0.A.4). During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Reactor Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other personnel qualified to perform this function may be designated by the station superintendent.

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 3. 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.

On an early BWR, some degradation of control rod scram performance occurred during plant startup and was determined to be caused by particulate material (probably construction debris) plugging an internal control rod drive filter. The design of the present control rod drive (Model CRDB144B) is grossly improved by the relocation of the filter to a location out of the scram drive path; i.e,, it can no longer interfere with scram performance, even if completely blocked.

3.3 and 4.3 BASES: (Cont'd)

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

In the analytical treatment of the transients, 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 deenergizes. Approximately 120 milliseconds later, the control rod motion is estimated to actually begin. However, 200 milliseconds is conservatively assumed for this time interval in the transient analyses and this is also included in the allowable scram insertion times of Specification 3.3.C. The time to deenergize the pilot valve scram solenoid is measured during the calibration tests required by Spec 4.1.

D. Reactivity Anomalies

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

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

E. Recirculation Pumps

Until analyses are submitted for review and approval by the NRC which prove that recirculation pump startup from natural circulation does not cause a reactivity insertion transient in excess of the most severe coolant flow increase currently analyzed, Specification 3.3.E prevents starting recirculation pumps while the reactor is in natural circulation above 1% of rated thermal power.

REFERENCES

1. NEDO-10527, "Rod Drop Accident Analysis for Large Boiling Water Reactors," Paone, Stirn & Woolley, 3-72, Class I.

2. NEDO-10427, Supplement 1, "Rod Drop Accident Analysis for Large Boiling Water Reactors," Stirn, Paone & Yound, 7-72, Class I.

3. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1, Reload 4, December 1978 (NEDO-24170).

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT

3.11 FUEL RODS

Applicability

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

Objective

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

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

During steady state power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figure 3.11-1. If at any time during steady state operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

B. Linear Heat Generation Rate (LHGR)

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

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

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

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

4.11 FUEL RODS

Applicability

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

Objective

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

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

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

B. Linear Heat Generation Rate (LHGR)

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

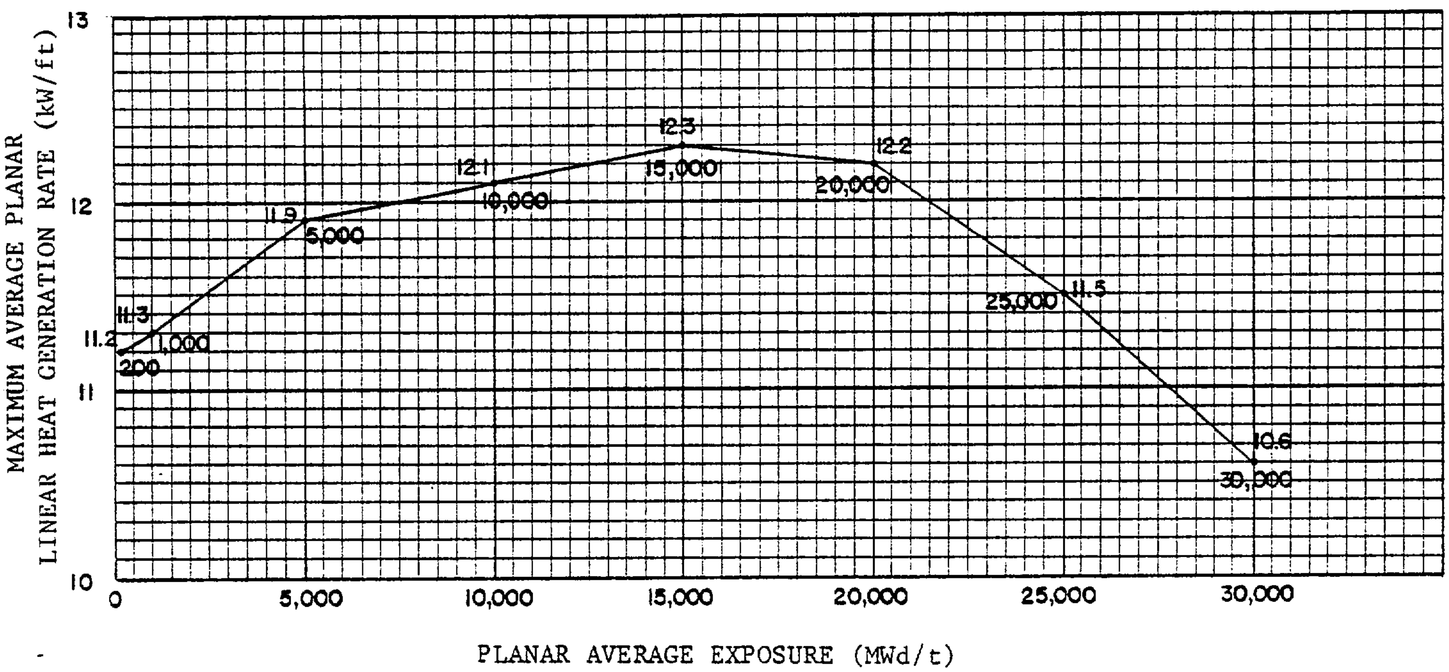

Figure 3.11-1.3. Maximum Average Planar Linear Heat Generation Rate versus Exposure with LPCI Modification and Bypass Flow Holes Plugged, 8D250 Fuel

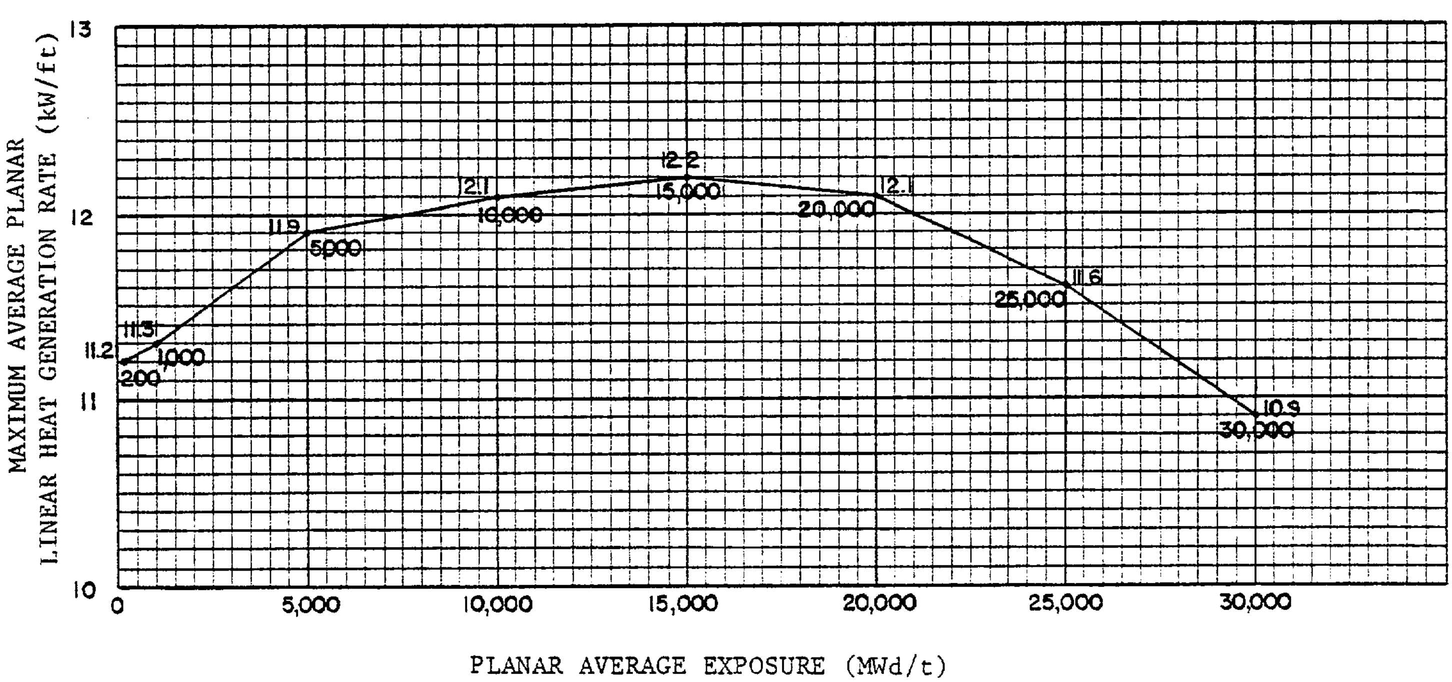

Figure 3.11-1.4. Maximum Average Planar Linear Heat Generation Rate versus Exposure with LPCI Modification and Bypass Flow Holes Plugged, 8D274L Fuel

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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT

LT = Total core length - 12 feet

L = Axial position above bottom of core

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

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

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

C. Minimum Critical Power Ratio (MCPR)

During steady state power operation MCPR shall be $\geq$ 1.23 for 7x7 bundles and $\geq$ 1.23 for 8x8 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.

3.11 Bases: (Cont'd)

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. Pellet densification power spiking in 8x8 fuel has been accounted for in the safety analysis presented in Reference 5; thus no adjustment to the LHGR limit for densification effects is required for 8x8 fuels.

C. Minimum Critical Power Ratio (MCPR)
Operating Limit MCPR

The required operating limit MCPR's at steady state operating conditions a 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 rotated bundle loading error for 8x8 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 5-2 of Reference 2.

The evaluation of a given transient begins with the system initial parameters shown in Table 5-2 of Reference 2 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 will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated for the same postulated transient event.

3.11 Bases: (Cont'd)

The K_f factor curves shown in Figure 3.11-2 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's 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 Unit 1, Reload 4," December 1978 (NEDO-24170).

6. April 18, 1978 letter from J. M. Pilant (NPPD) to G. E. Lear (NRC).

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 Unit 1, Reload 4," December 1978 (NEDO-24170), 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 55 TO FACILITY LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 Introduction

By letter[1] dated January 31, 1979 and supplemented by letters[2][3] March 22, 1979 and April 16, 1979, the Nebraska Public Power District (the licensee) requested amendment to the Technical Specifications appended to Operating License DPR-46 for Cooper Nuclear Station. The proposed changes relate to the fourth refueling of CNS, involving the replacement of 112 exposed 7x7 fuel assemblies loaded with the initial core, with an equivalent number of fresh, two water rod, 8x8R fuel assemblies designed and fabricated by the General Electric Company (GE). In support of this reload application for CNS, the licensee has submitted a supplemental reload licensing document[3] prepared by GE, proposed plant Technical Specification changes[4] and provided responses[2] to our request[5] for additional information on the reload application.

This reload (Reload 4) is the second for CNS to utilize GE's new 8x8R fuel design. Previously for Reload 3, 100 retrofit 8x8R assemblies were loaded into the core. In addition, numerous other BWRs have already refueled once with the new GE fuel design while four lead retrofit test assemblies, previously loaded into another operating reactor, have performed satisfactorily for at least two cycles.

The descriptions of the nuclear and mechanical design of the fresh and exposed 8x8R fuel assemblies and the exposed standard 8x8 fuel assemblies, which were used in connection with prior CNS reloads, are contained in GE's generic licensing topical report[6] for BWR reloads. Reference 6 contains a complete set of references to other GE topical reports which describe GE's BWR reload methodologies for the nuclear, mechanical, thermal-hydraulic, transient and accident analysis calculations. Information addressing the applicability of these methods to reload cores containing a mixture of 7x7, 8x8 and 8x8R fuel is also contained in Reference 6. Portions of the plant-specific data, such as operating conditions and design parameters used in transient and accident calculations, have also been included in the topical report.

Our safety evaluation[7] of GE's generic reload licensing topical report concluded that the nuclear and mechanical design of the 8x8R fuel and GE's analytical methods for nuclear, thermal-hydraulic and transient and accident calculations, as applied to mixed cores containing 7x7, 8x8 and 8x8R fuel, are acceptable. Our acceptance of the nuclear and mechanical design of the standard 8x8 fuel was provided in the staff's evaluation[8] of the information contained in Reference 9.

As part of our evaluation[7] of Reference 6, we found the cycle-independent input data for the reload transient and accident analyses for CNS to be acceptable. The supplementary cycle-dependent information and input data are provided in Reference 3, which follows the format and content of Appendix A of Reference 6. Finally, the licensee has changed the initial core pressure used in the transient analyses from 1045 psia to 1035 psia which appears in Reference 8, to reflect actual plant operating data.

As a result of the staff's generic evaluation[7] of a substantial number of safety considerations relating to the use of 8x8R fuel in mixed core loadings with 7x7 and 8x8 fuel, only a limited number of additional review items are included in this evaluation of Cycle 5 of CNS. These items include the plant and cycle-specific input data and safety analysis results presented in Reference 3, those items identified in our evaluation[7] as requiring special consideration during reload reviews, and the proposed Technical Specification changes.[4]

2.0 Evaluation

2.1 Nuclear Characteristics

For Cycle 5, up to 112 fresh 8x8R fuel bundles with a bundle average enrichment of 2.83 wt/% U-235 will be loaded into the core, replacing a like number of exposed 7x7 assemblies. The remainder of the 548 fuel assembly reconstituted core will consist of irradiated 7x7, 8x8 and 8x8R fuel assemblies exposed during earlier cycles. Thus, about 20 percent of the fuel bundles are being replaced for this reload. The reference core loading for Cycle 5, which is shown in Figure 1 of Reference 3, will result in quarter core symmetry, which is consistent with previous cycles.

The information provided in Section 6 of Reference 3 indicates that the fuel temperature and void dependent characteristics of the refueled core are not significantly different from previous cycles of CNS. Additionally, scram effectiveness, as shown in Figure 2 of Reference 3, is also similar to earlier cycles. The 1.4%Δk/k calculated design shutdown margin for the reconstituted core meets the Technical Specification requirement that the core be subcritical by at least 0.25%Δk/k in the most reactive operating state when the single most reactive control rod is fully withdrawn and all other rods are fully inserted. Finally, Reference 3 indicates that a boron concentration of 600 ppm in the moderator will make the reactor subcritical by 3.6%Δk at 20°C, xenon free. Therefore, the alternate shutdown requirement of the General Design Criteria can be achieved by the Standby Liquid Control System.

2.2 Thermal Hydraulics

2.2.1 Fuel Cladding Integrity Safety Limit MCPR

As stated in Reference 6, for BWR cores which reload with GE's 8x8R fuel, the allowable minimum critical power ratio (MCPR) resulting from either core-wide or localized abnormal operational transients is equal to 1.07. When meeting this MCPR safety limit during a transient, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition.

The 1.07 safety limit minimum critical power ratio (SLMCPR) to be used for Cycle 5 is unchanged from the SLMCPR previously approved for Cycle 4. The basis for this safety limit is addressed in Reference 6, while our generic approval of the new limit is given in Reference 7.

2.2.2 Operating Limit MCPR

Various transient events can reduce the MCPR from its normal operating level. To assure that the fuel cladding integrity safety limit MCPR will not be violated during any abnormal operational transient, the most limiting transients have been reanalyzed for this reload by the licensee, in order to determine which event results in the largest reduction in the minimum critical power ratio. These events have been analyzed for both the exposed 7x7 and 8x8 fuel and the exposed and fresh 8x8R fuel. Addition of the largest reductions in critical power ratio to the safety limit MCPR establishes the operating limits for each fuel type.

2.2.2.1 Abnormal Operational Transient Analysis Methods

The generic methods used for these calculations, including cycle-independent initial conditions and transient input parameters, are described in Reference 6. Our acceptance of the cycle-independent values appears in Reference 7. Additionally, our evaluation of the transient analysis methods, together with a description and summary of the outstanding issues associated with these methods, appears in Reference 7. Supplementary cycle-dependent initial conditions and transient input parameters used in the transient analyses appear in the tables in Sections 6 and 7 of Reference 3. Our evaluation(7) of the methods used to develop these supplementary input values has also been completed.

At the time we completed our evaluation of the generic methods, the acceptability of the GEXL critical power correlation(10), for use in connection with the retrofit fuel design, had not been adequately documented by GE. The staff found, however, that the then available 8x8R critical power test data was sufficient to support the acceptability of GE's 8x8R fuel design for BWR core reloads for one operating cycle. Accordingly, we stated(7) that future BWR core reload applications involving retrofit 8x8 fuel for a second operating cycle would have to include additional information which adequately justified the correlation for application to 8x8R fuel operating beyond one cycle. Since Cycle 5 of CNS involves 8x8R fuel operating for a second cycle and since the initial CNS Reload 5 licensing submittal(3) did not address this issue, we requested(5) that the licensee provide the required additional information. The licensee responded to our request by referencing information(11) furnished to the staff by GE which references a report(12) prepared by GE on this same subject.

Reference 12 provides the results of full scale critical power tests performed on 8x8R fuel bundles. The tests, which included both transient and steady-state simulations, followed the same approved procedures(10) used for the standard 8x8 (single water rod) and 7x7 (all fueled rods) fuel designs. The analysis of a total of 577 steady-state data points was performed using methods also previously approved by the staff. The data, involving nine test assemblies which spanned a range of local power peaking and flow conditions, showed according to GE, that the GEXL correlation was applicable to the 8X8R fuel if adjustment were made to the additive constants used in the formulation of the rod-by-rod R-factors. The local power peaking dependent R-factors, used by the GEXL correlation to evaluate 8x8R bundle critical power, are based on the new additive constants shown in Figure 3-1 of Reference 12, which were also used for the CNS-1, Cycle 5, 8x8R critical bundle power predictions. Using these new additive constants, GE performed a data analysis to assess the accuracy and precision of the GEXL correlation. The results of this analysis showed that the correlation fit provides for a mean predicted-to-measured critical power ratio of 0.9879 with a standard deviation of 0.0234.

When viewed over the range of its applicability (which is the same as the standard 8x8 fuel), the GEXL correlation is therefore somewhat conservatively biased while the statistical variation between the predicted and measured critical power is somewhat less than that associated with the standard 8x8 assembly[10], i.e., 2.34% vs 2.8%. Thus, when viewed over its range of applicability, the 8x8R GEXL correlation (with new additive constants) has somewhat better precision in predicting 8x8R critical bundle powers than the 7x7 and 8x8 GEXL formulations are for predicting 7x7 and 8x8 critical bundle powers respectively. Furthermore, from these results it may also be concluded that the 3.6% standard deviation and best estimate assumption of the GEXL correlation (which were actually used in the GETAB statistical analysis to derive the 1.07 safety limit MCPR) bound the statistical characteristics associated with the subject 8x8R GEXL correlation.

The additional information furnished by GE is also intended to be applicable to all BWR cores which contain 8x8R fuel. Accordingly, this information is also currently being generically reviewed by the staff. Although our evaluation is not yet complete, based on our review to date, we believe that for the range of testing, the 8x8R GEXL correlation has an acceptability and applicability which is equivalent to the 7x7 and 8x8 GEXL correlations previously approved by the staff. From our review of the subject data to date, we have also observed that for those critical power test conditions specifically representative of second cycle fuel operating at normal operating thermal-hydraulic state point, the correlation is somewhat nonconservative in its predictions. This observation focuses in on a correlation behavioral concern not explicitly addressed in the overall GETAB methods approved[13] for the 7x7 and 8x8 fuel types.

Again, this subject is being generically reviewed by the staff. However, until this review is complete, we believe that for Cycle 5 of CNS, there is sufficient conservatism implicit in the generic determination of the 1.07 safety limit MCPR to offset a possible nonconservatism associated with this concern. That is, specifically, the generic GETAB statistical analysis assumed a 3.6% correlation uncertainty while GE's analysis of the 8x8R test data results in a 2.34% standard deviation. Additionally, the generic evaluation considered an all 8x8R equilibrium core, whereas the Cycle 5 CNS core involves 7x7, 8x8 and 8x8R fuel in a non-equilibrium condition. In view of these conservatisms (which are representative of a typical non-equilibrium 8x8R reload core) we believe that the overall thermal-hydraulic (GETAB) methods are adequate for establishing conservative MCPR operating limits for Cycle 5 of CNS. However, as 8x8R equilibrium conditions are approached, this conservatism will diminish. In order that this conservatism not be substantially eroded, we require that this issue be resolved prior to the next reload cycle of CNS.

2.2.2.2 Abnormal Operational Transient Analysis Results

The transient events analyzed for this reload were of the following types: pressurization (turbine trip without bypass, load rejection without bypass and feedwater controller failure), feedwater temperature reduction (loss of 100°F feedwater heating) and local reactivity insertion (control rod withdrawal error).

The licensee reports that the most limiting event in the above categories for both the exposed 8x8 assemblies and the reload and exposed 8x8R assemblies is the load rejection without bypass. This transient results in a CPR reduction of 0.15 for these fuel types. The most limiting transient for the exposed 7x7 assemblies is the control rod withdrawal error, which results in 0.16 change in critical power ratio with an Average Power Range Monitor rod block setpoint of 105%. Addition of these CPRs to the 1.07 SLMCPR establishes fuel-type dependent operating limit MCPRs (i.e., 1.22 for the 8x8/8x8R assemblies and 1.23 for the 7x7 fuel) sufficient to assure that the SLMCPR will not be violated during Cycle 5, were any of the aforementioned events to occur.

The licensee also has considered the effects of possible fuel loading errors (FLE) on bundle CPR. The results of the licensee's FLE analysis (see Section 2.3.3 herein) shows that a somewhat higher MCPR operating limit would be required for the 8x8/8x8R assemblies in order to assure that the MCPR safety limit would not be violated in the event of the most severe FLE. In view of these results, the licensee has proposed that for Cycle 5, the 8x8/8x8R MCPR operating limit be adjusted upward from the aforementioned 1.22 to 1.23. These operating limit MCPRs, i.e., 1.23 for the 8x8/8x8R bundles and 1.23 for the 7x7 bundles, are acceptable to the staff.

2.2.3 Fuel Cladding Integrity Safety Limit LHGR

The control rod withdrawal error and fuel loading error events were reanalyzed by the licensee to determine the maximum transient linear heat generation rates (LHGRs). The results for CNS Cycle 5 show that the fuel type and exposure-dependent safety limit LHGRs, shown in Table 2-3 of Reference 6, will not be violated should these events occur. Thus, fuel failure due to excessive cladding strain will be precluded should either of these events occur. We find these results, which adequately account for the effects of fuel densification power spiking, to be acceptable.

2.3 Accident Analysis

2.3.1 ECCS Appendix K Analysis

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License, implementing the requirements of 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that prior to any license amendment authorizing any core reloading... "the licensee shall submit a re-evaluation of ECCS performance calculated in accordance with an acceptable evaluation model which conforms to the provisions of 10 CFR Part 50.46." The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendments as may be necessary to implement the evaluation assumptions.

For Cycle 4, the licensee re-evaluated(14) the adequacy of CNS ECCS performance in connection with the 8x8R reload fuel design. The methods used in this analysis were previously approved by the staff. For Reload 3, we reviewed the ECCS analysis results submitted by the licensee for the Cycle 4 reload fuel and concluded(15) that CNS would be in conformance with all the requirements of 10 CFR 50.46

and Appendix K to 10 CFR 50 when operated in accordance with the MAPLHGR 8x8R versus Average Planar Exposure values which appeared in the proposed plant Technical Specifications. Since the Reload 4 fuel is of the same design as the Reload 3 fuel, we find this same LOCA-ECCS safety analysis and related Technical Specifications to be equally acceptable for showing compliance with the requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50 for the current Cycle 5 reload fuel.

2.3.2 Control Rod Drop Accident

For Cycle 5, the key plant-specific and cycle-specific nuclear characteristics for the worst case control rod drop accident (CRDA) occurring during hot startup conditions are conservatively bounded by the values used in bounding CRDA analysis given in Reference 6. The bounding analysis, which includes the adverse effects of fuel densification power spiking, shows that the peak fuel enthalpy will not exceed the 280 cal/gm design limit. Therefore, for Cycle 5 of CNS, the peak fuel enthalpy associated with a CRDA from the hot startup condition will also be within the 280 cal/gm design limit.

For the worst case control rod drop accident occurring during cold startup conditions, however, not all of the key plant-specific and cycle-specific nuclear characteristics are within the values used in the generic CRDA analysis. That is, although the actual Cycle 5 Doppler coefficient and scram reactivity shape function conservatively fall within the values assumed in the bounding analysis, the accident reactivity shape function does not. Therefore, the licensee has performed a plant-specific control rod drop accident analysis applicable to CNS for Cycle 5. The results of this analysis, using the approved methods described in Reference 6, show that the positive reactivity insertion rate of the dropped rod is sufficiently compensated by Doppler feedback and scram reactivity effects to limit the peak energy deposition in the fuel to 214 cal/gm.

Thus, we conclude that the results of a control rod drop accident from any in-sequence control rod movement will be below the 280 cal/gm design limit.

2.3.3 Fuel Loading Error

The licensee has considered the effect of postulated fuel loading errors on bundle CPR. An analysis of the most severe fuel loading errors were performed using GE's revised analysis methods[16,17], which have previously been reviewed and approved[18] by the staff. The results show that the worst possible fuel bundle misloadings will not cause a violation of the 1.07 safety limit MCPR assuming the proposed 1.23 OLMCPR for the 8x8 and 8x8R fuel assemblies as well as a 1.23 OLMCPR for the 7x7 fuel assemblies. These results include the application of a 0.02 penalty factor applied to the CPR results of the misoriented fuel bundle analysis, as required by our approval of the revised methods. Thus, these operating limit MCPRs will effectively preclude DNB related fuel failures caused by either fuel cladding overheating or cladding oxidation, which might otherwise occur because of a fuel loading error. These results are acceptable to the staff.

2.4 Overpressure Analysis

For Cycle 5, the licensee has reanalyzed the limiting pressurization event to demonstrate that the ASME Boiler and Pressure Vessel Code requirements are met for CNS. The methods used for this analysis, when modified to account for one failed safety valve, have also been previously approved(7) by the staff. The acceptance criteria for this event is that the calculated peak transient pressure not exceed 110% of design pressure, i.e., 1375 psig. The reanalysis shows that the peak pressure at the bottom of the reactor vessel does not exceed 1296 psig for worst case end-of-cycle conditions, even when assuming the effects of one failed safety valve. This is acceptable to the staff.

2.5 Thermal-Hydraulic Stability

A thermal-hydraulic stability analysis was performed for this reload using the methods described in Reference 6. The results show that the fuel type dependent channel hydrodynamic stability decay ratios and reactor core stability decay ratio at the least stable operating state (corresponding to the intersection of the natural circulation power curve and the 105% rod line) are 0.37 (8x8/8x8R), 0.23 (7x7) and 0.79 respectively. These predicted decay ratios are all well below the 1.0 Ultimate Performance Limit decay ratio proposed by GE.

The staff has expressed generic concerns regarding reactor core thermal-hydraulic stability at the least stable reactor condition. This condition could be reached during an operational transient from high power if the plant were to sustain a trip of both recirculation pumps without a reactor trip. The concerns are motivated by increasing decay ratios as equilibrium fuel cycles are approached and as reload fuel designs change. The staff concerns relate to both the consequences of operating with a decay ratio of 1.0 and the capability of the analytical methods to accurately predict decay ratios. The General Electric Company is addressing these staff concerns through meetings, topical reports and a stability test program. It is expected that the test results and data analysis, as presented in a final test report, will aid considerably in resolving the staff concerns.

Prior to Cycle 4 operation, the staff as an interim measure, added a requirement to the CNS Technical Specifications which restricted planned plant operation in the natural circulation mode. Continuation of this restriction will also provide a significant increase in the reactor core stability operating margins during Cycle 5. On the basis of the foregoing, the staff considers the thermal-hydraulic stability of CNS during Cycle 5 to be acceptable.

3.0 Physics Startup Testing

Several of the key reload safety analysis inputs and results can be assured via preoperational testing. In order to provide this assurance, the licensee will perform a series of physics startup tests, which are described in Reference 20. Based on our review, this program is acceptable. A written report, describing the results of the physics startup tests, will also be provided by the licensee for staff review following completion of the tests.

4.0 Technical Specifications

The proposed Technical Specification changes(4) for Cycle 5 include: revised operating limit minimum critical power ratios (MCPRs) for each fuel type in the core, a new rod block monitor (RBM) setpoint, deletion of the linear heat generation rate (LHGR) densification power spiking penalty and a reduction in the low pressure main steamline isolation setpoint.

The licensee has proposed a single operating limit MCPR of 1.23 for the 7x7, 8x8 and 8x8R fuel assembly types. Based on our evaluation appearing in Section 2.2.2 herein, the staff finds this operating limit MCPR to be consistent with and adequately supported by the Reload 4 safety analyses. The licensee has proposed to decrease the flow biased RBM trip level setting from 106% to 105% at full flow. The change was proposed in order to limit the ΔCPR of rod withdrawal error event, so that it would not be a limiting transient for any fuel type. Since the revised setpoint is consistent with and adequately supported by the safety analysis, as evaluated in Section 2.2.2 herein, we find the proposed change to be acceptable. The licensee has also proposed to delete the 2.2% linear heat generation rate power spiking penalty factor for both the 8x8 and 8x8R fuel types from the CNS Technical Specifications. The original purpose of the subject penalty was to account for the adverse effects of fuel densification power spiking, which were not originally explicitly considered in the safety analyses. Sections 2.2.3 and 2.3.2, herein, found acceptable those transient and accident analysis results which explicitly considered the effects of densification power spiking, as required by Reference 19. Accordingly, we find the proposed deletion of the 2.2% spiking penalty factor to be acceptable. Finally, the licensee has proposed to reduce the low pressure main steamline isolation valve setpoint from 850 psig to 825 psig. From our review, we find this change to be acceptable.

5.0 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 the amendment.

6.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Dated: April 27, 1979

5.0 References

1. Nebraska Public Power District letter (J. Pilant) to USNRC (T. Ippolito), dated January 31, 1979.

2. Nebraska Public Power District letter (J. Pilant) to USNRC (T. Ippolito), dated March 22, 1979.

3. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1 Reload 4," NEDO-24170, December 1978.

4. Proposed Changes to Cooper Nuclear Station Technical Specifications, Attachment 1 to Nebraska Public Power District letter to USNRC, dated January 31, 1979.

5. USNRC letter (T. Ippolito) to Nebraska Public Power District, dated March 13, 1979.

6. "Generic Reload Fuel Application," NEDE-24011-P-A, August 1978.

7. USNRC letter (D. Eisenhut) to General Electric (R. Gridley), dated May 12, 1978.

8. "Status Report on the Licensing Topical Report 'General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel', NEDO-20360 Revision 1 and Supplement 1 by Division of Technical Review, ONRR, USNRC, April 1975.

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

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

11. General Electric letter (R. Engle) to USNRC (D. Eisenhut and R. Tedesco), dated March 30, 1979.

12. General Electric letter (R. Gridley) to USNRC (D. Eisenhut and D. Ross), dated October 5, 1978, transmitting "General Electric Information Report NEDE-24131, Basis for 8x8 Retrofit Fuel Thermal Analysis Application."

13. USNRC letter (W. Butler) to General Electric (I. Stuart), dated October 2, 1974.

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

15. USNRC letter (G. Lear) to Nebraska Public Power District, dated May 2, 1978.

16. General Electric letter (R. Engle) to USNRC (D. Eisenhut), "Fuel Assembly Loading Error," dated June 1, 1977.

17. General Electric letter (R. Engle) to USNRC (D. Eisenhut), dated November 30, 1977..

18. USNRC letter (D. Eisenhut) to General Electric (R. Engle), dated November 30, 1977.

19. USNRC letter (D. Eisenhut) to General Electric (R. Gridley), dated June 9, 1978.

20. Nebraska Public Power District letter (Jay Pilant) to USNRC (T. Ippolito) dated April 16, 1979.

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. 55 to Facility Operating License No. DPR-46, issued to Nebraska Public Power District, which revised the Technical Specifications for operation of the Cooper Nuclear Station, located in Nemaha County, Nebraska. The amendment is effective as of the date of its issuance.

The amendment modifies the Technical Specifications to: (1) permit operation of the facility during Cycle 5 with 112 exposed 7x7 fuel assemblies loaded with the initial core replaced with an equivalent number of fresh 8x8R fuel assemblies, designed and fabricated by the General Electric Company (GE) and (2) revise limits based on transient and accident analysis for the Cycle 5 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. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR Section 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 January 31, 1979 and supplemented March 22 and April 16, 1977, (2) Amendment No. 55 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 Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 27th day of April 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors