delivered 10/25/77

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Docket No. 50-298

OCT 14 1977

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

Gentlemen:

In response to your request dated July 20, 1977, as supplemented by letters dated September 1 and October 3 and 13, 1977, the Commission has issued the enclosed Amendment No. 39 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS).

The amendment authorizes operation of the facility with 64 General Electric 8 x 8 reload fuel assemblies of a type previously approved for use at CNS.

To meet our requirements, certain changes to the Technical Specifications which your proposed were necessary. These changes have been discussed with and agreed to by your staff.

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

Sincerely,

Original signed by

Don K. Davis

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 39 to License No. DPR-46
2. Safety Evaluation
3. Notice

* w/cy ea. of Refs, 13, 14, 15 of encl. 2 to this ltr.

with changes noted - SER changed pgs attached

OFFICE	ORB#2:DOR	ORB#2:DOR	OT/EEB:DOR	OELD	AD:DOR	ORB#2:DOR
SURNAME	MFletcher:nm	RDiggs	BGrimes	McGowan	KGoller	DDavis
DATE	10/14/77	10/14/77	10/ /77	10/14/77	10/ /77	10/14/77

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

Docket No. 50-298

OCT 14 1977

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ATTN: Mr. J. M. Pilant, Director
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Enclosures:
1. Amendment No. 39 to License No. DPR-46
2. Safety Evaluation
3. Notice

cc w/enclosures:

Mr. G. D. Watson, General Counsel
Nebraska Public Power District
P. O. Box 499
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Auburn Public Library (w/cy ea. of Refs 13, 14, 15
118 - 15th Street of encl. 2 to this ltr.)
Auburn, Nebraska 68305

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

Chief, Energy Systems Analyses Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
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401 M St., S. W.
Washington, D. C. 20460

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

cc w/enlosures & NPPD filings dtd:
7/20/77, 9/1/77 & 10/3, 13/77

Director
Department of Environmental Control
Executive Bldg., 2nd Floor
Lincoln, Nebraska 68305

Cooper Nuclear Station
ATTN: Mr. L. Lessor
Station Superitendent
P. O. Box 98
Brownville, Nebraska 68321

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. 39
License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District (the licensee) dated July 20, 1977, as supplemented by letters dated September 1, and October 3 and 13, 1977, 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. 39 , 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

Don K. Davis

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: OCT 14 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 39

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 bearing the same numbers (except as otherwise indicated). Changed areas on the revised pages are reflected by a marginal line.

7
8
14
17
19
20
22
26
27
31
42
43
63
63a (new)
86
212
214
214a
214b
214c
214d

1.1. (Cont'd)

D. Reactor Water Level (Shutdown)

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

2.1.A (Cont'd)

In the event of operation with a maximum total peaking factor (MTPF) greater than the design value of A, the setting shall be modified as follows:

$$S \leq (0.66\ W + 54\%)\ \frac{A}{MTPF}$$

where:

A = 2.64 for 7x7 fuel
= 2.44 for 8x8 fuel

MTPF = The value of the existing maximum total peaking factor

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

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

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

c. IRM

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

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

2.1.A (Cont'd)

d. APRM Rod Block Trip Setting

The APRM rod block trip setting shall be:

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

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 equals 34.2 million lb/hr)

In the event of operation with a maximum total peaking factor (MTPF) greater than the design value of A, the setting shall be modified as follows:

$$S_{RB} \geq (0.66\ W + 42\%)\ \frac{A}{MTPF}$$

where:

A = 2.64 for 7x7 fuel
= 2.44 for 8x8 fuel

MTPF = The value of the existing maximum total peaking factor

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

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

Table 1.1-1

UNCERTAINTIES USED IN THE DETERMINATION

OF THE FUEL CLADDING SAFETY LIMIT

Quantity	Standard Deviation (% of Point)
Feedwater Flow	1.76
Feedwater Temperature	0.76
Reactor Pressure	0.5
Core Inlet Temperature	0.2
Core Total Flow	2.5
Channel Flow Area	3.0
Friction Factor Multiplier	10.0
Channel Friction Factor Multiplier	5.0
TIP Readings	8.7
R Factor	1.6
Critical Power	3.6

Table 1.1-2

NOMINAL VALUES OF PARAMETERS USED IN

THE STATISTICAL ANALYSIS OF FUEL CLADDING INTEGRITY SAFETY LIMIT

Core Thermal Power	3293 MW
Core Flow	102.5 Mlb/hr
Dome Pressure	1010.4 psig
Channel Flow Area	0.1078 ft^2
R-Factor	1.098 (7x7 Bundle)
	1.100 (8x8 Bundle)

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 1.20 for 7x7 fuel and 1.22 for 8x8 fuel is conservatively assumed to exist prior to initiation of the transients. (See Reference 3)

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

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 MTPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1.a, when the maximum total peaking factor is greater than 2.64 for 7x7 fuel and 2.44 for 8x8 fuel.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.06 when the transient is initiated from MCPR > 1.20 for 7x7 fuel and 1.22 for 8x8 fuel.

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 procedures 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 switch 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 1.06. 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 1.06. 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 total peaking factor exceeds 2.64 for 7x7 fuel and 2.44 for 8x8 fuel, thus preserving the APRM rod block safety margin.

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 fo 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. "Cooper Nuclear Station Reload No. 2 Licensing Amendment Submittal", June 1977 (NEDO-24033).

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 87 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 30 psi. A plant specific analysis for the Cooper 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. "Cooper Nuclear Station Reload No. 2 License Amendment Submittal", June 1977 (NEDO-24033).

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 peaking factor 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 the peaking factor exceeds 2.64 for 7x7 fuel and 2.44 for 8x8 fuel.

C. During reactor power operation with TPF $\geq$ 2.64 for 7x7 fuel and 2.44 for 8x8 fuel, 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.

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.
    A = 2.64 for 7x7 fuel
    = 2.44 for 8x8 fuel

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.

3.1 BASES (cont'd)

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

4.1 BASES (cont'd)

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

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

B. The peak heat flux 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 peaking factors are not expected to change significantly and thus a daily check of the peak heat flux 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 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 TPF at 2.64 for 7x7 fuel and 2.44 for 8x8 fuel that MCPR will be as low as 1.06. Usually with peaking factors 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 TPF < 2.64 for 7x7 fuel and 2.44 for 8x8 fuel there are no technical specification requirements for calculating MCPR. With TPF greater than 2.64 for 7x7 fuel and 2.44 for 8x8 fuel a daily check of MCPR per Section 3.11 is sufficient since power distribution shifts are very slow when there have not been significant power or control changes. The requirement for calculating MCPR when a limiting control 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.D

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

System	Instrument I. D. No.	Setting Limit	Number of Sensor Channels Provided by Design	Action (1)
Steam Jet Air Ejector Off-Gas System	RMP-RM-150 A & B	$\leq$ 1 ci/sec	2	A
Reactor Building Isolation and Standby Gas Treatment Initiation	RMP-RM-452 A & B	$\leq$ 100 mr/hr	2	B
Liquid Radwaste Discharge Isolation	RMV-RM-2	(2)	1	C
Main Control Room Ventilation Isolation	(RMV-RM_1)	$4x10^3$ CPM	1	D
Mechnical Vacuum Pump Isolation	RMP-RM-251 A-D	3 times normal full power background. Alarm at 1.5 times normal full power background.	4	E

NOTES FOR TABLE 3.2.D

1. Action required when component operability is not assured.

A.(1) If radiation level exceeds 1.0 ci/sec (prior to 30 min. delay line) for a period greater than 15 consecutive minutes, the off-gas isolation valve shall close and reactor shutdown shall be initiated immediately and the reactor placed in a cold shutdown condition within 24 hours.

A.(2) Refer to Section 2.4.3.a.7 of the Environmental Technical Specifications.

B. Cease refueling operations, isolate secondary containment and start SBGT.

C. Refer to Sections 2.4.1.b of the Environmental Technical Specifications

D. Refer to Section entitled "Additonal Safety Related Plant Capabilities".

NOTES FOR TABLE 3.2.D (cont.)

E. Refer to Section 3.2.d.5 and the requirements for Primary Containment Isolation on high main steam line radiation. Table 3.2.A

2. Trip setting to correspond to Specification 2.4.1.b.1 of the Environmental Technical Sepcifications.

prevention of critical heat flux in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented. The downscale trips are set at 2.5 indicated on scale.

The flow comparator and scram discharge volum high level components have only one logic channel and are not required for safety.

The RSCS Rod Group C Bypass function is required only during the first 6500 MWD/T of the initial core loading. This function is provided by two pressure transducers which sense turbine first stage pressure which is then correlated with core thermal power. This bypass function assures that control rod worths are controlled as described in the Bases for Sepcification 3.3.B.3.

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

The effective emergency core cooling for small pipe breaks, the HPCI system, must function since reactor pressure does not decrease rapid enough to allow either core spray of LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

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

Both instruments are required for trip but the instruments are so designed that any instrument failure gives a downscale trip. The trip setting of 1.0 ci/sec (prior to 30 min. delay) provides an improved capability to detect fuel pin cladding failures to allow prevention of serious degradation of fuel pin cladding integrity which might result from plant operation with a misoriented or misloaded fuel assembly. This limit is more restrictive than 0.39 ci/sec noble gas release rate at the air ejectors (after 30 min. delay) which was used as the source term for an accident analysis of the augmented off-gas system. Using the .39 ci/sec source term, the maximum off-site total body dose would be less than the 5 rem limit.

Two radiation monitors are provided which initiate the Reactor Building Isolation function and operation of the standby gas treatment system. The trip is actuated by one hi-hi or two downscale indications.

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 prescibed 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 ≥1.20 for 7x7 fuel and ≥ 1.22 for 8x8 fuel 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 precribed 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 >1.20 for 7x7 fuel and >1.22 for 8x8 fuel times K, 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

A. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10CFR50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm$ 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10CFR50 Appendix K limit. The limiting value for APLHGR is shown in Figure 3.11-1.

The calculational procedure used to establish the APLHGR shown on Figure 3.11.1 is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10CRF50. A complete discussion of each code employed in the analysis is presented in References 1 and 3. Differences in this analyses as compared to previous analyses performed with Reference 1 are discussed in Reference 3.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Table 3.11-1.

Table 3.11-1

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS

PLANT PARAMETERS:

Core Thermal Power	2486 MWt which corresponds to 105% of rated steam flow
Vessel Steam Output	10.04×10^6 lbm/h which corresponds to 105% of rated steam flow
Vessel Steam Dome Pressure	1055 psia
Recirculation Line Break Area for Large Breaks - Discharge	2.4 ft^2 (DBA) 1.9 (80% DBA)
- Suction	4.1 ft^2
Recirculation Line Break Area for Small Breaks	1.0, 0.1 and 0.35
Number of Drilled Bundles	All

FUEL PARAMETERS:

	Fuel Type	Fuel Bundle Geometry	Peak Technical Specification Linear Heat Generation Rate (KW/ft)	Design Axial Peaking Factor	Initial Minimum Critical Power Ratio*
A.	IC Type 2	7 x 7	18.5	1.5	1.2
B.	IC Type 3	7 x 7	18.5	1.5	1.2
C.	8D250	8 x 8	13.4	1.4	1.2
D.	8D274	8 x 8	13.4	1.4	1.2

*To account for the 2% uncertainty in bundle power required by Appendix K, the SCAT calculation is performed with an MCPR of 1.18 (i.e., 1.2 divided by 1.02) for a bundle with an initial MCPR of 1.20.

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.

3. Loss of Coolant Accident Analysis Report for Cooper Nuclear Station, NEDO-24045, August 1977.

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 3 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 1.06, 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.

The limiting transient which determines the required steady state MCPR limit is the loss of 100°F feedwater heater. This transient yields the largest Δ MCPR. When added to the safety limit MCPR of 1.06 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 4.4 of NEDO-20360[2] and on core parameters shown in Table 6-1 of Reference 5.

The evaluation of a given transient begins with the system initial parameters shown in Table 6-1 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 the 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 1.20 for 7x7 and 1.22 for 8x8 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.

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 sump 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 1.20 for 7x7, 1.22 for 8x8 are as great as 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. "General Electric BWR Generic Reload Application for 8 x 8 Fuel", Supplement 4 to Revision 1, 4/1/76 (NEDO-20360).

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. "Cooper Nuclear Station Reload No, 2 License Amendment Submittal", June 1977 (NEDO-24033).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 39 TO FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

0.0 INTRODUCTION

By letter dated July 20, 1977, supplemented by letters dated September 1 and October 3 and 13, 1977, Nebraska Public Power District (NPPD - the licensee) requested an amendment to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS). The amendment authorizes operation of CNS with 64 General Electric 8 x 8 reload fuel assemblies of a type previously authorized for use at CNS.

1.0 DISCUSSION

NPPD has proposed to operate CNS with 64 8 x 8 (8D274L) reload fuel assemblies with 80 mil channels. The enrichment of each new 8 x 8 reload fuel assembly is 2.74 wt. % U-235. The balance of the 548 element core will consist of exposed 8 x 8 and 7 x 7 fuel assemblies previously loaded for cycles 1 and 2. All Cycle 3 reload and exposed assemblies will have two 9/32-inch holes drilled in each lower tie plate, with the 1-inch bypass flow holes in the core support plate plugged. The 9/32" holes in the fuel assembly lower tie plates permit cooling water to flow into the bypass region between fuel assemblies to cool the in-core nuclear instrumentation, and the plugging of 1" bypass flow holes was done to eliminate in-core vibrations. (Reference 11)

The reactor is expected to operate in the configuration just described at the licensed power level of 2381 MWt for a short operating Cycle 3 of approximately 6 months. In support of the reload application the licensee has provided the GE BWR Reload 2 licensing submittal for Cooper (Reference 1), proposed Technical Specifications changes (Reference 2), a Loss of Coolant Accident (LOCA) analysis report (Reference 3) and responses to NRC requests for additional information (References 4 and 10).

The information presented in the licensing submittal closely follows the guidelines of Appendix A of NEDO-20360 (Reference 5). Although later supplements to this report are undergoing review by the staff, portions of this topical have been found applicable for reactors containing 8 x 8 reload fuel and are acceptable to the staff when supplemented with information required by our status report (Reference 6). The supplemental information provided by the licensee and the staff's evaluation thereof are summarized below.

2.0 EVALUATION

2.1 Nuclear Characteristics

For Cycle 3 operation of CNS, a total of 64 fresh 8 x 8 fuel bundles with an enrichment of 2.74% U235 by weight will be loaded into the core. In addition, 72 exposed 8D250 and 48 exposed 8D274 8 x 8 assemblies from reload 1 and 364 7 x 7 assemblies from the original loading will remain in the core (Reference 1).

Thus, for Cycle 3 approximately 12% of the 548 fuel assemblies will be fresh, 22% will have been exposed for one cycle, and 66% will have been exposed for two cycles. As indicated by the loading diagram presented in Reference 1, no more than two fresh assemblies are to be loaded into any four bundle array, and all fresh fuel will be located near the outer edge of the core.

The data in Reference 1 indicate that the nuclear characteristics of the Reload 3 core are similar to the previous core. Thus, the total control system worth, and temperature and void dependent behavior of the reconstituted core will not differ significantly from those values previously reported for the CNS reactor. The shutdown margin of the reconstituted core meets the Technical Specification requirement that the core be at least 0.38% Δk subcritical in the most reactive operating state with the most reactive rod fully withdrawn and with all the other rods fully inserted. For Cycle 3, the minimum shutdown margin has been calculated to be 0.013 Δk which occurs at beginning of cycle.

The information presented in Reference 1 indicates that a boron concentration of 600 ppm in the moderator will bring the reactor subcritical by at least 0.03 Δk at 20°C, xenon free. Therefore, the alternate shutdown requirement of the General Design Criteria is met by the Standby Liquid Control System.

The Technical Specification requirement for the storage of fuel for CNS is that the effective multiplication factor (k_{eff}) of the fuel as stored in the fuel storage rack is equal to, or less than, 0.90 for normal storage conditions. This is achieved if the uncontrolled k_∞ of any single fuel bundle is less than 1.30 at 65°C. The peak uncontrolled k_∞ of all 8 x 8 fuel bundles, within the applicable exposure and fuel pool temperature range, is less than 1.30 so that storage requirements for CNS are met (Reference 5).

The void and Doppler coefficients of reactivity for Cycle 3 are given in Table 5-1 of Reference 1. The void coefficient of reactivity at the core average void fraction is expected to vary from -11.94 to -10.95×10^{-4} $(\Delta k/k)/\Delta V$. The Doppler coefficient at a fuel temperature of 650°C will range from -1.162×10^{-5} to 1.200×10^{-5} $(\Delta k/k)/\Delta T$.

Thus, based on our review of the information presented in the CNS licensing submittal (Reference 1) as supplemented by applicable portions of the generic 8 x 8 reload report (Reference 5) and the staff's acceptance thereof (Reference 6), we have determined that the nuclear characteristics and performance of the reconstituted core for Cycle 3 are similar to those of the earlier fuel cycle and are acceptable.

2.2 Mechanical Design

The Reload fuel has the same mechanical configuration and fuel bundle enrichments as the 8D274L assemblies described in the 8 x 8 generic reload report (Reference 5) except that two 9/32 inch holes are drilled in the lower tie plate of the reload assemblies to provide bypass flow as discussed in paragraph 1.0, above. Also, the improved water rod design described in Section 3.1 of Reference 5 has been adopted.

The generic 8 x 8 reload report (Reference 5), supplements of which are under review, has been found acceptable for use for reactors containing 8 x 8 reload fuel when supplemented with information required by our status report (Reference 6) on the GE generic report evaluation. On the basis of our review of the generic 8 x 8 reload report and the reload submittal, we conclude that the mechanical design of the CNS Reload 2 is acceptable.

2.3 Thermal-Hydraulics

The GE generic 8 x 8 fuel reload topical report (Reference 5) and GETAB (Reference 7) are referenced to provide the description of the thermal-hydraulic methods which were used to calculate the thermal margins. Application of the GETAB establishes:

(1) the fuel damage safety limit,

(2) the limiting conditions of operation (LCO) such that the safety limit is not exceeded for normal operation and anticipated transients, and

(3) the limiting conditions of operation such that the initial conditions assumed in the accident analyses are satisfied.

We have evaluated the CNS Cycle 3 thermal margins based on the GETAB report (Reference 7) and plant specific input information provided by the licensee. The staff evaluation of these margins is reported herein.

2.3.1 Fuel Cladding Integrity Safety Limit MCPR

The fuel cladding safety limit minimum critical power ratio (MCPR) of 1.06 has been established, based on the GETAB (Reference 7) statistical analysis, to assure that 99.9% of the fuel rods in the core will not experience boiling transition during abnormal operational transients (Reference 8). This limit is applied for both core-wide and localized transients or pertubations to the expected CPR distribution.

The uncertainties in core and system operating parameters and the GEXL correlation uncertainties assumed for Cycle 3 operation of CNS are the same as those used for the original statistical analysis (Table 4-5 of Reference 5) on which the fuel cladding safety limit MCPR is based. The bundle power distribution for Cycle 3 is expected to include fewer high power bundles than the distribution assumed for the original statistical analysis as is indicated by comparing Figure 4-2 with Figures 4-4.1 through 4-4.4 of Reference 5. Therefore, it is conservative to apply the fuel cladding safety limit MCPR of 1.06 to Cycle 3 operation of CNS.

2.3.2 Operating Limit MCPR

Various transients or perturbations to the CPR distribution could reduce the MCPR below the intended operating limit during Cycle 3 operation of CNS. The most limiting operational transients have been analyzed to determine which could potentially induce the largest reduction in MCPR.

The transients evaluated were the turbine trip with failure of the bypass valves, loss of a 100° feedwater heater, feedwater controller failure, and the control rod withdrawal error. Initial conditions and transient input parameters as specified in Table 4-3, Table 6-1 and Figure 6-1 of Reference 1 were assumed. For most of the parameters which vary with exposure, end-of-cycle values were assumed because the reduction in CPR for the most severe transients is largest for end-of-cycle (Reference 1). The exceptions to this are the local peaking factor and GEXL R-factor which were conservatively assumed to be those of fresh fuel.

The input to the transient calculations and the application of the analysis methods of Reference 5 have been reviewed and determined to provide appropriate conservatism for determination of the operating limit MCPR for CNS during Cycle 3.

The calculated reductions in CPR during each of the operational transients have been tabulated in Table 4-2 of Reference 1. For 7 x 7 fuel, the most severe transient is the loss of a 100°F feedwater heater with a ΔCPR of 0.14. For 8 x 8 fuel, either the loss of a 100°F feedwater heater or a rod withdrawal error would produce the maximum ΔCPR of 0.16.

Addition of these ΔCPR's to the safety limit MCPR of 1.06 gives the operating limit MCPR's 1.20 for 7 x 7 and 1.22 for 8 x 8 fuel. These operating limit MCPR's will provide adequate margin to the 1.06 safety limit MCPR during Cycle 3.

2.3.3 Operating MCPR Limits for Less than Rated Power and Flow

For the limiting transient of recirculation pump speed control failure at lower than rated power and flow condition, the licensee will conform to the limiting conditions for operation previously approved and stated in the Technical Specifications. This requires that for core flows less than the rated flow, the licensee maintain the MCPR greater than the operating minimum values. The minimum MCPR values for less than rated flow are the rated flow value multiplied by the respective K_f factors appearing in Figure 3.11-2 of the Technical Specifications. The K_f factor curves were generically derived and assure that the most limiting transient occurring at less than rated flow will not exceed the safety limit MCPR of 1.06. We conclude that the calculated consequences of the anticipated operational transients do not violate the thermal limits of the fuel or the pressure limits of the reactor coolant boundary.

2.4 Accident Analysis

2.4.1 Control Rod Drop Accident

The licensee has performed a plant specific calculation of the peak fuel enthalpy which could be reached should the worst case control rod drop accident occur during Cycle 3. The resulting peak fuel enthalpy has been calculated to be below the design limit of 280 cal/gm, and the associated radiological consequences would be far below the guidelines set forth in 10 CFR 100.

The input parameters for the calculations have been provided in Reference 2 and 4, and the calculational methods were described in Reference 5, 6, and 9. Both the input and methods have been reviewed by the staff, and the licensee's analysis is acceptable.

On this basis, it is concluded that no control rod drop accident could occur during Cycle 3 which would pose a significant threat or hazard to the health and safety of the public.

2.4.2 Fuel Handling Accident

With respect to fuel handling accidents, in Reference 1 the applicant noted that the description and analyses of this event provided in the FSAR and discussed in the generic 8 x 8 reload report (Reference 5) are applicable to this reload. That is, the total activity released to the environment and the radiological exposures for the 8 x 8 fuel will be less than those values presented in the FSAR for the 7 x 7 core. As identified in the FSAR the radiological exposures for this accident with 7 x 7 fuel are well below the guidelines set forth in 10 CFR 100. Therefore, it is concluded that the consequences of this accident for the 8 x 8 fuel will also be well below the 10 CFR 100 guidelines.

2.4.3 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 and 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 reevaluation of ECCS cooling 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 results.

In December of 1976, the NRC staff was informed that certain input errors and computer code errors had been made in the evaluation that was provided under the requirements described above. An Order for Modification of License was issued to NPPD on March 11, 1977 (Reference 12) which required that corrected revised calculations fully conforming to the requirements of 10 CFR 50.46 be provided for Cooper Nuclear Station as soon as possible.

The required revised analysis (Reference 3), which is based on the approved new GE ECCS models (Reference 13) and which includes correct input, has been submitted by the licensee. Furthermore it has been established in References 3 and 4 that CNS qualifies as one of the plants which may reference the NRC evaluation of the new ECCS analysis methods as applied to the James A. Fitzpatrick Nuclear Power Plant (References 14 and 15).

The submitted analysis demonstrates that continued operation with the linear heat generation rate limits in the current Technical Specifications will ensure that the peak clad temperature during the worst case loss of coolant accident possible for Cycle 3 will not exceed the safety limit of 2200^{o}F. On the basis of its review of the submitted analysis the staff concludes that operation of CNS during Cycle 3 will conform to the requirements of 10 CFR Part 50.46 and Appendix K and will provide reasonable assurance that the health and safety of the public will not be endangered.

2.4.4 Steam Line Break Accident

Steam line break accidents which are postulated to occur inside containment are covered by the ECCS analysis discussed in section 2.4.3. The analysis of steam line break accidents occurring outside containment as presented by the licensee is acceptable based on our generic review of NEDO-20360 (References 5 and 6).

2.4.5 Fuel Loading Error

In References 3 and 4, the licensee has shown that the worst fuel loading error which could occur for Cycle 3 would be the misplacing of a fresh 8 x 8 assembly in a 7 x 7 site. Although the misloading would not result in a linear heat generation rate in excess of the LHGR safety limit, the resulting critical power ratio would be significantly below the MCPR safety limit. Consequently, should the error occur, several fuel rods within the misloaded assembly would be expected to experience boiling transition which would eventually lead to severe degradation of the cladding and fuel pin activity release.

One method of avoiding these consequences of the fuel loading error would be the adoption of an operating limit MCPR restrictive enough to ensure that transition boiling would not occur. The licensee has instead proposed that the present Technical Specification limit on the steam jet air ejector off-gas radioactivity release rate be lowered to 1.0 curie per second (ci/sec) from the current limit of 5.4 ci/sec. If the air ejector radioactivity reaches or exceeds 1.0 ci/sec for 15 minutes, automatic closure of the off-gas isolation valve and manual shutdown of the reactor will be initiated immediately. Based on estimates for similar Technical Specifications issued for other plants (Reference 15), this relatively low offgas release limit could be exceeded by the rapid, sequential release of the noble gas inventory in the gaps of only a few fuel pins if the failure occurred over a 15 minute period. Therefore, a fuel loading error involving activity release from several pins would be detectable by this method, and the reactor would be shutdown.

In addition, the CNS Technical Specifications on reactor coolant radioactivity surveillance and main steam line radiation monitoring provide additional means to detect fuel pin cladding failures. The licensee is being required to submit an analysis of the means to detect fuel pin failures to provide added assurance that they may be detected rapidly and reliably. The analysis will include substantiation of the relation between pin failures, offgas rates and primary coolant activity, among other parts of the estimation of fuel failure detection capabilities at CNS. Any radioactivity released to the offgas by sudden, significant fuel pin cladding failures prior to reactor shutdown would be retained on the charcoal beds of the offgas treatment system. Even in the unlikely event that all the activity collected on the charcoal beds was released by some other event, the resultant offsite exposures would be well within the dose guidelines of 10 CFR Part 100.

The staff therefore concludes that (1) the fuel pin cladding failures which might result from plant operation with a misloaded fuel bundle would be detected by the air ejector offgas radiation monitor, (2) the proposed Technical Specification change (limiting the air ejector offgas radioactivity to 1.0 Curie per second and requiring immediate initiation of reactor shutdown and automatic closure of the offgas ioslation valve if the limit is exceeded for

15 minutes) provides greater assurance that reactor operation would not continue if significant increases in the offgas radioactivity were observed, and (3) that no threat to the health and safety of the public would be involved.

2.5 Overpressure Analysis

The licensee has presented an analysis to demonstrate that during the most severe overpressure event, an adequate margin exists between the peak vessel pressure and the ASME code allowable vessel pressure which is 110% of the vessel design pressure (Reference 1). The event analyzed was the closure of all main steam line isolation valves (MSIV) with indirect (high flux) scram.

The input to the calculation is listed in Table 6-1 of Reference 1, and conservatively includes end of cycle scram characteristics, void coefficient and Doppler coefficient.

A sensitivity study was presented which demonstrates that should the MSIV closure transient be initiated at the maximum dome pressure allowed by the Technical Specifications rather than that assumed for the analysis, there would be adequate margin to the pressure limit (Reference 4). It has also been shown that the increase in peak vessel pressure during an MSIV closure due to a failed safety valve would not be sufficient to reduce the margin to the limit by a significant amount (Reference 4).

Therefore, based on the analysis and sensitivity studies submitted by the licensee, the overpressure analysis for CNS for Cycle 3 has been found acceptable.

2.6 Thermal Hydraulic Stability Analysis

The thermal hydraulic stability analyses and results are described in References 5 and 1, respectively. The results of the Cycle 3 analysis show that the 8 x 8 channel hydrodynamic stability, at either rated power and flow conditions or at the low end of the flow control range, is within the operational design guide in terms of decay ratio. Calculations were also performed by the licensee to assess the reactor power dynamic response at the two aforementioned reactor operating conditions. The results of this analysis showed that the reactor core decay ratios at both conditions are well within the operational design guide decay ratio. These results are acceptable to the NRC staff.

The NRC staff has expressed generic concerns regarding the least stable reactor condition allowed by Technical Specifications. This condition could be reached during an operational transient from high power where the plant sustains a trip of both recirculation pumps. The concerns are motivated by increasing decay ratios as equilibrium fuel cycles are approached and as fuel designs improve. The staff concerns relate to both the consequences of operating at an ultimate decay ratio and the capacity of analytical methods to accurately predict decay ratios. The General Electric Company is addressing the staff concerns through meetings, topical reports and a test program.

The staff previously imposed a requirement on CNS which restricted operation in the natural circulation flow mode. The licensee adopted this Technical Specification limitation in Amendment No.32 dated November 10, 1976. The restriction provides a significant increase in the reactor core stability margins. On the basis of the foregoing, the NRC staff considers the thermal-hydraulic stability of Cooper Nuclear Station to be acceptable for cycle 3 operation.

3.0 PHYSICS STARTUP TESTING

The licensee will carry out a startup testing program which will provide additional assurance that the Cycle 3 core as loaded is consistent with input to the transient and accident analyses contained in the reload licensing submittal (Reference 1). The results of the tests will be available within 90 days of startup completion of the test program.

4.0 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.

5.0 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.

Date: October 14, 1977

REFERENCES

1. General Electric Boiling Water Reactor Reload No. 3 Licensing Submittal for Cooper Nuclear Power Station Unit 1, NEDO-24033, June 1977.

2. Nebraska Public Power District Application for Amendment of Facility Operating License DPR-46, July, 1977.

3. Cooper Nuclear Station Emergency Core Cooling Systems Reevaluation, September 1, 1977.

4. Additional Information on Reload for Cooper Nuclear Station, October 3, 1977.

5. "General Electric Reload Licensing Application for 8 x 8 Fuel," Revision 1, Supplement 3, September 1975, NEDO-20360.

6. Status Report on the Licensing Topical Report "General Electric Boilding Water Generic Reload Application for 8 x 8 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.

7. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," NEDO-10958, 73 NED9, Class 1, November 1973.

8. General Electric Letter (John A. Hinds) to U. S. Atomic Energy Commission (Walter Butler), "Responses to the Third Set of AEC Questions on the General Electric Licensing Topical Reports, NEDO-10958 and NEDE-10958, "General Electric BWR Thermal Analysis Bases (GETAB): Data, Correlation and Design Application," July 24, 1974.

9. "Rod Drop Accident Analysis for Large Boilding Water Reactors", March 1972, NEDO-10527.

10. Letter J.M. Pilant (NPPD) to D.K. Davis (NRC), Additional Information on Reload 2 for Cooper Nuclear Station, October 13, 1977.

11. "Safety Evaluation Report on the Reactor Modification to Eliminate Significant In-Core Vibration in Operating Reactors with 1-inch Bypass Holes in the Core Support Plate," by Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, February 1976.

12. Letter, Dennis L. Ziemann (NRC) to J. M. Pilant (NPPD) Re: Cooper Nuclear Station Unit No. 1 dated March 11, 1977.

13. "Safety Evaluation for General Electric ECCS Evaluation Model Modifications," letter from K. R. Goller (NRC) to G. G. Sherwood (GE), dated April 12, 1977.

14. Letter, Darrell G. Eisenhut (NRC) to E. D. Fuller (GE), Documentation of the Reanalysis Results for the Loss of Coolant Accident (LOCA) of Lead and Non-lead plants." June 30, 1977.

15. Safety Evaluation Report for the James A. Fitzpatrick Nuclear Power Plant, Amendment No. 30, September 16, 1977.

(7590-01)

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. 39 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 its date of issuance.

The amendment authorized operation of the facility with 64 additional General Electric 8 x 8 reload fuel assemblies which replace exposed 7 x 7 fuel assemblies.

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 §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 July 20, 1977 and supplements dated September 1, October 3 and 13, 1977, (2) Amendment No. 39 to Licence No. DPR-46, (3) the Commission's related Safety Evaluation, (4) the Commission's Safety Evaluation for General Electric ECCS Evaluation Model Modifications transmitted to the General Electric Company by K. R. Goller letter dated April 12, 1977, (5) Letter dated June 30, 1977 from D. G. Eisenhut (NRC) to E. D. Fuller (GE) on Documentation of the Reanalysis Results for the loss of Coolant Accident (LOCA) of Lead and Non-Lead Plants, and (6) Safety Evaluation Report by the Office of Nuclear Reactor Regulation for the James A. Fitzpartrick Power Plant dated September 16, 1977, issued with Amendment No. 30 in Docket No. 50-333. 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 14th day of October, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors