Docket No. 50-298

Nebraska Public Power District
ATTN: Mr. J. M. Pilant, Director
Licensing and Quality Assurance
Post Office Box 499
Columbus, Nebraska 68601

NOV 1976

Gentlemen:

In response to your requests dated April 7 and August 9, 1976, supplemented by letters dated July 20, September 20, October 8, 9, 15, 20 and 26 and November 4 and 10, 1976, the Commission has issued the enclosed Amendment No. 32 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station Unit No. 1.

The amendment consists of Technical Specification changes to authorize operation with (1) 120 General Electric 8 x 8 type reload fuel assemblies, (2) bypass flow holes drilled in the lower tie plates of all fuel assemblies in the core (with the exception of Type 1 assemblies) as part of the facility modifications to eliminate in-core vibration of instrument and source tubes, (3) a modified Rod Sequence Control System, and (4) modifications to improve the performance of the Low Pressure Coolant Injection System of the Emergency Core Cooling System. Some of your requested changes have been modified to meet our requirements. These changes have been discussed with and accepted by your staff.

This amendment constitutes approval for operation of Cooper Nuclear Station for fuel cycle 2.

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

Sincerely,

Original signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Leo Lessor, NPPD and Eric Johnson OI&E advised by telephone on 11/10/76 that these amendments had been approved DLZiemann

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

SEE PREVIOUS YELLOW FOR CONCURRENCE AND DISTRIBUTION

DOR:AD/OT	DOR:RSB/OT	DOR:PSB/OT	DOR:ORB #2	
DEisenhut	RBaer	WButler	JGuibert	
11/11/76	11/10/76	11/10/76	11/10/76	

OFFICE		DOR:ORB #2	DOR:ORB #2	EOLD	DOR:ORB #2	DOR:AD/ORs
SURNAME		MHFletcher:ah	RMDiggs	JGray	DLZiemann	KRGoller
DATE		11/10/76	11/10/76	11/11/76	11/10/76	11/10/76

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Division of Operating Reactors

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2. Safety Evaluation
3. Notice

		DOR:RSB/OT RBaer 11/ /76	DOR:PSB/OT WButler 11/ /76	DOR:ORB #3 JGuibert 11/10/76	
OFFICE➤	DOR:ORB #2	DOR:ORB #2	OELD	DOR:ORB #2	DOR:AD/ORs
SURNAME➤	MHFletcher:ah	RMDiggs	J.R.GRAY	DLZiemann	KRGoller
DATE➤	11/4/76	11/3/76	11/9/76	11/ /76	11/ /76

cc w/enclosures:
Gene Watson, Attorney
Barlow, Watson & Johnson
P. O. Box 81686
Lincoln, Nebraska 68501

Mr. Arthur G. Gehr, Attorney
Snell & Wilmer
400 Security Building
Phoenix, Arizona 85004

Auburn Public Library
118 - 15th Street
Auburn, Nebraska 68305

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

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

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

cc w/enclosures and cy of NPPD
filings dtd. 4/7/76, 8/9/76,
7/20/76, 9/20/76 and
10/8, 9, 15, 20/76, and 11/4,10/76:
Director, Department of Environmental
Control
Executive Building, 2nd Floor
Lincoln, Nebraska 68509

also filings 6/21/74
6/30/74
10/1/74

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 32
License No. DPR-46

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

   A. The applications for amendment by Nebraska Public Power District (the licensee) dated April 7 and August 9, 1976, as supplemented by letters dated July 20, September 20, October 8, 9, 15, 20 and 26 and November 4 and 10, 1976, comply 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.

OFFICE➤						
SURNAME➤						
DATE➤						

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance:

OFFICE ➤						
SURNAME ➤						
DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NO. 72

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. The changed areas on the revised pages are reflected by a marginal line.

Pages to be Removed	New Pages to be Inserted
i, ii, iv	i, ii, iv
7	7
8	8
10	10
14	14
15	15
16	16
-	16a (Addition)
17	17
19	19
20	20
22	22
26	26
27	27
31	31
42	42
43	43
54	54
55	55
61	61
62	62
62a (Delete)	-
71	71
72	72
94	94
-	94a (Addition)
95	95
-	95a (Addition)
96	96
96a (Delete)	-
97	97
98	98
100	100
100a (Delete)	-
101	101
101a	101a
104	104
110	110
115	115
137	137
151	151
197	197
211	211
-	211a (Addition)
212	212
-	212a (Addition)
214b	214b
214c	214c
214d	214d
214e	214e
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RADIOLOGICAL TECHNICAL SPECIFICATIONS

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Amendment No. 13, 32

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TEMPORARY RESTRICTIONS

1. When the results of the FitzPatrick hot (up to full power) vibration tests are available, they will be evaluated and compared with the results of CNS. In the event that the FitzPatrick tests, or results of startup programs and power operation of Browns Ferry Unit 1, indicate potential problems in areas which could not have been detected in CNS due to differences in instrumentation, appropriate corrective action will be required.

2. Drilling of alternate flow path holes in the lower tie plates of unirradiated fuel bundles at the CNS site is permitted provided the procedures of Section 3 of General Electric Document NEDE 21156 are followed and GE personnel, or personnel properly trained by the General Electric Company, perform the drilling.

3. Machining of alternate flow path holes in the lower tie plates of irradiated fuel bundles by means of electrical-discharge machining at the CNS site is permitted provided that the procedures described in letters from G. C. Ross, GE, to D. G. Eisenhut, NRC, dated April 1, and April 23, 1976 are followed and GE personnel, or personnel properly trained by the General Electric Company, perform the machining.

4. The above restrictions apply until removed by written instructions of the NRC staff.

1.1.D (Cont'd)

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

2.1.A (Cont'd)

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 = 2.61 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. <u>APRM Flux Scram Trip Setting (Refuel or Start and Hot Standby Mode)</u>

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. <u>IRM</u>

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

2.1.A (Cont'd)

d. APRM Rod Block Trip Setting

The APRM rod block trip setting shall be:

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

where:

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

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate 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:

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

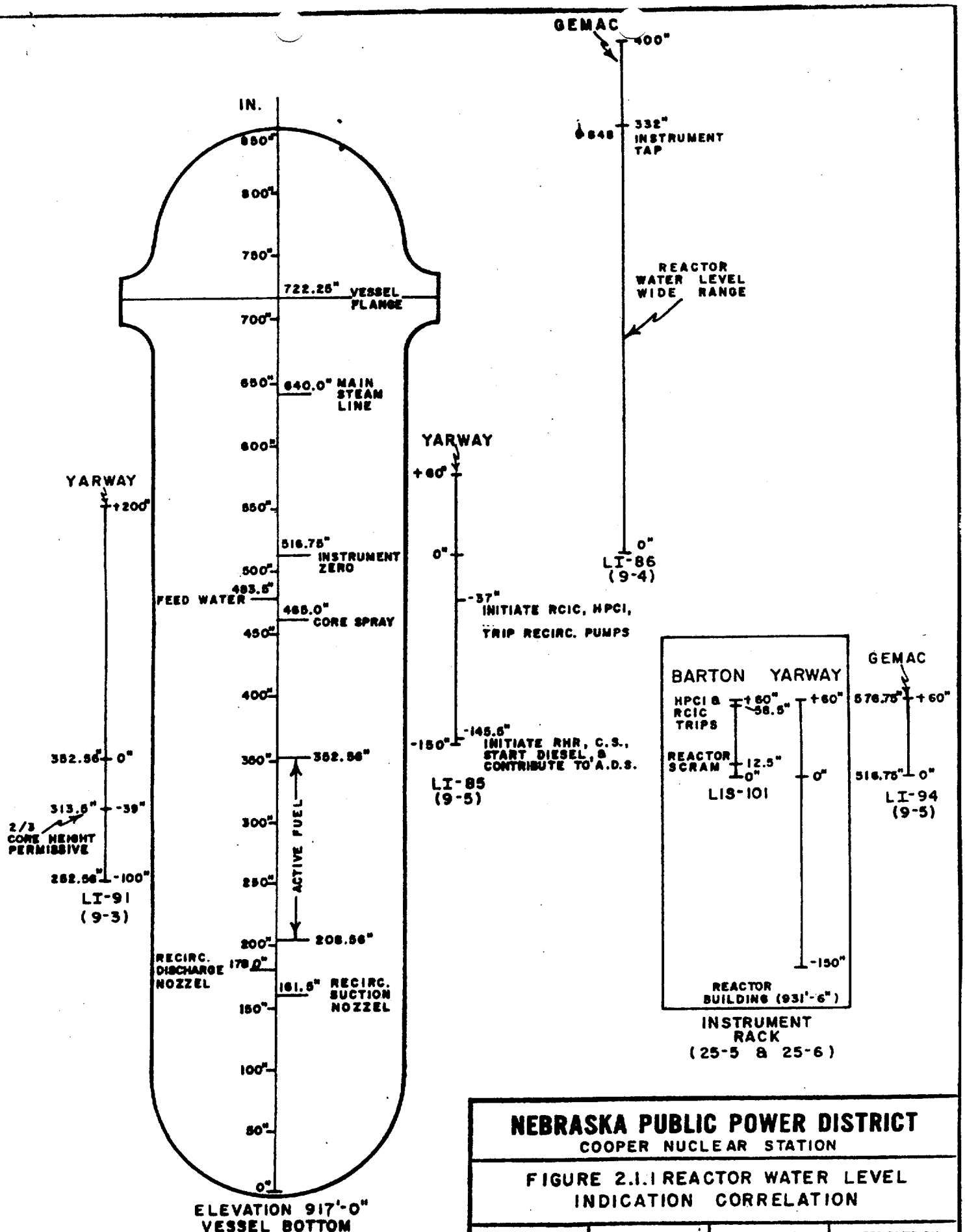
where:

A = 2.61 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.



NEBRASKA PUBLIC POWER DISTRICT
COOPER NUCLEAR STATION

FIGURE 2.1.1 REACTOR WATER LEVEL INDICATION CORRELATION

DRAWN BY	TRACED BY	CHECKED BY	APPROVED BY
SCALE		DRAWING NO.	

Amendment No. 16, 32

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
Bypass void effect on TIP	3.58 (core midplane) 4.25 (core exit)
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)

1.1 Bases: (cont'd)

TABLE 1.1-3

RELATIVE BUNDLE POWER DISTRIBUTION

USED IN THE GETAB STATISTICAL ANALYSIS

Range of Relative Bundle Power	Percent of Fuel Bundles Within Power Interval
1.375-1.425	6.8
1.325-1.375	3.7
1.275-1.325	16.1
1.225-1.275	11.0
1.175-1.225	6.8
1.125-1.175	5.2
1.075-1.125	9.4
1.025-1.075	4.2
0.975-1.025	2.6
0.925-0.975	4.2
0.875-0.925	2.6
0.825-0.875	4.2
0.775-0.825	1.5
0.725-0.775	3.1
0.675-0.725	0.5
0.625-0.675	0
0.575-0.625	2.6
0.525-0.575	3.1
0.475-0.525	1.5
0.425-0.475	0
0.375-0.425	3.1
0.325-0.375	2.1
0.275-0.325	3.1
0.225-0.275	2.1
0.175-0.225	0.5
	100.0

TABLE 1.1-4

R-FACTOR DISTRIBUTION USED IN GETAB STATISTICAL ANALYSIS

(7x7/8x8 RELOAD CORES)

Rod Sequence No.	R-Factor 7x7 Rod Array	R-Factor 8x8 Rod Array
1	1.098	1.100
2	1.088	1.100
3	1.078	1.095
4	1.066	1.095
5	1.056	1.093
6	1.046	1.093
7	1.046	1.091
8	1.031	1.076
9	1.026	1.076
10	1.025	1.071
11	1.025	1.071
12	1.025	1.068
13	1.024	1.052
14	1.024	1.052
15	1.019	1.051
16	1.013	1.051
17	1.012	1.039
18	1.011	1.039
19	1.011	1.036
20	1.011	1.036
21	1.010	1.036
22	1.010	1.036
23	1.004	1.029
24	1.004	1.029
25	1.000	1.028
26	0.998	1.028
27	0.992	1.027
28	0.989	1.027
29	0.988	1.022
30	0.978	1.022
31	0.968	1.015
32	0.964	1.015

1.1 Bases: (Cont'd)

TABLE 1.1-4 (Cont'd)

Rod Sequence No.	R-Factor 7x7 Rod Array	R-Factor 8x8 Rod Array
33	0.964	1.015
34	0.961	1.015
35	0.961	1.014
36	0.927	1.005
37	0.917	1.005
38	0.915	0.986
39	0.907	0.973
40	0.904	0.973
41	0.902	0.969
42	0.898	0.969
43	0.894	0.957
44	0.861	0.957
45	0.859	0.953
46	0.682	0.953
47	0.679	0.951
48	0.624	0.951
49	0.614	0.948
50		0.948
51		0.945
52		0.943
53		0.943
54		0.939
55		0.939
56		0.931
57		0.931
58		0.926
59		0.739
60		0.739
61		0.714
62		0.714
63		0.703
64		0.020

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 greatest 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 1.25 for 7x7 fuel and 1.28 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

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 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 2.61 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 > 1.25 for 7x7 fuel and 1.28 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 accommodate 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 2.61 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 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. "Cooper Nuclear Station Reload No. 1 Licensing Amendment Submittal", July 1976 (NEDO-21292).

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 1279 psig at the vessel bottom. This represents a 96 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 1 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. 1 License Amendment Submittal", July 1976 (NEDO-21292).

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 2.61 for 7x7 fuel and 2.44 for 8x8 fuel.

C. During reactor power operation with TPF $\geq$ 2.61 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 = 2.61 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 2.61 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

flux is very high. Therefore, with TPF < 2.61 for 7x7 fuel and 2.44 for 8x8 fuel there are no technical specification requirements for calculating MCPR. With TPF greater than 2.61 for 7x7 fuel and 2.44 for 8x8 fuel MCPR 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 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.

COOPER NUCLEAR STATION
TABLE 3.2.B (PAGE 2)
RESIDUAL HEAT REMOVAL SYSTEM (LPCI MODE) CIRCUITRY REQUIREMENTS

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required When Component Operability Is Not Assured
Drywell High Pressure	PC-PS-101 A,B,C & D	$\leq$2 psig	2	A
Reactor Low Water Level	NBI-LIS-72, A,B,C, & D #1	$\geq$-145.5" Indicated Level	2	A
Reactor Vessel Shroud Level Below Low Level Trip	NBI-LITS-73, A & B #1	$\geq$-39 Indicated Level	1	B
Reactor Low Pressure	RR-PS-128, A & B	$\leq$75 psig	1	B
Reactor Low Pressure (Injection Valve Permissive)	NBI-PS-52A & C NBI-PIS-52B & D	$\leq$450 psig	1	A
Drywell Pressure Containment Spray	PC-PS-119, A,B,C, &D	$\leq$2 psig	2	A
RHR Pump Discharge	RHR-PS-120, A,B,C,&D RHR-PS-105, A,B,C,&D	100 $\leq$ P $\leq$ 165 psig 100 $\leq$ P $\leq$ 165 psig	2 2	A A
Reactor Low Pressure (Recirc. Discharge Permissive)	NBI-PS-52A & C NBI-PIS-52B & D	185$\leq$p$\leq$235 psig	1	A

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TABLE 3.2.B (PAGE 3)
RESIDUAL HEAT REMOVAL SYSTEM (LPCI MODE) CIRCUITRY REQUIREMENTS

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required When Component Operability Is Not Assured
RHR Pump Low Flow	RHR-dPIS-125 A & B	$\geq$2500 gpm	1	A
Time Delays	RHR-TDR-K45, 1A&1B	$4.25 \leq T \leq 5.75$ min.	1	A
RHR Pump Start Time Delay	RHR-TDR-K75A, K70B RHR-TDR-K70A, K75B	$4.5 \leq T \leq 5.5$ sec. $\leq .5$ sec.	1 1	A A
RHR Heat Exchanger Bypass T.D.	RHR-TDR-K93, A & B	$1.8 \leq T \leq 2.2$ min.	1	B
RHR Crosstie Valve Position	RHR-LMS-2	Valve not closed	(3)	E
Bus 1A Low Volt. Aux. Relay	27 X 3/1A	Loss of Voltage	1	B
Bus 1B Low Volt. Aux. Relay	27 X 3/1B	Loss of Voltage	1	B
Bus 1F Low Volt. Aux. Relays	27 X 1/1F 27 X 2/1F	Loss of Voltage Loss of Voltage	1 1	B B
Bus 1G Low Volt. Aux. Relays	27 X 1/1G 27 X 2/1G	Loss of Voltage Loss of Voltage	1	B
Pump Discharge Line	CM-PS-266	$\geq$5 psig	(3)	D

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)	$\leq (0.66W + 42\%) \frac{A}{MTPF}$ (2)	2(1)
APRM Upscale (Startup)	$\leq$ 12%	2(1)
APRM Downscale (9)	$\geq$ 2.5%	2(1)
APRM Inoperative	(10b)	2(1)
RBM Upscale (Flow Bias)	$\leq (0.66W + 40\%)$ (2)	1
RBM Downscale (9)	$\geq$ 2.5%	1
RBM Inoperative	(10c)	1
IRM Upscale (8)	$\leq$ 108/125 of Full Scale	3(1)
IRM Downscale (3) (8)	$\geq$ 2.5%	3(1)
IRM Detector Not Full In (8)		3(1)
IRM Inoperative (8)	(10a)	3(1)
SRM Upscale (8)	$\leq 1 \times 10^5$ Counts/Second	1(1) (6)
SRM Detector Not Full In (4) (8)	($\geq$ 100 cps)	1(1) (6)
SRM Inoperative (8)	(10a)	1(1) (6)
Flow Bias Comparator	$\leq$ 10% Difference In Recirc. Flows	1
Flow Bias Upscale/Inop.	$\leq$ 110% Recirc. flow	1
SRM Downscale (8) (7)	$\geq$ 3 Counts/Second (0.3 counts/second prior to achieving burnup of 3500 MWD/T on first core)	1(1) (6)
RSCS Rod Group C Bypass	$\geq$ 20% Core Thermal Power	(11)

NOTES FOR TABLE 3.2.C

1. For the startup and run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM and IRM blocks need not be operable in "Run" mode, and the APRM (Flow biased) and RBM rod blocks need not be operable in "Startup" mode. If the third column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system with the inoperable channel shall be tripped. If the first column cannot be met for both trip systems, both trip systems shall be tripped.

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

2. W is the recirculation loop flow in percent of design. Trip level setting is in percent of rated power (2381 MWt). A = 2.61 for 7x7 fuel
   = 2.44 for 8x8 fuel.

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

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

5. One instrument channel; i.e., one APRM or IRM or RBM, per trip system may be bypassed except only one of four SRM may be bypassed.

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

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

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

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

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

    a. SRM and IRM

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

    b. APRM

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

    c. RBM

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

COOPER NUCLEAR STATION
TABLE 4.2.B (Page 2)
RHR SYSTEM TEST & CALIBRATION FREQUENCIES

Item	Item I.D. No.	Functional Test Freq.	Calibration Freq.	Instrument Check
Instrumentation				
1. Drywell High Pressure	PC-PS-101, A,B,C & D	Once/Month (1)	Once/3 Months	None
2. Reactor Vessel Shroud Level	NBI-LITS-73, A & B #1	Once/Month (1)	Once/3 Months	Once/Day
3. Reactor Low Pressure	RR-PS-128 A & B	Once/Month (1)	Once/3 Months	None
4. Reactor Low Pressure	NBI-PS-52 A & C NBI-PIS-52 B & D	Once/Month (1)	Once/3 Months	None
5. Drywell Press.-Containment Spray	PC-PS-119, A,B,C & D	Once/Month (1)	Once/3 Months	None
6. RHR Pump Discharge Press.	RHR-PS-120, A,B,C & D	Once/Month (1)	Once/3 Months	None
7. RHR Pump Discharge Press.	RHR-PS-105, A,B,C & D	Once/Month (1)	Once/3 Months	None
8. RHR Pump Low Flow Switch	RHR-dPIS-125 A & B	Once/Month (1)	Once/3 Months	None
9. RHR Pump Start Time Delay	RHR-TDR-K70, A & B	Once/Month (1)	Once/Oper. Cycle	None
10. RHR Pump Start Time Delay	RHR-TDR-K75, A & B	Once/Month (1)	Once/Oper. Cycle	None
11. RHR Heat Exchanger Bypass T.D.	RHR-TDR-K93, A & B	Once/Month (1)	Once/Oper. Cycle	None
12. RHR Cross Tie Valve Position	RHR-LMS-2	Once/Cycle (1)	N.A.	
13. Low Voltage Relays	27 X 3/1A	(7)		None
14. Low Voltage Relays	27 X 3/1B	(7)		None
15. Low Voltage Relays	27 X 2/1F, 27 X 2/1G	(7)		None
16. Low Voltage Relays	27 X 1/1F, 27 X (1)/1G	(7)		None
17. Pump Disch. Line Press. Low	CM-PS-266	Once/3 Months	Once/3 Months	None

COOPER NUCLEAR STATION
TABLE 4.2.B (Page 3)
RHR SYSTEM TEST & CALIBRATION FREQUENCIES

Item	Item I.D. No.	Functional Test Freq.	Calibration Freq.	Instrument Check
Logic (4)(6)				
1. Logic Bus Power Monitor		Once/6 Months	N.A.	N.A.
2. RHR Initiation		Once/6 Months	N.A.	N.A.
3. RHR Pump & Valve Control		Once/6 Months	N.A.	N.A.

Amendment No. 32

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.3.A (cont'd.)	4.3 (cont'd)

Specification 3.3.C.3 are inoperable, but if they can be inserted with control rod drive pressure they need not be disarmed electrically.

d. Control rods with a failed "Full-in" or "Full-out" position switch may be bypassed in the Rod Sequence Control System and considered operable if the actual rod position is known. These rods must be moved in sequence to their correct positions (full in on insertion or full out on withdrawal).

e. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.

f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reactor power operation, inoperable control rods shall be separated by at least two control rod cells.

If this Specification cannot be met the reactor shall not be started, or if at power, the reactor shall be brought to a shutdown condition within 24 hours.

All rods within a notch group containing an inoperable rod will be positioned within 1 (one) notch of the inoperable rod whenever the Rod Sequence Control System is required.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.3 (cont'd)	4.3 (cont'd)
B. Control Rods	B. Control Rods
1. Each control rod shall be coupled to its drive or completely inserted and the control rod directional control valves disarmed electrically. This requirement does not apply in the refuel condition when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.	1. The coupling integrity shall be verified for each withdrawn control rod as follows:
	a. When a rod is withdrawn the first time after each refueling outage or after maintenance, observe discernible response of the nuclear instrumentation and rod position indication. However, for initial rods when response is not discernible, subsequent exercising of these rods after the reactor is above 30% power shall be performed to verify instrumentation response.
	b. When the rod is fully withdrawn the first time after each refueling outage or after maintenance, observe that the drive does not go to the overtravel position.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	c. During each refueling outage observe that any drive which has been uncoupled from and subsequently recoupled to its control rod does not go to the overtravel position.
2. The control rod drive housing support system shall be in place during reactor power operation or when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.	2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.
3.a. Whenever the reactor is in the startup or run mode below 20% rated power, the rod sequence control system shall be operable. If the system is determined to be inoperable during power descent in accordance with checks in Specification 4.3.B.3, power may be maintained above 20% rated power until repairs are made.	3. Prior to the start of control rod withdrawal towards criticality and prior to attaining 20% rated power during rod insertion at shutdown, the capability of the Rod Worth Minimizer and Rod Sequence Control System to properly fulfill their functions shall be verified by the following checks:
b. During the shutdown procedure rod movement is restricted to the RSCS sequence following the testing performed between 35% and 20% power level and the automatic reinstatement of the RSCS restraints at the preset power level. Alignment of rod groups shall be accomplished prior to the reinstatement of the Rod Sequence Control System restraints.	a. The capability of the RSCS to properly fulfill its function shall be verified by the following tests: (1) <u>During Startup</u> Sequence portion - Select a sequence and attempt to withdraw a rod in the remaining sequences. Move one rod in a sequence and select the remaining sequences and attempt to move a rod in each. Repeat for all sequences. Group Notch Portion - Test the six comparator circuits. Go through each comparator inhibit, initiate test, verify error, and reset. After comparator checks initiate test and observe completion of cycle indicated by illumination of test complete light.
c. Whenever the reactor is in the startup or run modes below 20% rated power the Rod Worth Minimizer shall be operable or a second licensed operator or other qualified employee shall verify that the operator at the reactor console is following the control rod program.	

3.3 (cont'd)

d. Control rod withdrawal sequence shall be established such that the drop of any increment of any one in-sequence control rod would not result in a peak fuel enthalpy greater than 280 cal./gm.

4.3 (cont'd)

(2) <u>During Shutdown</u>

Sequence portion - Select a sequence and attempt to move a rod in the remaining sequences.

Group Notch Portion - Test the six comparator circuits. Go through each comparator inhibit, initiate test, verify error, and reset. After comparator checks initiate test and observe completion of cycle indicated by illumination of test complete light.

b. The capability of the Rod Worth Minimizer (RWM) shall be verified by the following checks:

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
e. If Specifications 3.3.B.3a through d cannot be met, the reactor shall not be started, or if the reactor is in the run or startup modes at less than 20% rated power, it shall be brought to a shutdown condition immediately.	1) The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified.
	2) The RWM computer on line diagnostic test shall be successfully performed.
f. The sequence restraints imposed on the control rods may be removed by the use of the individual rod position bypass switches for scram testing only those rods which are fully withdrawn in the 100% to 50% rod density range.	3) Proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group shall be verified.
	4) The rod block function of the RWM shall be verified by withdrawing the first rod as an out-of-sequence control rod no more than to the block point.
	c. When required, the presence of a second licensed operator or other qualified employee to verify the following of the correct rod program shall be verified.
4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.	4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.
5. During operation with limiting control rod patterns, as determined by the designated qualified personnel, either:	5. When a limiting control rod pattern exists an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s).
a. Both RBM channels shall be operable: or	
b. Control rod withdrawal shall be blocked: or	
c. The operating power level shall be limited so that the MCPR will remain above 1.06 assuming a single error that results in complete withdrawal of any single operable control rod.	

LIMITING CONDITION FOR OPERATION

3.3 (cont'd)

C. Scram Insertion Times

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.0
90	3.50

2. The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two-by-two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.71

SURVEILLANCE REQUIREMENT

4.3 (cont'd)

C. Scram Insertion Times

1. After each refueling outage all operable rods shall be scram time tested from the fully withdrawn position with the nuclear system pressure above 800 psig (with saturation temperature) and the requirements of Specification 3.3.B.3.a met. This testing shall be completed prior to exceeding 40% power. Below 20% power, only rods in those sequences (A_{12} and A_{34} or B_{12} and B_{34}) which were fully withdrawn in the region from 100% rod density to 50% rod density shall be scram time tested. During all scram time testing below 20% power, the Rod Worth Minimizer shall be operable or a second licensed operator or other qualified employee shall verify that the operator at the reactor console is following the control rod program.

2. At 16-week intervals, 10% of the operable control rod drives shall be scram timed above 800 psig. Whenever such scram time measurements are made. An evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.3.C (cont'd)	**4.3.C**
3. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.	
D. Reactivity Anomalies	**D. Reactivity Anomalies**
At a specific steady state base condition of the reactor actual control rod inventory will be periodically compared to a normalized computer prediction of the inventory. If the difference between observed and predicted rod inventory reaches the equivalent of 1% Δk reactivity, the reactor will be shut down until the cause has been determined and corrective actions have been taken as appropriate.	**During the startup test program and startup following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.**
E. Recirculation Pumps	
A recirculation pump shall not be started while the reactor is in natural circulation flow and reactor power is greater than 1% of rated thermal power.	
F. **If Specifications 3.3.A through D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Shutdown condition within 24 hours.**	

cannot be moved with drive pressure. If the rod is fully inserted and then disarmed electrically*, it is in a safe position of maximum contribution to shutdown reactivity. If it is disarmed electrically in a non-fully inserted position, that position shall be consistent with the shutdown reactivity limitation stated in Specification 3.3.A.1. This assures that the core can be shutdown at all times with the remaining control rods assuming the strongest operable control rod does not insert. An allowable pattern for control rods valved out of service, which shall meet this Specification, will be determined and made available to the operator.

In order to perform shutdown margin and control rod drive scram time tests subsequent to any fuel loading operation as required by the Technical Specifications, the relaxation of the following Rod Sequence Control System restraints is required: (a) The sequence restraints imposed on the control rods may be removed by the use of the individual rod position bypass switches for scram testing only those rods which are fully withdrawn in the 100% to 50% rod density range. (b) Verify that subsequent to the use of the rod position bypass switches rod movement in the 50% rod density to preset power level range is restricted to the single notch mode.

If damage within the control rod drive mechanism and in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occured in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing and requiring increased surveillance after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings.

B. Control Rod

1. Control rod drop accidents as discussed in the FSAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement could indicate an uncoupled condition. Rod position indication is required for proper function of the rod sequence control system and the rod worth minimizer (RWM).

*To disarm the drive electrically, four Amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the rod incapable of withdrawal. This procedure is equivalent to valving out the drive and is preferred because, in this condition, drive water cools and minimizes crud accumulation in the drive. Electrical disarming does not eliminate position indication.

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in subsection III.8.2 of the FSAR and the safety evaluation is given in subsection VIII.8.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated, since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. The Rod Worth Minimizer (RWM) and the Rod Sequence Control System (RSCS) restrict withdrawals and insertions of control rods to prespecified sequences. These sequences are established such that the drop of any in-sequence control rod from the fully inserted position to the position of the control rod drive would not cause the reactor to sustain a power excursion resulting in a peak fuel enthalpy in excess of 280 cal./gm. An enthalpy of 280 cal./gm. is well below the level at which rapid fuel dispersal could occur (i.e., 425 cal./gm.). Primary system damage in this accident is not possible unless a significant amount of fuel is rapidly dispersed. Ref. Subsections III.6.6, VIII 7.4.5, and XIV.6.2 of the FSAR and NEDO-10527 including Supplements 1 and 2 to NEDO-10527.

   In performing the function described above, the RWM and RSCS are not required to impose any restrictions at core power levels in excess of 20% of rated. Material in the cited references shows that it is impossible to reach 280 calories per gram in the event of a control rod drop occurring at power greater than 20%, regardless of the rod pattern. This is true for all normal and abnormal patterns including those which maximize the individual control rod worth.

   At power levels below 20% of rated, abnormal control rod patterns could produce rod worths high enough to be of concern relative to the 280 calories per gram rod drop limit. In this range the RWM and the RSCS constrain the control rod sequences and patterns to those which involve only acceptable rod worths.

   The Rod Worth Minimizer and the Rod Sequence Control System provide automatic supervision to assure that out of sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. They serve as a backup to procedural control of control rod sequences, which limit the maximum reactivity worth of control rods. In the event that the Rod Worth Minimizer is out of service, when required, a second licensed operator or other qualified technical plant employee whose qualifications have been reviewed by the NRC can manually fulfill the control rod pattern conformance functions of this system. In this case, the RSCS is backed up by independent procedural control to assure conformance.

The functions of the RWM and RSCS make it unnecessary to specify a license limit on rod worth to preclude unacceptable consequences in the event of a control rod drop. At low powers, below 20%, these devices force adherence to acceptable rod patterns. Above 20% of rated power, no constraint on rod pattern is required to assure that rod drop accident consequences are acceptable. Control rod pattern constraints above 20% of rated power are imposed by power distribution requirements as defined in Section 3.3.B.5 of these Technical Specifications. Power level for automatic cutout of the RSCS function is sensed by first stage turbine pressure. Because the instrument has an instrument error of $\pm$ 2% of full power, the nominal instrument setting is 22% of rated power. Power level for automatic cutout of the RWM function is sensed by feedwater and steam flow and is set nominally at 30% of rated power to be consistent with the RSCS setting.

Functional testing of the RWM prior to the start of control rod withdrawal at startup, and prior to attaining 20% rated thermal power during rod insertion while shutting down, will ensure reliable operation and minimize the probability of the rod drop accident.

The RSCS can be functionally tested prior to control rod withdrawal for reactor startup. By selecting, for example, A_{12} and attempting to withdraw, by one notch, a rod or all rods in each other group, it can be determined that the A_{12} group is exclusive. By bypassing to full-out all A_{12} rods, selecting A_{34} and attempting to withdraw, by one notch, a rod or all rods in group B, the A_{34} group is determined exclusive. The same procedure can be repeated for the B groups. After 50% of the control rods have been withdrawn (e.g., groups A_{12} and A_{34}), it is demonstrated that the Group Notch mode for the control drives is enforced. This demonstration is made by performing the hardware functional test sequence. The Group Notch restraints are automatically removed above 20% power.

During reactor shutdown, similar surveillance checks shall be made with regard to rod group availability as soon as automatic initiation of the RSCS occurs and subsequently at appropriate stages of the control rod insertion.

4. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron

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.

3.4 BASES

STANDBY LIQUID CONTROL SYSTEM

A. The conditions under which the Standby Liquid Control System must provide shutdown capability are identified via the Plant Nuclear Safety Operational Analysis (Appendix G). If no more than one operable control rod is withdrawn, the basic shutdown reactivity requirement for the core is satisfied and the Standby Liquid Control system is not required. Thus, the basic reactivity requirement for the core is the primary determinant of when the liquid control system is required.

The purpose of the liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown condition assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron that produces a concentration of 600 ppm of boron in the reactor core in less than 125 minutes. The 600 ppm concentration in the reactor core is required to bring the reactor from full power to a 3.0 percent Δk subcritical condition, considering the hot to cold reactivity difference, xenon poisoning, etc. The time requirement for inserting the boron solution was selected to override the rate of reactivity insertion caused by cooldown of the reactor following the xenon poison peak.

The minimum limitation on the relief valve setting is intended to prevent the recycling of liquid control solution via the lifting of a relief valve at too low a pressure. The upper limit on the relief valve settings provides system protection from overpressure.

B. Only one of the two standby liquid control pumping loops is needed for operating the system. One inoperable pumping circuit does not immediately threaten shutdown capability, and reactor operation can continue while the circuit is being repaired. Assurance that the remaining system will perform its intended function and that the long term average availability of the system is not reduced is obtained for a one out of two system by an allowable equipment out of service time of one third of the normal surveillance frequency. This method determines an equipment out of service time of ten days. Additional conservatism is introduced by reducing the allowable out of service time to seven days, and by increased testing of the operable redundant component.

C. Level indication and alarm indicate whether the solution volume has changed, which might indicate a possible solution concentration change. The test interval has been established in consideration of these factors. Temperature and liquid level alarms for the system are annunciated in the control room.

The solution is kept at least 10°F above the saturation temperature to guard against boron precipitation. The margin is included in Figure 3.4.2.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.5.A (cont'd.)	4.5.A (cont'd.)
2. From and after the date that one of the core spray subsystems is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding seven days provided that during such seven days all active components of the other core spray subsystem and active components of the LPCI subsystem and the diesel generators are operable.	2. When it is determined that one core spray subsystem is inoperable, the operable core spray subsystem, the LPCI subsystem and the diesel generators shall be demonstrated to be operable immediately. The operable core spray subsystem shall be demonstrated to be operable daily thereafter.
3. Both LPCI Subsystems shall be operable: (1) prior to reactor startup from a Cold Condition, or (2) when there is irradiated fuel in the vessel and when the reactor vessel pressure is greater than atmospheric pressure, except as specified in 3.5.A.4 and 3.5.A.5 below.	3. LPCI Subsystem Testing shall be as follows: Item — Frequency a. Simulated Automatic Actuation Test — Once/Operating Cycle b. Pump Operability — Once/month c. Motor Operated valve operability — Once/month d. Pump Flow Rate — Once/3 months Each RHR pump shall deliver at least 8400 gpm but no more than 8800 gmp against a reactor vessel pressure at atmospheric conditions and a total system flow of at least 16,000 gpm if two RHR pumps are injecting into the same recirculation loop. e. Recirculation Pump discharge valves shall be tested each refueling outage to verify full open to full closed in $20 \leq t \leq 26$ seconds.
4. From and after the date that one of the RHR (LPCI) pumps is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding thirty days provided that during such thirty days the remaining active components of the LPCI Subsystem and all active components of both core spray subsystems and the diesel generators are operable.	4. When it is determined that one of the RHR (LPCI) pumps is inoperable at a time when it is required to be operable the remaining active components of the LPCI Subsystems, the containment cooling subsystem, both core spray systems and the diesel generators shall be demonstrated to be operable immediately and the operable LPCI pumps daily thereafter.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.E. Jet Pumps	4.6.E. Jet Pumps
1. Whenever the reactor is in the start-up or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instruments failures occur and cannot be corrected within 24 hours, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.	1. Whenever there is recirculation flow with the reactor in the startup or run modes, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously: a. The recirculation pump flow differs by more than 15% from the established speed flow characteristics. b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%. c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.
F. Jet Pump Flow Mismatch	F. Jet Pump Flow Mismatch
1. Deleted.	1. Deleted.
2. Following one-pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is equal to or less than 50% of its rated speed.	
3. The reactor shall not be operated for a period in excess of 24 hours with one recirculation loop out of service.	
G. Structural Integrity	G. Structural Integrity
The structural integrity of the primary system boundary shall be maintained at the level required to assure safe operation throughout the life of the station. The reactor shall be maintained in a Cold Shutdown condition until each indication of a defect has been investigated and evaluated.	The nondestructive inspections listed in Table 4.6.1 shall be performed as specified. The results obtained from compliance with this specification will be evaluated after 5 years and the conclusions of this evaluation will be reviewed with the NRC.

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

Specification 4.6.E.b will not be met until start of Commercial Operation since the required data is obtained during the Startup Test Program.

F. Jet Pump Flow Mismatch

Requiring the discharge valve of the lower speed loop to remain closed until the speed of faster pump is equal to or less than 50% of its rated speed provides assurance when going from one to two operation that excessive vibration of the jet pump risers will not occur.

A loss-of-coolant accident analysis occurring during operation with one recirculation loop has not been performed. Therefore, operation with a single loop is prohibited except for a limited interval of 24 hours.

G. Structural Integrity

A preservice inspection of accessible components listed in Table 4.6.1 will be conducted before initial fuel loading to assure the system is free of gross defects and as a reference base for later inspections. Construction orientated nondestructive testing is being conducted as systems are fabricated to assure applicable code requirements are met. Prior to operation, the primary system boundary will be free of gross defects. In addition, the facility has been designed such that gross defects should not occur throughout the life of the station. The inspection program given in Table 4.6.1 is based on the requirements of Section IS-242: Table IS-251, Components, Parts and Methods of Examination, and Table IS-251, Examination Categories, all of Section XI of the 1970 ASME Boiler and Pressure Vessel Code, except where accessibility for inspection was not provided. The initial program was revised to update to the summer 1972 Addendum Table IS-261. Modifications were made to vessel nozzle insulation and nozzle blockout removable shielding designs with the intent to make the inspection areas more accessible by reducing the personnel radiation exposure required for inspection utilizing available equipment.

The inspection program and the modifications described above were developed

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.9.B (cont'd)	4.9.B
6. From and after the date that one of the 125 or 250 volt battery systems is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding ten days within electrical safety considerations, provided repair work is initiated in the most expeditious manner to return the failed component to an operable state, and Specifications 3.5.A.5 and 3.5.F are satisfied. The NRC shall be notified within 24 hours of the situation, the precautions to be taken during this period and the plans to return the failed components to an operable state.	

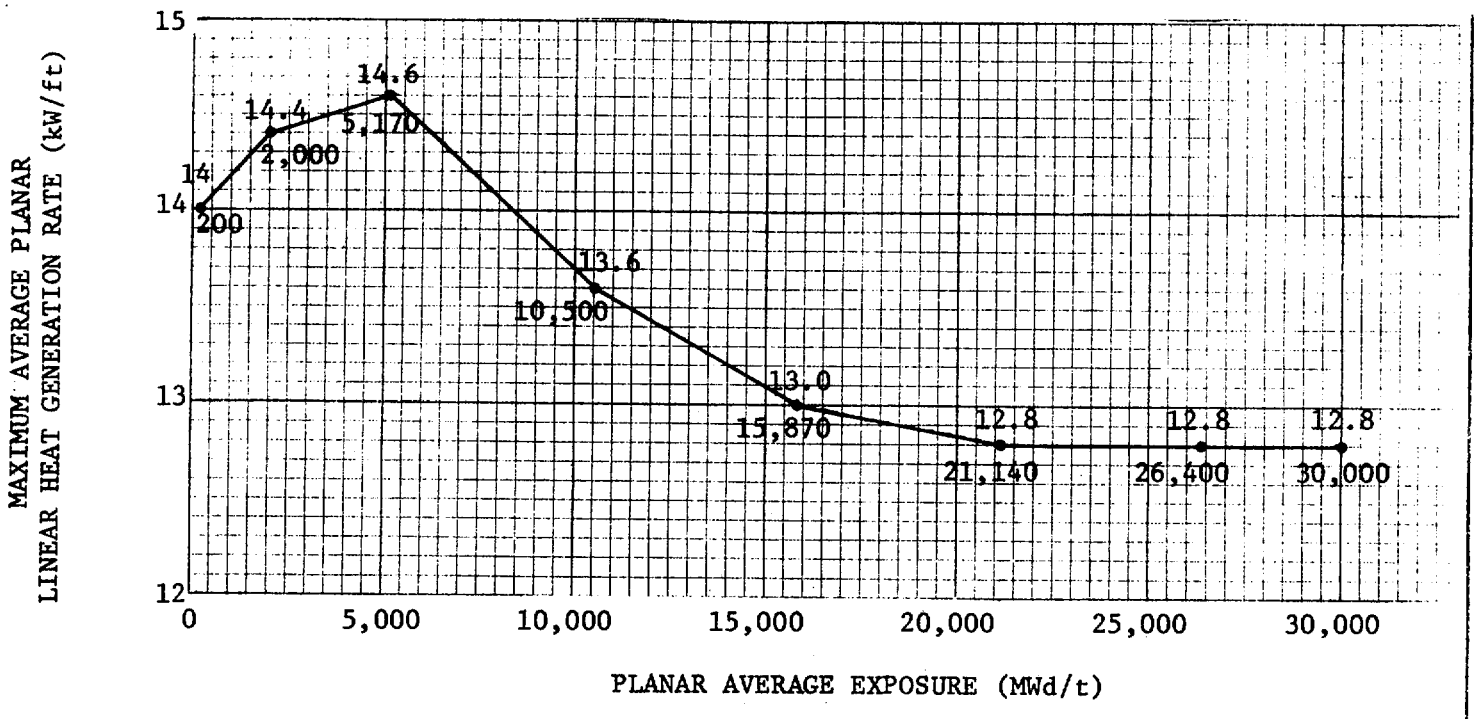

Figure 3.11-1.1 Maximum Average Planar Linear Heat Generation Rate versus Exposure with LPCI Modification and Bypass Flow Holes Plugged, Initial Core Fuel Types 1 and 3

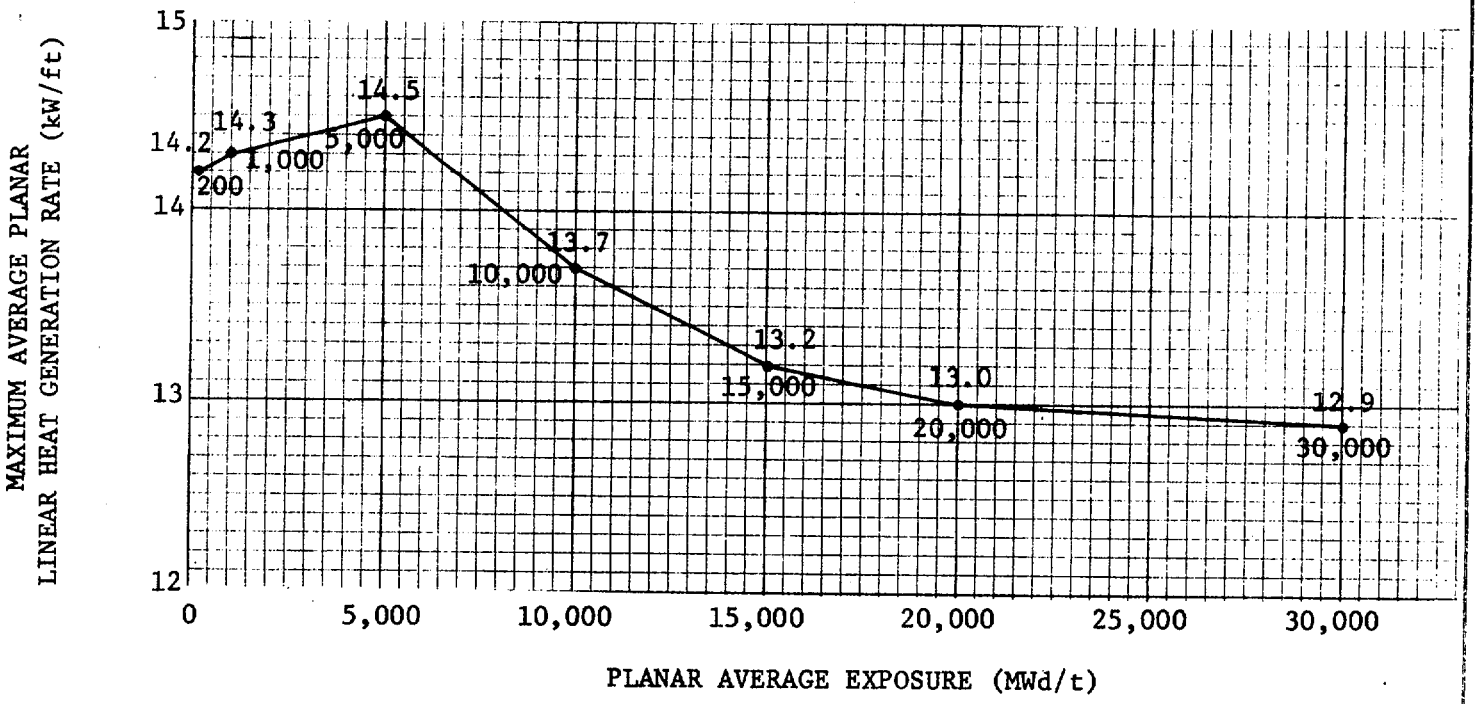

Figure 3.11-1.2 Maximum Average Planar Linear Heat Generation Rate versus Exposure with LPCI Modification and Bypass Holes Plugged, Initial Core Fuel Type 2

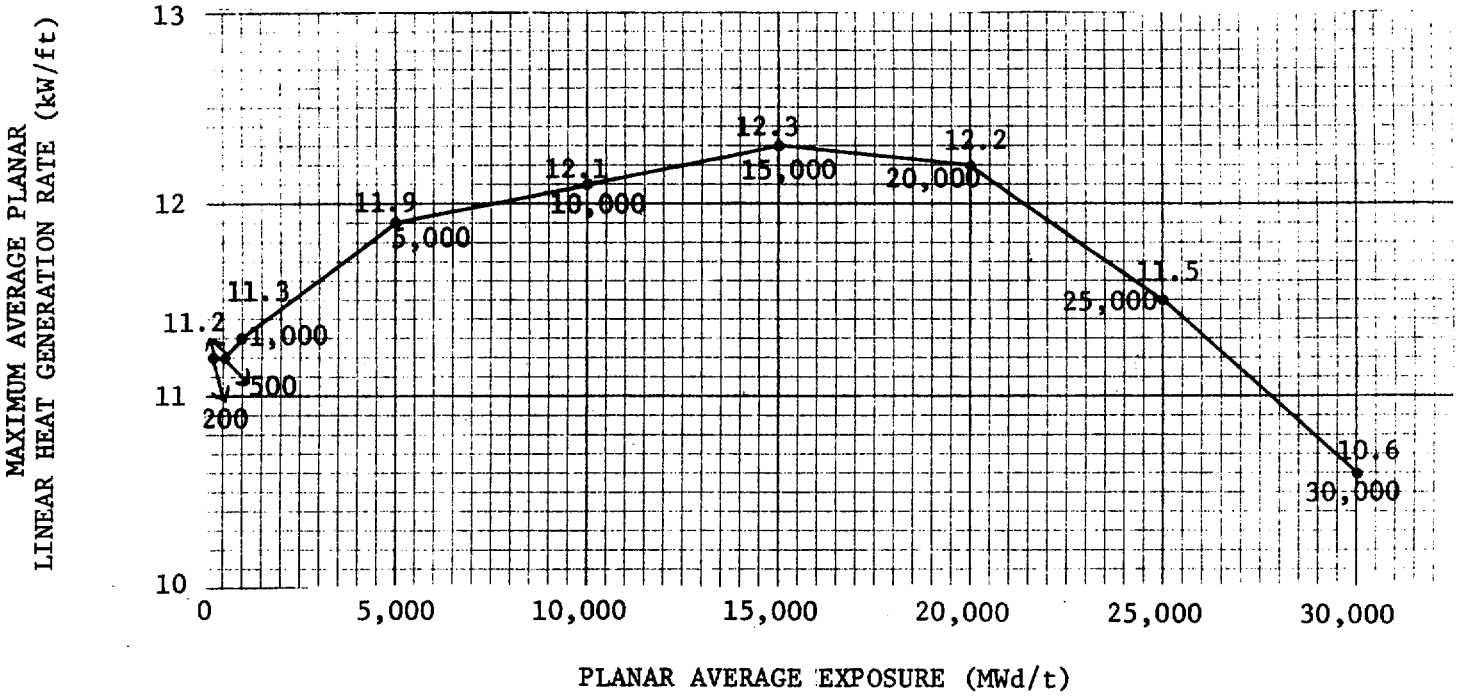

Figure 3.11-1.3. Maximum Average Planar Linear Heat Generation Rate versus Exposure with LPCI Modification and Bypass Flow Holes Plugged, 8D250 Fuel

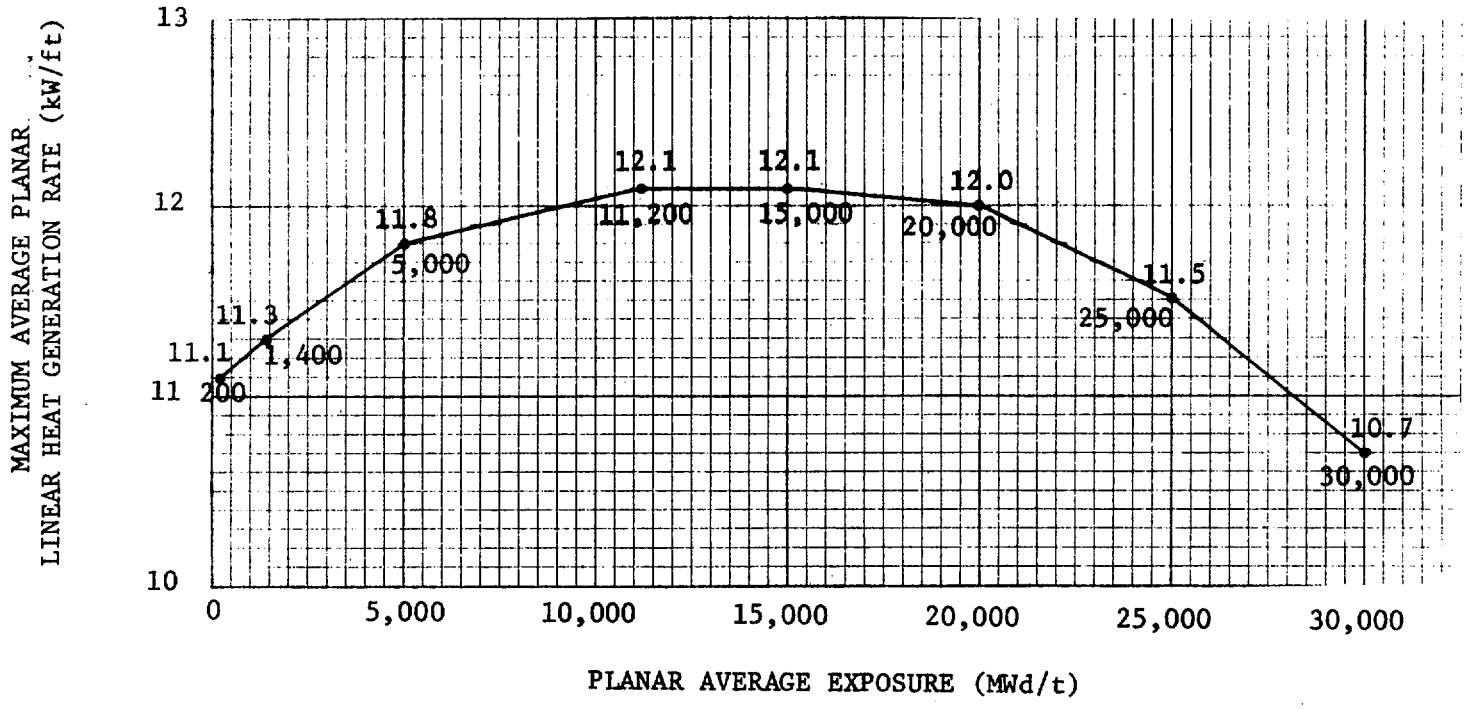

Figure 3.11-1.4. Maximum Average Planar Linear Heat Generation Rate versus Exposure with LPCI Modification and Bypass Flow Holes Plugged, 8D274L Fuel

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 be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR 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.

C. Minimum Critical Power Ratio (MCPR)

During steady state power operation MCPR shall be $\geq$ 1.25 for 7x7 fuel and $\geq$ 1.28 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 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 >1.25 for 7x7 fuel and > 1.28 for 8x8 fuel 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.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.11.D. Thermal-hydraulic Stability Core thermal power shall not exceed 25% of rated thermal power without forced recirculation.	

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 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 ≥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 most 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 turbine trip with failure of the turbine bypass. 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 this evaluation is the reduction in MCPR caused by the transient.

D. MCPR Limits for Core Flows Other than Rated

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

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR of 1.25 for 7x7 and 1.28 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 1.25 for 7x7, 1.28 for 8x8 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. "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. 1 License Amendment Submittal", July 1976 (NEDO-21292).

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 recriculation 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. Thermal-hydraulic Stability

The calculations, regarding reactor core stability, presented in General Electric BWR Reload 1 License Amendment Submittal for Cooper Nuclear Station, July 1976 (NEDO-21292), 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) of 0.36 which is significantly less than the ultimate stability limit of 1.0.

5.0 MAJOR DESIGN FEATURES

5.1 Site Features

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

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

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

5.2 Reactor

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

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

5.3 Reactor Vessel

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

5.4 Containment

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

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 32 TO LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

INTRODUCTION

By letters dated April 7 and August 9, 1976, supplemented by letters dated July 20, September 20, October 8, 9, 15, 20, 26, and November 4 and 10, 1976, Nebraska Public Power District (NPPD - the licensee) requested amendments (hereinafter referred to as amendment) to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS). The amendment would modify the Technical Specifications for CNS to permit operation with (1) 120 General Electric (GE) 8 x 8 reload fuel bundles, (2) bypass flow holes drilled in the lower tie plates of all fuel assemblies in the core (with the exception of 11 Type 1 assemblies) as part of facility modifications to eliminate in-core vibration of instrument and source tubes, (3) a modified Rod Sequence Control System (RSCS), and (4) modifications to improve the performance of the Low Pressure Coolant Injection (LPCI) System of the Emergency Core Cooling System (ECCS).

DISCUSSION

NPPD has proposed to reload the CNS reactor with 72 GE 8 x 8 fuel assemblies with an enrichment of 2.50 wt% U-235 and 48 GE 8 x 8 reload assemblies with an enrichment of 2.74 wt% U-235.

The neutronic, thermal-hydraulic, and mechanical design of 8 x 8 fuel assemblies during normal operation, operational transients and postulated accidents was evaluated, and found acceptable, by the NRC staff in a report entitled "Technical Report on the General Electric Company 8 x 8 Fuel Assembly" dated February 5, 1974. The use of 8 x 8 fuel assemblies for reloads was also reviewed, and found acceptable, by the Advisory Committee on Reactor Safeguards and discussed in its report dated February 12, 1974. The information presented in the licensee's application to reload using 8 x 8 fuel assemblies closely follows the guidelines of NEDO-20360[2]. Although this report is still undergoing review by the NRC staff, we have determined that the applicable sections of the report provide an acceptable analytical basis for the evaluation of the CNS reload.

The lower tie plates of both the reload assemblies and the remaining assemblies in the core, with the exception of type 1 assemblies, will be modified to include alternate reactor coolant bypass flow holes. NRC approval to modify the reload fuel assemblies was granted in Amendment No. 30 (dated June 11, 1976) to the CNS license; approval to modify the remaining assemblies, in Amendment No. 28 (dated September 27, 1976). Installation of alternate bypass flow holes is part of the facility modifications for eliminating significant in-core vibration of instrument and source tubes.

As part of the CNS Reload 1 submittal of August 9, 1976, NPPD proposed a modification to the RSCS to replace group "C" mode with the group notch mode that would be completed during the current refueling outage.

By letter dated April 7, 1976, supplemented by letters dated July 20, August 9, October 9 and 26, and November 4 and 10, 1976, NPPD requested authorization to modify the LPCI system at CNS to increase the reliability and availability of the LPCI system in the event of a postulated loss-of-coolant accident. Amendment No. 31 (dated September 28, 1976) to the CNS license authorized the installation of the LPCI system modifications. However, as noted in the Safety Evaluation for Amendment No. 31, parts of the LPCI modification would require further review by the NRC staff prior to operation of CNS. This evaluation addresses the remaining modifications which were not completely reviewed in Amendment No. 31.

By letter dated May 20, 1976, we requested NPPD to analyze and determine if long term heat removal capabilities for CNS could be impaired because of LPCI pump runout following a postulated loss-of-coolant accident. NPPD responded to our request by letters dated June 21 and 30, and October 1, 1976. This evaluation addresses the acceptability of NPPD's response.

EVALUATION

Nuclear Characteristics

Reference 1 indicates that 120 GE 8 x 8 reload fuel bundles will be loaded throughout the core. Seventy-two reload fuel assemblies will have an average enrichment of 2.50 wt% U-235 (8D250) and the remaining forty-eight of the reload fuel assemblies will have an average fuel enrichment of 2.74 wt% U-235 with a low gadolinia content (8D274L). The 8D250 fuel assemblies have a total active fuel length of 144 inches compared with 146 inches for the other assemblies. The 8D274L assemblies will include the improved water rod design described in reference 2. The core contains a total of 548 fuel assemblies. Thus, about 22% of the initial 7 x 7 fuel assemblies will be replaced by the 8 x 8 reload fuel assemblies.

The proposed loading pattern consists of 8 x 8 reload assemblies inserted in a symmetrical pattern throughout the core among the 7 x 7 initial cycle fuel assemblies. The lower enrichment reload assemblies will be loaded in the interior of the core while the low gadolinia high enrichment reload assemblies will be loaded in the periphery of the core. The data in reference 1 indicate that the nuclear characteristics of the reload 1 core are similar to the previous core. Thus, the total control system worth and the temperature and void dependent behavior of the reconstituted core will not differ significantly from those values previously reviewed and approved for cycle 1 of the CNS reactor. The shutdown margin of the reconstituted core meets the Technical Specification requirement that the core be at least 0.38% subcritical in the most reactive operating state with the most reactive control rod fully withdrawn and with all the other rods fully inserted. For cycle 2 a minimum shutdown margin of 1.14%ΔK was calculated.

Information presented in reference 1 indicates that a boron concentration of 600 ppm in the moderator will bring the reactor subcritical by 0.03 ΔK at 20°C for a xenon free condition. Therefore, the shutdown margin requirement of the General Design Criteria is met by the Standby Liquid Control System.

The full power scram reactivity curve for reload 1 is shown in Figure 6-6 of reference 1. The scram curves used in the transient analyses include a design conservatism of 0.8.

Thus, based on our review of the information presented in the NPPD application[1] and supplemented by sections of the generic 8 x 8 reload report[2], the nuclear characteristics and performance of the reconstituted core for cycle 2 are acceptable.

Mechanical Design

The two types of reload 1 fuel have the same mechanical configuration and fuel assembly enrichments as the 8D274L and 8D250 fuel assemblies described in the 8 x 8 generic reload report, except that holes are drilled in the reload assembly lower tie plates. Two 9/32 inch holes are drilled in the lower tie plate of all fuel assemblies, with the exception of 11 of the type 1 7 x 7 fuel assemblies, to provide bypass coolant flow. The 8D274L fuel assemblies of Reload 1 incorporate the improved water rod design analyzed in reference 2. Justification for operation with bypass flow holes has been given in reference 6. This

report has been reviewed and accepted by the staff[10]. Based on information provided by NPPD, we have determined that the decreased active fuel length in the 8D250 fuel assemblies will have negligible effect on the mechanical, nuclear and thermal-hydraulic characteristics of the reload 1 core. On the basis of our review of the generic 8 x 8 reload report and the reload submittal, the staff concludes that the design of the fuel proposed for cycle 2 operation at Cooper Nuclear Station is acceptable.

Thermal-Hydraulic

The GE 8 x 8 fuel reload topical[2] and the GE/BWR Thermal Analysis Basis Licensing Topical Report - GETAB[7] are referenced to provide a basis for the core thermal hydraulics analysis for cycle 2. Application of the GETAB establishes:

(1) the fuel damage safety limit,

(2) the limiting conditions of operation 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 2 thermal margins based on the GETAB report and plant specific input information provided by the licensee. The staff evaluation of these margins follows.

Fuel Cladding Integrity Safety Limit MCPR

The fuel cladding integrity safety limit MCPR is 1.06. It is based on the GETAB statistical analysis which assures that 99.9% of the fuel rods in the core are not expected to experience boiling transition "during normal operation or transients that are anticipated to occur with moderate frequency". The uncertainties in the core and system operating parameters and the GEXL correlation (Table 4-1 of reference 1), combined with the relative bundle power distribution in the core, form the basis for the GETAB statistical determination of the safety limit MCPR[2]. The tabulated list of uncertainties for CNS cycle 2 are the same as those used in the initial cycle except for the bypass void effect on TIP uncertainty. The analysis includes revised uncertainty in the standard deviation for the TIP readings (3.58-4.25%) due to the decreased bypass void content resulting from drilling of the bypass holes in most of the fuel assemblies.

Part of the review of the proposed cycle 2 operation dealt with the question of the acceptability of the flow distribution and thermal margin from the loading of reload and first cycle fuel with drilled lower tie plates. The concern was that the inlet bypass flow might be nonuniform and that this, combined with nonuniform heat generation in the fuel bundles, might cause more voiding in the hotter bypass regions than had been taken into account. The bypass voiding is applied as an uncertainty in the determination of the safety limit Minimum Critical Power Ratio (MCPR).

For the Safety Evaluation Report on Modification to Eliminate Significant Incore Vibration[11], the staff reviewed and found acceptable operation with both the completely drilled and completely plugged cores. The staff's position on the operation of partially drilled cores was as follows:

> "For those reactors in which the 1-inch bypass flow holes are plugged but not all fuel bundles are drilled, we conclude that that outreactor flow test sufficiently demonstrated that the modification will reduce significantly in-core tube vibration and hence channel box damage. However, the allowable power level after such modification must be reviewed individually for each reactor considering normal operation, anticipated transients and accidents (NEDE 21156)."[6]

The thermal-hydraulic analyses were performed as described in reference 7. This conservatively predicts the thermal margins for individual fuel bundles based on the predicted bypass flow rates. Because of the small number of undrilled fuel assemblies, the flow maldistribution problem is not significant. The amount of bypass voiding is calculated using a core hydraulics model which calculates the flow distribution in both the bundles and the bypass region assuming a given core pressure drop and total flow rate. The bypass voiding is conservatively calculated by assuming no mixing with bypass flow from cooler regions.

The staff has determined that the thermal hydraulics performance with bypass voiding as a result of partial loading with drilled fuel has been conservatively considered for cycle 2 operation for CNS. The calculational methods[2] and input data[1] have been conservatively represented and used in this analysis. Therefore, we find the operating limit MCPR to be acceptable for CNS cycle 2 operation.

The bypass voiding will affect the power distribution in the bundle and this will have an effect on the R factor used in the GEXL correlation as well as the uncertainty in the R factor which is used in calculating the Safety Limit MCPR. In generic discussions with the NRC staff, General Electric has stated that these effects on the R factor are considered and are within the conservatism used in obtaining the uncertainty in the R factor. We consider this to be acceptable. The statistical thermal hydraulic analysis of the reactor core was performed with uncertainty input described in references 2 and 9. The results of the analyses show that at a MCPR of 1.06 or greater at least 99.9% of the fuel rods in the core will avoid boiling transition. We conclude that the proposed fuel integrity safety limit MCPR of 1.06 is acceptable for CNS cycle 2.

Operating Limit MCPR

Various transient events will reduce the MCPR below the operating MCPR. To assure that the fuel cladding integrity safety limit (MCPR of 1.06) is not violated during anticipated abnormal operational transients, the most limiting transients have been analyzed to determine which one results in the largest reduction in critical power ratio (ΔCPR). The licensee has submitted the results of analyses of those transients which produce a significant decrease in MCPR[1].

The most limiting transient was the rod withdrawal error. This transient results in ΔCPR's of 0.16 and 0.19 for 7 x 7 and 8 x 8 fuel assemblies, respectively. Addition of these ΔCPR's to the safety limit MCPR (1.06) gives the minimum operating limit MCPR for each fuel type required to avoid violation of the safety limit, should this limiting transient occur. However, these limits would be further increased by 0.03 to compensate for the reduction in CPR that could result from a fuel loading error, as discussed later. Thus, based on the fuel cladding integrity safety limit, the results of the abnormal operational transient analysis, and the fuel loading error analysis, the operating limit MCPR is 1.25 and 1.28 for 7 x 7 and 8 x 8 fuel assemblies, respectively.

Anticipated Transients

The licensee, in reference 1, stated that "all transients which are the basis of the existing license were reviewed, and those transients which have been limiting in the past with respect to safety margins and are significantly sensitive to the core transient parameter deviations were reanalyzed." These transients were analyzed on the basis that they were limiting for the previous cycle and that there was no significant change to the system parameters for this cycle which would cause another transient to be more restrictive. The methods used for these analyses are described in references 8 and 9. The analyses were performed at the end of cycle burnup and two intermediate burnup conditions. The transients reanalyzed for cycle 2 were Turbine Trip Without Bypass, Loss of Feedwater Heater, Rod Withdrawal Error, and Over-pressurization Protection from Main Steam Isolation Valve Closure. The bases and methods for the transient analysis have been reviewed and found acceptable. The results of these reanalyses are given in table 6-2 of reference 1. The most limiting transients are the overpressurization and rod withdrawal error transients for both the cycle 1 and cycle 2 cores.

Overpressure Protection Analysis

In reference 1, NPPD presented the results of an overpressure analysis to demonstrate that an adequate margin exists below the ASME Code allowable vessel pressure of 110% of the 1250 psig vessel design pressure. The transient analyzed was the closure of all main steam isolation valves with high neutron flux scram. The analysis was performed for 104% initial power with the end of cycle scram reactivity insertion rate curve, scram initiated by high neutron flux, void reactivity applicable to this reload, no credit for the relief function of the safety/relief valves, and with all safety valves operable. The results of this analysis indicate that the peak pressure at the vessel bottom would be 1279 psig. Furthermore, generic analysis applied to CNS showed that for the aforementioned overpressure event, the failure of one safety valve would cause the maximum vessel pressure to increase by 20 psig. Hence, the maximum peak reactor vessel pressure for MSIV closure with flux scram, no relief function of the safety relief valves and one failed safety valve is calculated to be 1299 psig; this results in about a 76 psi margin below the code allowable pressure which is an acceptable margin for overpressure protection. The NRC staff has concluded that the overpressure protection analysis methods and results for CNS cycle 2 are acceptable.

Rod Withdrawal Error Transient

The licensee has analyzed the Rod Withdrawal Error with the assumptions given in reference 1. The results show that a rod block monitor (RBM) set point of 106% will stop rod withdrawal when the critical power ratio is at the 1.06 MCPR safety limit. Based on analysis of the worst case condition for CNS, the proposed rod block monitor set point of <0.66W + 40% (where W is the fraction of full coolant flow) is acceptable.

Loss-of-Coolant Accident

NPPD has submitted in reference 4 a reanalysis of the loss-of-coolant accident for cycle 2 operation with modified Low Pressure Coolant Injection. This analysis was performed with the assumption that bypass flow holes in the core support plate are completely plugged, and in accordance with the Appendix K requirements of 10 CFR Part 50. In the analysis of the loss-of-coolant accident, a range of break sizes were studied with a range of single failures being considered for each break size. The input parameters and models for the loss-of-coolant accident are presented in reference 4. The spectrum of pipe breaks yields calculated peak cladding temperatures less than or equal to 2200°F, calculated maximum local metal water reactions less than or equal to 17% and calculated core-wide metal-water reactions less than or equal to 1% at the Maximum Average Planar Linear Heat Generation Rates employed in the analysis. Compliance with the 10 CFR 50.46 criteria for coolable geometry and long-term cooling has been demonstrated in reference 2. Therefore, the staff has determined that the CNS loss-of-coolant accident analysis meets the requirements of 10 CFR Part 50, §50.46.

Main Steam Line Break, Refueling Accident

The analyses of these accidents were presented by the licensee as being covered by the generic analyses given in reference 2. Based on the previous review of the referenced material, the staff concludes that the results provided by the generic analyses are acceptable.

Control Rod Drop Accident

A reanalysis of the Control Rod Drop Accident was performed, because the hot startup accident reactivity shape functions were not bound by the reference analysis. This analysis used the technical bases as described in reference 1 and all plant specific parameters of CNS. The analysis demonstrated that the design limit on fuel rod peak enthalpy (280 cal/gm) will not be exceeded for a control rod drop accident from any in-sequence control rod motion during startup.

Based on the submitted analysis and approval of the RSCS Group Notch mode, discussed below, the staff finds the consequences of the control rod drop accident acceptable.

Fuel Loading Error Accident

Fuel loading errors are discussed in Reference 1 for a fuel bundle inserted in an improper location or rotated 180 degrees in a location near the center of the core and the error not discovered in the subsequent core verification. The information in Reference 1 indicates that this fuel loading error results in a peak linear heat generation rate (LHGR) of 16.2 kw/ft and a MCPR of 1.03 in the misplaced reload bundle. Fuel elements adjacent to a misplaced bundle will not be significantly affected by the presence of a misplaced bundle. The 16.2 kw/ft LHGR is acceptable; however, the MCPR of 1.03 is below the safety limit of 1.06.

The fuel loading accident is being generically reviewed by the NRC staff and generic resolution is anticipated. In the interim the licensee has agreed to increase the operating limit MCPR to 1.25 and 1.28 for 7 x 7 and 8 x 8 fuel assemblies, respectively, thereby assuring that the safety limit of 1.06 will not be violated by this accident. This will keep the rod bundle from boiling transition during steady state operation even if the worst misloading error should occur. We have concluded that this is acceptable.

RSCS Modification

NPPD has proposed to modify the RSCS by replacing the group "C" mode with the Group Notch mode. The proposed Group Notch mode would require that, during portions of the rod withdrawal or insertion sequences, a group of rods remain within one notch of each other. Repeated notch movement of one rod would be electrically prohibited. In this way, the maximum reactivity worth of a rod would be limited to an acceptable value such that a postulated control rod drop accident will not exceed the peak fuel enthalpy limit of 280 cal/gm. The Group Notch mode of the RSCS for CNS is presented in reference 5. Generic approval for the Group Notch mode has been granted in reference 11.

Based on the information submitted by NPPD and the acceptability of the generic approval of the Group Notch mode to CNS, the staff concludes that the use of the Group Notch mode of RSCS at CNS is acceptable.

Thermal-Hydraulic Stability Analysis

Descriptions of the types of thermal-hydraulic stability analyses are given in reference 2. The results of the analyses are contained in reference 1. General Electric is preparing a topical report to address the NRC staff's concerns regarding stability margin at conditions of low flow and low power. Until this topical report has been reviewed and accepted by the staff, operation in natural circulation at power levels approaching rod block power limits will be prohibited. NPPD has proposed a Technical Specification to preclude exceeding 25% of rated thermal power without forced recirculation. This power level assures a dynamic response decay ratio of less than or equal to 0.36 and is acceptable.

Recirculation Pump Startup From The Natural Recirculation Mode

During the review of the CNS reload application, NPPD was asked to provide analyses and startup test results which prove that recirculation pump startup from natural circulation operation does not cause a reactivity insertion transient in excess of the most severe coolant flow increase currently analyzed. Until the analyses and test results can be submitted, the NRC staff required that a requirement be incorporated into the CNS Technical Specifications to prevent starting recirculation pumps while the reactor is in natural circulation flow and reactor power is greater than 1% of rated thermal power. NPPD agreed to this requirement.

Physics Startup Testing

As part of our review of reload 1, NPPD was requested to provide a description of the Physics Startup testing program. In response to our request, this program was described in NPPD letter dated October 8, 1976. The results of these tests will be reported in a NPPD report within 30 days of completion of the tests. We find the Startup Physics testing program and reporting schedule acceptable.

Densification Power Spike Penalty

NPPD has requested a change in the Technical Specifications to remove a densification power spike penalty of 2.2% from the 8 x 8 fuel to increase the operating limit on the maximum linear heat generation rate. GE described a method to justify removal of the densification power spike penalty in Appendix B to reference 2 which we have not approved. Since the review is not complete, in the interim Cooper must use 13.4 kw/ft as the peak linear heat generation rate limit with the appropriate densification power spike penalty of 2.2%.

Modifications To The LPCI System

The LPCI system modifications are similar to those previously approved at the Vermont Yankee, Brunswick Unit No. 2, FitzPatrick, Peach Bottom Unit Nos. 2 and 3, and the Hatch Unit No. 1 facilities and are described in reference 3. The modifications consist of:

1. Selection Logic and Cross-tie Valve

   Removing the recirculation loop selection logic, key-locking closed the cross-tie valve (10-20) between the two LPCI system discharge piping headers, and providing an annunciator to indicate an open condition of the cross-tie valve;

2. Injection Signals and Valve Operation

   Rewiring of the system so that the automatic initiation signals direct both of the normally closed injection valves (10-25 A&B) to open and both of the recirculation pump discharge valves (2-53 A&B) to close upon detection of LOCA conditions;

3. Power Supply

   Supplying electrical power to the LPCI injection valve and recirculation pump discharge valve on each recirculation loop from one of the 250 volt DC station batteries; and

4. LPCI Pump Power Supply

   Changing the power supplies for LPCI pumps "1B" and "1C" such that LPCI pump "1B" will be assigned as an electrical load in Division 1 and LPCI pump "1C" will be assigned to Division II electrical loads.

In our Safety Evaluation issued with Amendment No. 31 to the CNS license, we found the installation of these modifications to be acceptable because the performance of the LPCI system would be improved. However, the Safety Evaluation for Amendment No. 31 stated that the acceptability of operation of CNS with the modified LPCI system would be contingent upon the NRC staff review and approval of (1) the conformance of the instrumentation and electrical system changes to the appropriate standards for electrical separation, (2) a study of the effects of water jet impingement from a postulated loss-of-coolant accident on the operability of the recirculation pump discharge valves, and (3) proposed LPCI modification Technical Specifications.

The modifications to the LPCI system submitted by NPPD on August 9, 1976, would have resulted in the retention of an automatic bus transfer (ABT) switch in the power supply to motor-control-center (MCC) - RB. Although the NRC staff has approved the use of ABT devices for other plants with LPCI system modifications on a short term or interim basis, the staff believes that such devices do not meet the requirements of Regulatory Guide 1.6, "Independence Between Standby Power Sources and Their Distribution Systems" and that the loop "A" valves (10-27A, 2-43A, and 2-54A) and the loop B valves (10-27B, 2-43B, and 2-54B) should be placed on separate, electrically independent busses to meet the single failure criterion. Therefore, by letter dated October 26, 1976, NPPD proposed a design change to eliminate the ABT and provide for complete separation, by electrical division, of the valve operators on MCC-RB. In the proposed design change, the Division I valves, 10-27A, 2-43A, and 2-54A would be removed from MCC-RB and placed on a new Division I MCC-CA. The Division II valves would remain on MCC-RB. The ABT would be removed from the circuit and MCC-RB would receive power directly from the Division II MCC-S. Thus, the potential problem associated with the ABT circuitry, with regard to meeting the single failure criterion, has been eliminated by the proposed electrical modification.

Electric Separation

The proposed instrumentation and electrical system changes for the modifications to the LPCI system have been reviewed by the NRC staff and found acceptable on the basis that they conform to the applicable NRC standards and guides and are expected to meet or exceed the qualifications of the unmodified LPCI system.

Water Jet Impingement Study

Section 3.1.5.2 - "Break Effects" of reference 3 states that a study of the drywell geometry is being performed to determine the effects of water jet impingement resulting from a postulated recirculation line break. For the suction line break, protective devices or barriers will be provided, if required, to prevent discharge valve operator damage. By letter dated October 20, 1976, NPPD reported the completion of the drywell geometry study. NPPD determined that for the suction line break, the physical location of the system equipment provides a barrier to prevent possible discharge valve operator damage; and, therefore, no protective devices or barriers are necessary.

Based on the results of the licensee's study, we conclude that water jet impingement from a postulated recirculation line break will not prevent the recirculation pump discharge valve operators from functioning as designed.

LPCI Modification Technical Specifications

See Section on Technical Specifications.

Recirculation Pump Discharge Valve Operation

In the modified LPCI system, the recirculation pump discharge valves are required to close upon initiation of a postulated LOCA. The design requirements for the CNS discharge valves are: 1) the recirculation pump discharge valves, which include both main pipe valve and discharge bypass valve, shall be signalled to close on receipt of the LPCI initiation signal and a reactor vessel pressure signal between 300 psia and 350 psia, 2) the valves shall stroke fully closed in 30±3 seconds following receipt of these signals, and 3) the reactor vessel pressure permissive for the discharge valve shall be independent of any other initiation signals such as core spray or LPCI injection valve opening.

The initiation of discharge valve closure between 300 and 350 psia reactor vessel pressure is necessary to prevent the valves from experiencing a differential pressure greater than the design differential pressure of 200 psid. However, the measured closing time for the CNS discharge valves is 25 seconds rather than the required 30±3 seconds. No analysis has been performed, assuming this faster closing time, to verify that the design differential pressure of the discharge valves will not be exceeded following a postulated LOCA. If the design differential pressure is exceeded, it is possible that the valves may be incapacitated before they close. Failure of the valves to close would negate the validity of the CNS ECCS analysis.

To assure that the discharge valves will close and that the current CNS ECCS analysis remains valid, NPPD has requested to change the initially proposed reactor vessel pressure permissive set point in the CNS Technical Specifications to between 200 and 250 psia and to change the initially proposed specification for recirculation pump discharge valve closure time to 23±3 seconds.

A comparison of the pressure transients and discharge valve closure times over the spectrum of LOCA breaks has been performed for the proposed set points and valve closure times. This showed that, with a reactor vessel pressure permissive set point between 200 and 250 psia and a discharge valve closure time between 20 and 26 seconds, the time to achieve full closure of the valves was shorter than the closure time with the previously proposed Technical Specification for all LOCA analyses, except for the smallest break considered (0.1 ft^2 break). A shorter closure time would enhance reflood conditions and thereby would improve the LOCA safety margins. For the 0.1 ft^2 break, although the initiation signal for valve closure occurs slightly later with the lower reactor vessel pressure set point, the increased time to achieve full valve closure would not significantly reduce the core reflood rate. This effect has been evaluated using the approved ECCS models and results in an increase in peak clad temperature of approximately 60°F above the previously calculated value of 1230°F for the 30±3 second closure time. The increased calculated peak clad temperature is still well below the 2200°F limit of 10 CFR Part 50 §50.46. Therefore, the consequences of the LOCA analyses for all breaks meet the requirements of 10 CFR 50 §50.46. Thus, based on satisfying these requirements, we have concluded that the proposed technical specification change is acceptable.

LPCI Pump Runout

By letter dated May 20, 1976, we requested NPPD to analyze and determine if long term heat removal capabilities for CNS could be impaired due to RHR (LPCI) Pump runout following a postulated loss-of-coolant accident, to provide a schedule for making system modifications, and to describe the testing program that will be used to verify that the required design modifications: (1) adequately protect the RHR (LPCI) Pumps from potential runout conditions; and (2) assure that LPCI system core reflood performance is in accordance with the current ECCS analysis for CNS. Runout is defined as the operation of a LPCI pump in excess of its design flow rate. Runout may result in loss of net positive suction head, cavitation, and/or pump motor overload. NPPD responded to our request by letters dated June 21, June 30, and October 1, 1976.

As described in the licensees letter of June 21, 1976, the modifications to the LPCI system described in reference 3 require flow tests to be conducted, prior to reactor operation, to develop pump flow characteristics curves. Based on these curves, NPPD will install orifice plates in the LPCI system discharge piping. The orifice plates would be of the proper size to assure both that the minimum LPCI flow of 8400 gpm per pump will be provided to meet the ECCS Appendix K requirements (reference 4) and that a maximum LPCI flow of 8800 gpm per pump will not be exceeded which could lead to runout conditions. In addition, assuming the single failure which results in maximum LPCI pump flow, NPPD has shown that, if the flow for one LPCI pump is restricted to 8800 gpm, neither the pump motor nor the diesel generator supplying the LPCI pumps will exceed recommended load limits as established in the CNS Final Safety Analysis Report. NPPD has proposed a Technical Specification requirement to establish the maximum flow rate for a LPCI pump at 8800 gpm.

Based on our review of the information provided by NPPD, we conclude that the operation of CNS with LPCI system orifice plates installed will not reduce the short term or long term cooling capabilities of the LPCI system and is acceptable.

Technical Specifications

The proposed Technical Specification changes for the Cooper Nuclear Station cycle 2 operation include:

A. Incorporation of 8 x 8 fuel assembly Technical Specification limits

B. Fuel cladding integrity safety limit MCPR and operating limit MCPR

C. RSCS Group Notch Mode

D. Densification power spike penalty elimination

E. Modifications to the LPCI system

F. Thermal-hydraulic stability

Some modifications to the proposed Technical Specifications were necessary to meet NRC staff requirements. The staff finds that the proposed Technical Specifications, as modified and with the exception of the elimination of the densification power spike penalty which has not been approved, are acceptable and consistent with the information submitted by the licensee.

During the review of the CNS reload submittal, other Technical Specification requirements were found necessary by the staff. These include:

1. Increase in operating limit MCPR

2. Recirculation pump startup from natural circulation

The staff finds these Technical Specifications to be acceptable.

ENVIRONMENTAL CONSIDERATION

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.

CONCLUSION

Based on our review of the items identified as (1) and (2) in the introduction of this evaluation, and the considerations discussed in this evaluation, we have concluded that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. Based on our review of the remaining two items of this evaluation relating to modified RSCS and LPCI System, and the considerations discussed in this evaluation, we have concluded that (1) because the items on RSCS and LPCI do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in safety margin, they do not involve a significant hazards consideration, and (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. We also have concluded, based on the considerations discussed in this evaluation, that all of the activities discussed herein will be conducted in compliance with the Commission's regulations and the issuance of an amendment to the license will not be inimical to the common defense and security or to the health and safety of the public.

Date: November 10, 1976

References

1. "General Electric BWR Reload 1 License Amendment Submittal for Cooper Nuclear Station," NEDO-21292, July 1976, with Supplement No. 1, NEDO-21292-01, September 1976.

2. "GE/BWR Reload Licensing Application for 8 x 8 Fuel," Licensing Topical Report, NEDO 20360, Rev. 1 Supplement 4.

3. "Cooper Nuclear Station Emergency Core Cooling System Low Pressure Coolant Injection Modifications for Performance Improvement," NEDO-21337, July 1976.

4. "Cooper Nuclear Station Loss-of-Coolant Accident Analysis in Conformance with 10 CFR 50 Appendix K with Modified Low Pressure Coolant Injection," NEDO-21335, July 1976.

5. "Group Notch Mode of the Rod Sequence Control System for Cooper Nuclear Station," NEDO-21291, June 1976.

6. "Supplemental Information for Plant Modifications to Eliminate Significant In-Core Vibration," NEDE-21156, January 1976 (Proprietary Report).

7. "General Electric BWR Thermal Analysis Basis (GETAB): Data Correlation and Design Application," NEDO-10958, November 1975.

8. Lenford, R. B., "Analytical Methods of Plant Transient Evaluation for the General Electric Boiling Water Reactor," NEDO-10802, February 1973.

9. Lenford, R. B., "Analytical Methods of Plant Transient Evaluations for GE/BWR Amendment No. 1," NEDO-10802-1, April 1975.

10. "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," Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, February 1976.

11. "Safety Evaluation Report on the Rod Sequence Control Systems with Group Notch (RSCS/BWR-4)," for Tennessee Valley Authority, Browns Ferry Nuclear Plants, Units 1, 2, and 3, Directorate of Licensing, U. S. Atomic Energy Commission, June 6, 1974.

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. 32 to Facility Operating License No. DPR-46, issued to 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 revised the Technical Specifications for the facility to authorize operation with (1) up to 120 General Electric 8 x 8 type reload fuel assemblies, (2) modified fuel assembly lower tie plates as part of facility modifications to eliminate in-core vibration of instrument and source tubes, (3) a modified Rod Sequence Control System, and (4) modifications to improve the performance of the Low Pressure Coolant Injection System of the Emergency Core Cooling System.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in

10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with items (1) and (2) above was published in the FEDERAL REGISTER on October 4, 1976 (41 FR 43783). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action on items (1) and (2) above. Prior public notice of items (3) and (4) above was not required since these actions do 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 applications for amendment dated April 7 and August 9, 1976 and supplements dated July 20, September 20, October 8, 9, 15, 20 and 26, and November 4 and 10, 1976, (2) Amendment No. 32 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 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 10th day of November, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors