OCT 31 1975

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 16 to Facility License No. DPR-46 for the Cooper Nuclear Station. The amendment includes Change No. 19 to the Technical Specifications and is in response to your requests dated July 10, 1975 and July 14 and supplements there to dated September 12, 1975 and October 7, 17 and 24, 1975.

The amendment authorizes operation of Cooper (1) using operating limits based on the General Electric Thermal Analysis Basis (GETAB), and (2) with modified operating limits based on an acceptable evaluation model that conforms with the requirements of Section 50.46 of 10 CFR Part 50 of the Commission's regulations.

The Commission also has issued the enclosed Order for Modification of License which authorizes operation of the Cooper facility with plugged bypass flow holes, subject to the conditions set forth in Change No. 19 issued with Amendment No. 16 to the license, in accordance with your application dated September 11, 1975, as supplemented. The Order restricts operation with one recirculation loop out of service and modifies the restriction on operation with one automatic depressurization valve out of service. This Order supersedes the December 27, 1974 Order for Modification of License and the October 8, 1975 Order for Modification of License in their entirety.

The Commission's staff has evaluated the potential for environmental impact associated with operation of the Cooper Nuclear Station in the manner set forth in item (2) of the second paragraph above. From this evaluation, the staff has determined that there will be no change in

See note re Dck rlc

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effluent types or total amounts, no change in authorized power level and no significant environmental impact attributable to that action. Having made this determination, the Commission has further concluded pursuant to 10 CFR Section 51.5(c)(1) that no environmental impact statement need be prepared for this action. Copies of the related Negative Declaration and supporting Environmental Impact Appraisal also are enclosed. As required by Part 51, the Negative Declaration is being filed with the Office of the Federal Register for publication.

Copies of the related Safety Evaluation and Federal Register Notice are also enclosed.

Sincerely,

Original signed by
Dennis L. Ziemann

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

Enclosures:
1. Amendment No. 16 to License No. DPR-46 w/Change No. 19
2. Order for Modification of License
3. Negative Declaration with Impact Appraisal
4. Safety Evaluation
5. Federal Register Notice

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

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

Auburn Public Library
1118 - 15th Street
Auburn, Nebraska 68305

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

cc w/enclosures and cy of NPPD's filings dtd. 9/12/75, 10/7 17 (3 filings) and 24/75:
Mr. James L. Higgins, Director
Department of Environmental Control
Executive Building, 2nd Floor
Lincoln, Nebraska 68509

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EGCase
BCRusche

The licensee was notified by B. Rusche that the Order and Amendment # 16 were signed on 10/31. I&E was informed 8:30 AM on 11/3. R. Silver

OFFICE ➤	RL:ORB #2	TR	OELD	RL:AD/ORs	RL:A/D	NRR:D/DIR	NRR:DIR
SURNAME ➤	RMDiggs RDSilver:ah DLZiemann			KRGoller	RSBoyd	EGCase	BCRusche
DATE ➤	10/31/75	10/ /75		10/31/75	10/31/75	10/31/75	10/31/75

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 16
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 July 10, 1975, July 14, 1975 and September 11, 1975 and supplements thereto dated September 12, 1975 and October 7, 17 and 24, 1975, 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, as amended, 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; and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this License amendment and Paragraph 2.C(2) of Facility License No. DPR-46 is hereby amended to read as follows:

OFFICE➤						
SURNAME➤						
DATE➤						

"2.C(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 19."

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:
Karl R. Goller

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

Attachment:
Change No. 19 to the
Technical Specifications

Date of Issuance: OCT 31 1975

OFFICE ➤						
SURNAME ➤						
DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NO. 16

CHANGE NO. 19 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the existing pages of the Technical Specifications listed below with the attached revised pages bearing the same numbers, except as otherwise noted. Changed areas on these pages are shown by marginal lines:

i
ii
5
5a - Add
6 thru 22, inclusive
27
28
31
42
43
61
62
85
97
102
120
123
127
129
130
131
131a - Delete

137
151
210
210a - Delete
211
212
213
214
214a
214b - Add
214c - Add
214d - Add
214e - Add
215
*215a - Add
*215b - Add
*215c - Add
*215d - Add
*215e - Add
*215f - Add
*216
*216a - Add

*These pages are reissued because of Section number changes and page numbers.

RADIOLOGICAL TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS

19

TABLE OF CONTENTS (cont'd)

U. Safety Limits - The safety limits are limits within which the reasonable maintenance of the fuel cladding integrity and the reactor coolant system integrity are assured. Violation of such a limit is cause for unit shutdown and review by the Atomic Energy Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

V. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1. At least one door in each access opening is closed.

2. The standby gas treatment system is operable.

3. All automatic ventilation system isolation valves are operable or secured in the isolated position.

W. Shutdown - The reactor is in a shutdown condition when the mode switch is in the "Shutdown" or "Refuel" position.

1. Hot Shutdown means conditions as above with reactor coolant temperature greater than 212°F.

2. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F and the reactor vessel vented.

X. Surveillance Frequency - Unless otherwise stated in these specifications, periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted plus or minus 25%. The operating cycle interval as pertaining to instrument and electrical surveillance shall never exceed 15 months. In cases where the elapsed interval has exceeded 100% of the specified interval, the next surveillance interval shall commence at the end of the original specified interval.

Y. Surveillance Interval - The surveillance interval is the calendar time between surveillance tests, checks, calibrations and examinations to be performed upon an instrument or component when it is required to be operable. These tests may be waived when the instrument, component or system is not required to be operable, but the instrument, component or system shall be tested prior to being declared operable or as practicable following its return to service.

Z. Thermal Parameters

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

2. Maximum Total Peaking Factor - The Maximum Total Peaking Factor (MTPF) 19
is the lowest Total Peaking Factor which limits a fuel type to a Linear Heat Generation Rate (LHGR) corresponding to the operating limit at 100% power.

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

4. Total Peaking Factor - The ratio of the maximum fuel rod surface heat flux in an assembly to the average surface heat flux of the core. 19

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

SAFETY LIMITS

1.1 FUEL CLADDING INTEGRITY

Applicability

The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objective

The objective of the Safety Limits is to establish limits below which the integrity of the fuel cladding is preserved.

Specifications

19 A. Reactor Pressure >800 psia and Core Flow >10% or Rated

The existence of a minimum critical power ratio (MCPR) less than 1.05 shall constitute violation of the fuel cladding integrity safety.

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

When the reactor pressure is $\leq$800 psia or core flow is less than 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

C. Power Transient

To ensure that the Safety Limit established in Specification 1.1.A and 1.1.B is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal.

LIMITING SAFETY SYSTEM SETTINGS

2.1 FUEL CLADDING INTEGRITY

Applicability

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objective

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specifications

A. Trip Settings 19

The limiting safety system trip settings shall be as specified below:

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

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

where:

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

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2 million lb/hr)

1.1.D (Cont'd)

Whenever the reactor is in
the cold shutdown condition
19 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
= (TBS) 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 19
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.

SAFETY LIMITS	LIMITING SAFETY SYSTEM SETTINGS

2.1.A (Cont'd)

d. APRM Rod Block Trip Setting

The APRM rod block trip setting shall be:

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

where:

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

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2 million lb/hr) 19

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
= (TBS) 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.

SAFETY LIMITS	LIMITING SAFETY SYSTEM SETTINGS

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

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

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

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

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

   $\leq$10 percent valve closure when above 1000 psig reactor pressure, in 3 out of 4 main steam lines. 19

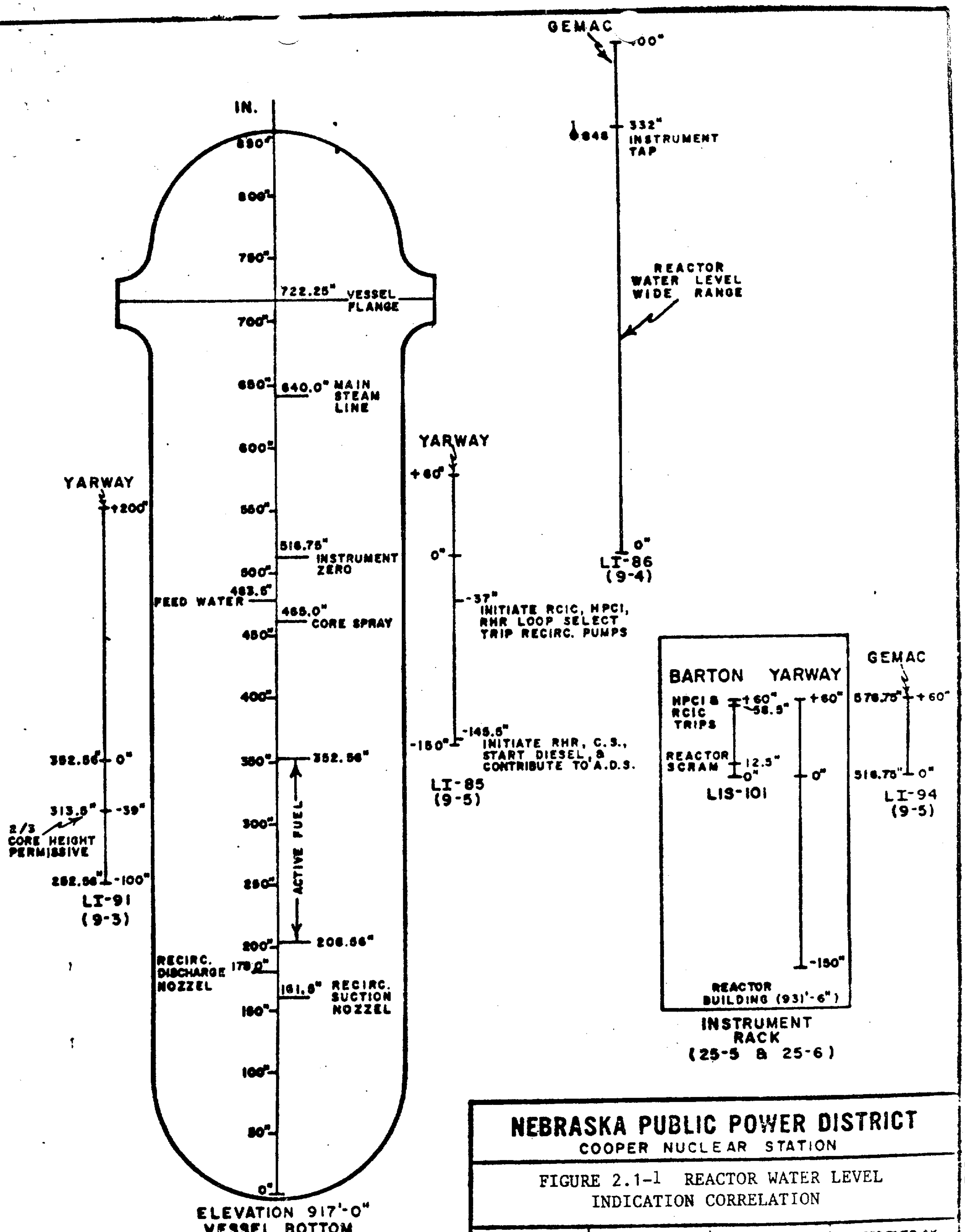
6. <u>Main Steam Line Isolation Valve Closure on Low Pressure</u>

   $\geq$850 psig when mode switch is in "Run".

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

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

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



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.	

1.1 Bases:

Fuel Cladding Integrity

A. Fuel Cladding Integrity Limit at Reactor Pressure >800 psia and Core Flow >10% of Rated

The fuel cladding integrity safety limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedure used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore the fuel cladding integrity safety limit is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within
19 the core and all uncertainties.

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB[1], which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) - Boiling Length (L), GEXL, correlation.

The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation. These conditions are:

Pressure:	800 to 1400 psia	
Mass Flux:	0.1 to 1.25 10^6 lb/hr	
Inlet Subcooling:	0 to 100 Btu/lb	
Local Peaking:	1.61 at a corner rod to	
	1.47 at an interior rod	
Axial Peaking:	Shape	Max/Avg.
	Uniform	1.0
	Outlet Peaked	1.60
	Inlet Peaked	1.60
	Double Peak	1.46 and 1.38
	Cosine	1.39

1.1 Bases: (Cont'd)

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

The required input to the statistical model are the uncertainties listed on Table 1.1-1, the nominal values of the core parameters listed in Table 1.1-2, and the relative assembly power distribution shown in Table 1.1-3. Table 1.1-4 shows the R-factor distributions that are input to the statistical model which is used to establish the safety limit MCPR. The R-factor distributions shown are taken near the beginning of the fuel cycle. The basis for the uncertainties in the core parameters is given in NEDO-20340[2] and the basis for the uncertainty in the GEXL correlation is given in NEDO-10958[1]. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution
19 in Cooper Nuclear Station during any fuel cycle would not be as severe as the distribution used in the analysis.

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

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low power and all flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psi or core flow less than 10% is conservative.

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 1.1A or 1.1B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main

1.1 Bases: (Cont'd)

turbine stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

The computer provided with Cooper has a sequence annunciation program which will indicate the sequence in which events such as scram, APRM trip initiation, pressure scram initiation, etc. occur. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 1.1.C will be relied on to determine if a Safety Limit has been violated.

19 D. Reactor Water Level (Shutdown Condition)

During periods when the reactor is shutdown, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 18 inches above the top of the fuel provides adequate margin.

References

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

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

1.1 Bases: (Cont'd)

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
19 Friction Factor Multiplier	10.0
Channel Friction Factor Multiplier	5.0
TIP Readings	6.3
Bypass void effect on TIP	4.12 (core midplane) 5.41 (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 (High Enriched Bundle)
	1.154 (Low Enriched 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
	1.275 to 1.325	16.8
	1.225 to 1.275	8.2
	1.175 to 1.225	7.2
	1.125 to 1.175	5.0
	1.075 to 1.125	12.0
19	1.025 to 1.075	4.6
	0.975 to 1.025	7.0
	0.875 to 0.975	4.0
	0.875 to 0.925	2.0
	0.825 to 0.875	4.4
	0.775 to 0.825	3.
	0.675 to 0.775	2.0
	0.625 to 0.675	5.0
	0.575 to 0.625	4.2
	0.275 to 0.575	14.6
		Sum = 100

1.1 Bases: (Cont'd)

Table 1.1-4

R-FACTOR DISTRIBUTION USED IN GETAB STATISTICAL ANALYSIS

7 x 7 Rod Array

	R-Factor	Rod Sequence No.
	1.098	1
	1.083	2
19	1.075	3
	1.062	4
	1.052	5
	1.042	6
	1.042	7
	≤1.027	8 thru 49

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 19
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 1.29 for 7x7 fuel and (TBS) for 8x8 fuel is conservatively assumed to exist prior to initiation of the transients.

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

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

2.1 <u>Bases</u>: (Cont'd)

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

In summary:

i. The abnormal operational transients were analyzed to a power level of 2381 MWt.

ii. The licensed maximum power level is 2381 MWt.

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

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

A. <u>Trip Settings</u>

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

1. <u>Neutron Flux Trip Settings</u> 19

a. <u>APRM Flux Scram Trip Setting (Run Mode)</u>

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

2.1 Bases: (Cont'd)

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of 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 (TBS) 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 > 1.29 for 7x7 fuel and (TBS) for 8x8 fuel.

b. APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode) 19

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 19
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 (TBS) for 8x8 fuel, thus preserving the APRM rod block safety margin.

2.1 **Bases:** (Cont'd)

2. <u>Reactor Water Low Level Scram and Isolation Trip Setting (except MSIV)</u>

The set point for low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR Subsection 14.5 show that scram at this level adequately protects the fuel and the pressure barrier, because MCPR remains well above 1.06 in all cases, and system pressure does not reach the safety valve settings. The scram setting is approximately 25 in. below the normal operating range and is thus adequate to avoid spurious scrams.

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

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of $\leq$10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above 1.06 even during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure.

4. <u>Turbine Control Valve Fast Closure Scram Trip Setting</u> 19

The turbine control valve fast closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to lead rejection exceeding the capability of the bypass valves. The reactor protection system initiates a scram when fast closure of the control valves is initiated by the acceleration relay. This setting and the fact that control valve closure time is approximately twice as long as that for the stop valves means that resulting transients, while similar, are less severe than for stop valve closure. No significant change in MCPR occurs. Relevant transient analyses are presented in Paragraph 14.5.1.1 of the Final Safety Analysis Report.

2.1 Bases: (Cont'd)

5. Main Steam Line Isolation Valve Closure on Low Pressure

The low pressure isolation of the main steam lines at 850 psig was provided to protect against rapid reactor depressurization.

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

The core standby cooling subsystems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to limit fuel clad temperature, to assure that core geometry remains intact and to limit any clad metal-water reaction to less than 1%. To accomplish their intended function, the capacity of each Core Standby Cooling System component was established based on the reactor low water level scram set point. To lower the set point of the low water level scram would increase the capacity requirement for each of the CSCS components. Thus, the reactor vessel low water level scram was set low enough to permit margin for operation, yet will not be set lower because of CSCS capacity requirements.

The design of the CSCS components to meet the above guidelines 19
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 Analysis 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).

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

To assure the operability of the reactor protection system.

Specification:

The setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.1. 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 (TBS) for 8x8 fuel. 19

C. During reactor power operation with TPF $\geq$ 2.61 for 7x7 fuel and (TBS) 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. 19

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

COOPER NUCLEAR STATION
TABLE 3.1.1
REACTOR PROTECTION SYSTEM INSTRUMENTATION REQUIREMENTS

Reactor Protection System Trip Function	Applicability Conditions — Mode Switch Position: Shutdown	Startup	Refuel	Run	Trip Level Setting	Minimum Number of Operable Channels Per Trip System (1)	Action Required When Equipment Operability is Not Assured (1)
Mode Switch in Shutdown	X(7)	X	X	X		1	A
Manual Scram	X(7)	X	X	X		1	A
IRM (17) High Flux	X(7)	X	X	(5)	≤120/125 of indicated scale	3	A
Inoperative		X	X	(5)		3	A
APRM (17)							
High Flux (Flow biased)				X	≤(0.66W+54%) (14) $\frac{A.}{MTPF}$	2	A or C
High Flux	X(7)	X (9)	X (9)	(16)	≤15% Rated Power		A or C
Inoperative		X (9)	X (9)	X	(13)	2	A or C
Downscale		(11)		X(12)	≥ 2.5% of indicated scale	2	A or C
High Reactor Pressure NBI-PS-55 A,B,C, & D		X (9)	X(10)	X	≤ 1045 psig	2	A
High Drywell Pressure PC-PS-12 A,B,C, & D		X (9)	X(8)	X	≤2 psig	2	A or D
Reactor Low Water Level NBI-LIS-101 A,B,C, & D		X	X	X	≥ + 12.5 in. indicated level	2	A or D
Scram Discharge Volume High Water Level CRD-LS-231 A,B,C, & D	X (2)(7)	X	X (2)	X	≤36 gallons	2	A

11. The APRM downscale trip function is only active when the reactor mode switch is in run.

12. The APRM downscale trip is automatically bypassed when the mode switch is not in RUN.

13. An APRM will be considered inoperable if there are less than 2 LPRM inputs per level or there is less than 11 operable LPRM detectors to an APRM.

14. W is the recirculation flow in percent of rated flow. 19
A = 2.61 for 7x7 fuel
  = (TBS) 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 19
and (TBS) for 8x8 fuel that MCPR will be as low as 1.06. Usually with peaking factors of this magnitude the peak occurs low in the core in a low quality region where the initial heat

3.1 BASES

4.1 BASES (Cont'd)

flux is very high. Therefore, with TPF <2.61 for 7x7 fuel and (TBS) for 8x8 fuel there are no technical specification requirements for calculating MCPR. With TPF greater than 2.61 for 7x7 fuel and (TBS) for 8x8 fuel MCPR is sufficient 19
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.

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TABLE 3.2.C
CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

Function	Trip Level Setting	Minimum Number Of Operable Instrument Channels/Trip System (5)
APRM Upscale (Flow Bias)	$\leq (0.66W + 42\%) \frac{A}{MTPF}$ (2) 19	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)	$\leq (0.66W + 41\%)$ (2) 19	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) 19

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
   = (TBS) for 8x8 fuel. 19

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) RMM fails to null
       (4) Less than required number of LPRM inputs for rod selected.

3.2 BASES (cont'd)

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

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

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

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

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

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

The control rod block functions are provided to prevent excessive control rod 19
withdrawal so that MCPR does not decrease to 1.06. The trip logic for this function is 1 out of n: e.g., any trip on one of six APRM's, eight IRM's, or four SRM's will result in a rod block.

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

The APRM rod block function is flow biased and prevents a significant reduction
in MCPR, especially during operation at reduced flow. The APRM provides gross 19
core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that
MCPR is maintained greater than 1.06. 19

The RBM rod block function provides local protection of the core; i.e., the

LIMITING CONDITION FOR OPERATION

3.3.B (cont'd)

6. During operation with limiting control rod patterns, as determined by the designated qualified personnel, either:

a. Both RBM channels shall be operable: or

b. Control rod withdrawal shall be blocked: or

19 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.

C. <u>Scram Insertion Times</u>

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	5.0

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	5.300

SURVEILLANCE REQUIREMENT

4.3.B (cont'd)

C. <u>Scram Insertion Times</u>

1. After each refueling outage all operable fully withdrawn insequence rods shall be scram time tested from the fully withdrawn position with the nuclear system pressure above 800 psig. This testing shall be completed prior to synchronizing the main turbine generator initially following restart of the plant.

Prior to exceeding 35% power all operable control rods shall be tested as described above.

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.

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

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

   A limiting control rod pattern is a pattern which results in the core
   being on a thermal hydraulic limit (i.e., MCPR = 1.06 or LHGR = 18.5kW/ft). 19
   During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Reactor Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other personnel qualified to perform this function may be designated by the station superintendent.

C. <u>Scram Insertion Times</u>

The control rod system is designed to bring the reactor subcritical
at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR 19
from becoming less than 1.06. The limiting power transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram (FSAR Figure III.6.15) with the average response of all the
drives as given in the above specification, provide the required protection, 19
and MCPR remains greater than 1.06.

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

3.5.E (cont'd.)

2. From and after the date that one
valve in the automatic depressuriza-
tion subsystem is made or found to be
inoperable for any reason, continued
reactor operation is permissible only
19 during the succeeding seven days un-
less such valve is sooner made opera-
ble, provided that during such seven
days the HPCI subsystem is operable.

3. If the requirements of 3.5.E.1 or 3.5.E.2 cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to at least 113 psig within 24 hours.

F. <u>Minimum Low Pressure Cooling and Diesel Generator Availability</u>

1. During any period when one diesel generator is inoperable, continued reactor operation is permissible only during the succeeding seven days unless such diesel generator is sooner made operable, provided that all of the low pressure core and containment cooling subsystems and the remaining diesel generator shall be operable and the requirements of 3.9.A.1 are met. If this requirement cannot be met, the requirements of 3.5.F.2 shall be met.

2. During any period when both diesel generators are inoperable, continued reactor operation is permissible only during the succeeding 24 hours unless one diesel generator is sooner made operable, provided that all the low pressure core & containment cooling subsystems are operable & the reactor power level is reduced to 25% of rated power and the requirements of 3.9.A.1 are met. If this requirement cannot be met, either the requirements shall be met or an orderly shutdown shall be initiated and the reactor placed in the cold shutdown condition within 24 hours.

4.5.E (cont'd.)

2. When it is determined that one valve of the ADS is inoperable, the ADS subsystem actuation logic for the other ADS valves and the HPCI subsystem shall be demonstrated to be operable immediately and at least weekly thereafter.

F. <u>Minimum Low Pressure Cooling and Diesel Generator Availability</u>

1. When it is determined that one diesel generator is inoperable, all low pressure core cooling and containment cooling subsystems shall be demonstrated to be operable immediately and daily thereafter. In addition, the operable diesel generator shall be demonstrated to be operable immediately and daily thereafter.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.5.H. <u>Engineered Safeguards Compartments Cooling</u> If the unit coolers serving the Reactor Core Isolation Cooling (RCIC), High Pressure Coolant Injection (HPCI), Core Spray or Residual Heat Removal (RHR) pump are out of service, the associated pump shall be considered inoperable for purposes of Specifications 3.5.A, 3.5.C, or 3.5.D as applicable. 19	4.5.H. <u>Engineered Safeguards Compartments Cooling</u> The unit coolers for the RCIC, HPCI, Core Spray, and RHR pumps shall be checked for operability during surveillance testing of the associated pumps. 19

3.5.C BASES (cont'd.)

fication to assure that when the reactor is being started up from a Cold Condition, the HPCI is not known to be inoperable.

D. RCIC System

The RCIC is designed to provide makeup to the nuclear system as part of the planned operation for periods when the main condenser is unavailable. The nuclear safety analysis,FSAR Appendix G, shows that RCIC provides water to cool the fuel when feed water is lost. In all other postulated accidents and transients, the ADS provides redundancy for the HPCI. Based on this and judgements on the reliability of the HPCI system, an allowable repair time of 7 days is specified. Immediate and weekly demonstrations of HPCI operability during RCIC outage is considered adequate based on judgement and practicality.

E. Automatic Depressurization System (ADS)

The limiting conditions for operating the ADS are derived from the Station Nuclear Operational Analysis (Appendix G) and a detailed functional analysis of the ADS (Section VI.).

This specification ensures the operability of the ADS under all conditions for which the automatic or manual depressurization of the nuclear system is an essential response to station abnormalities.

The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system so that the low pressure coolant injection (LPCI) and the core spray subsystems can operate to protect the fuel barrier.

Because the Automatic Depressurization System does not provide makeup to the reactor primary vessel, no credit is taken for the steam cooling of the core caused by the system actuation to provide further conservatism to the CSCS. Performance analysis of the Automatic Depressurization System is considered only with respect to its depressurizing effect in conjunction with LPCI or Core Spray . There are six valves provided and each has a capacity of 800,000 lb/hr at a set pressure of 1080 psig.

The allowable out of service time for one ADS valve is determined as seven 19
days because of the redundancy and because the HPCIS is demonstrated to be operable during this period. Therefore, redundant protection for the core with a small break in the nuclear system is still available.

The ADS test circuit permits continued surveillance on the operable relief valves to assure that they will be available if required.

F. Minimum Low Pressure Cooling and Diesel Generator Availability

The purpose of Specification F is to assure that adequate core cooling equip-

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4.5 BASES

Core and Containment Cooling Systems Surveillance Frequencies

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgement and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel, which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling systems, the components which make up the system; i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested each month to assure their operability. A simulated automatic actuation test once each cycle combined with frequent tests of the pumps and injection valves is doomed to be adequate testing of these systems.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure or design deficiency caused the outage, then the demonstration of operability should be thorough enough to assure that a generic problem does not exist. For example, if an out-of-service period were caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

Redundant operable components are subjected to increased testing during equipment out-of-service times. This adds further conservatism and increases assurance that adequate cooling is available should the need arise.

19

LIMITING CONDITIONS FOR OPERATION

3.6.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.

F. Jet Pump Flow Mismatch

1. When both recirculation pumps are in steady state operation, the speed of the faster pump may not exceed 110% the speed of the slower pump when core power is 80% or more of rated power or 115% the speed of the slower pump when core power is below 80% of rated power.

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
19 for a period in excess of 24 hours with one recirculation loop out of service.

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.

SURVEILLANCE REQUIREMENTS

4.6.E. Jet Pumps

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 two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

   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

1. Recirculation pump speeds shall be checked and logged at least once per day.

G. Structural Integrity

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

The LPCI loop selection logic has been previously described in the FSAR. For some limited low probability accidents with the recirculation loop operating with large speed differences, it is possible for the logic to select the wrong loop for injection. For these limited conditions the core spray itself is adequate to prevent fuel temperatures from exceeding allowable limits. However, to limit the probability even further, a procedural limitation has been placed on the allowable variation in speed between the recirculation pumps.

An analyses indicates that above 80% power the loop select logic could be expected to function at a speed differential up to 14% of their average speed. Below 80% power the loop select logic would be expected to function at a speed differential up to 20% of their average speed. This specification provides margin because the limits are set at ± 10% and ± 15% of the average speed for the above and below 80% power cases, respectively. If the reactor is operating on one pump, the loop select logic trips that pump before making the loop selection.

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. 19

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

3.11 FUEL RODS

Applicability

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

Objective

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

Specifications

19 A. Average Planar Linear Heat Generation Rate (APLHGR)

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

B. Linear Heat Generation Rate (LHGR)

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

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

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

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

SURVEILLANCE REQUIREMENTS

4.11 FUEL RODS

Applicability

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

Objective

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

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power. 19

B. Linear Heat Generation Rate (LHGR)

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

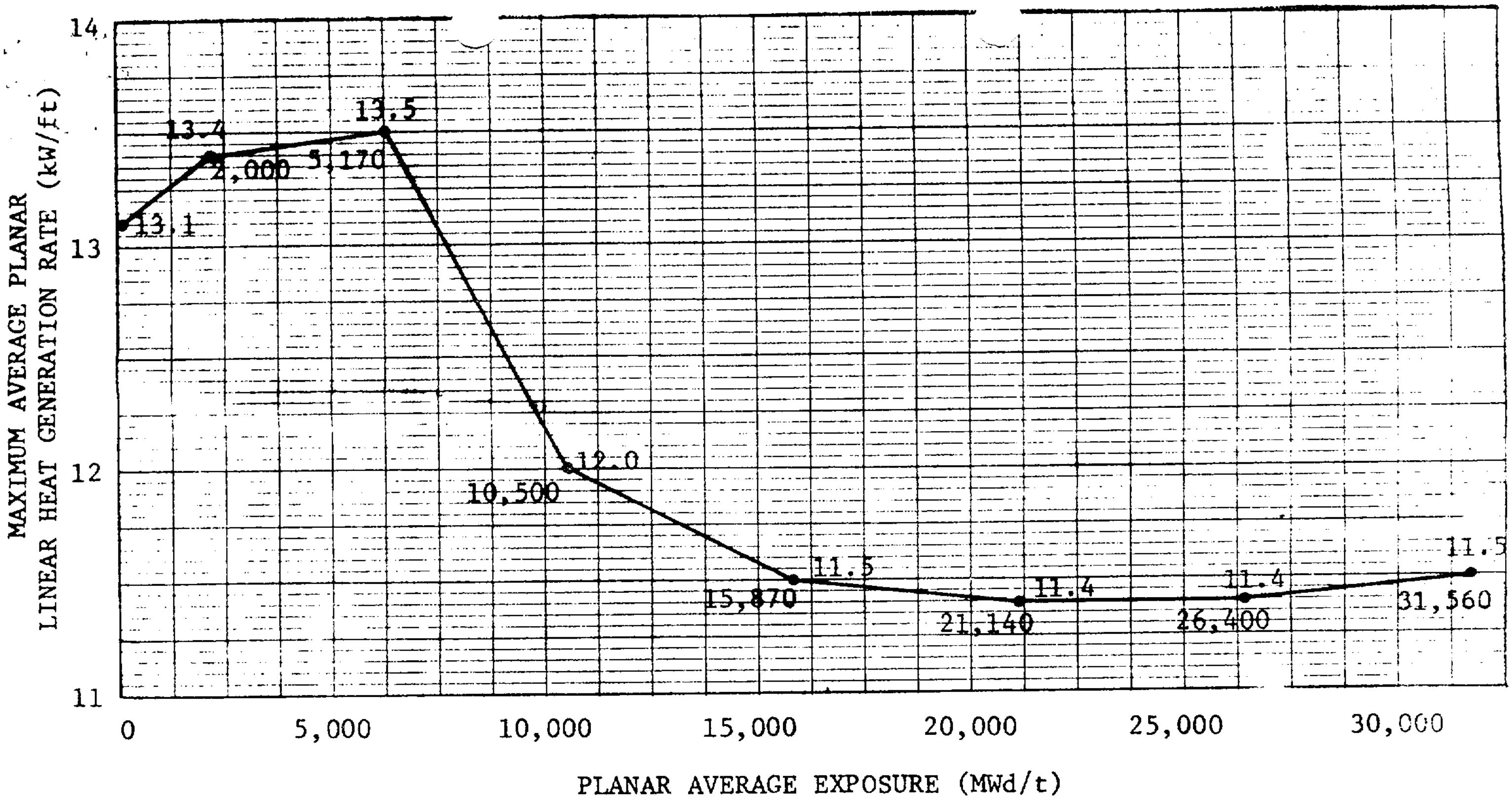


Figure 3.11-1.1. Maximum Average Linear Heat Generation Rate versus Planar Average Exposure, Initial Core, Fuel Types I and III.

19

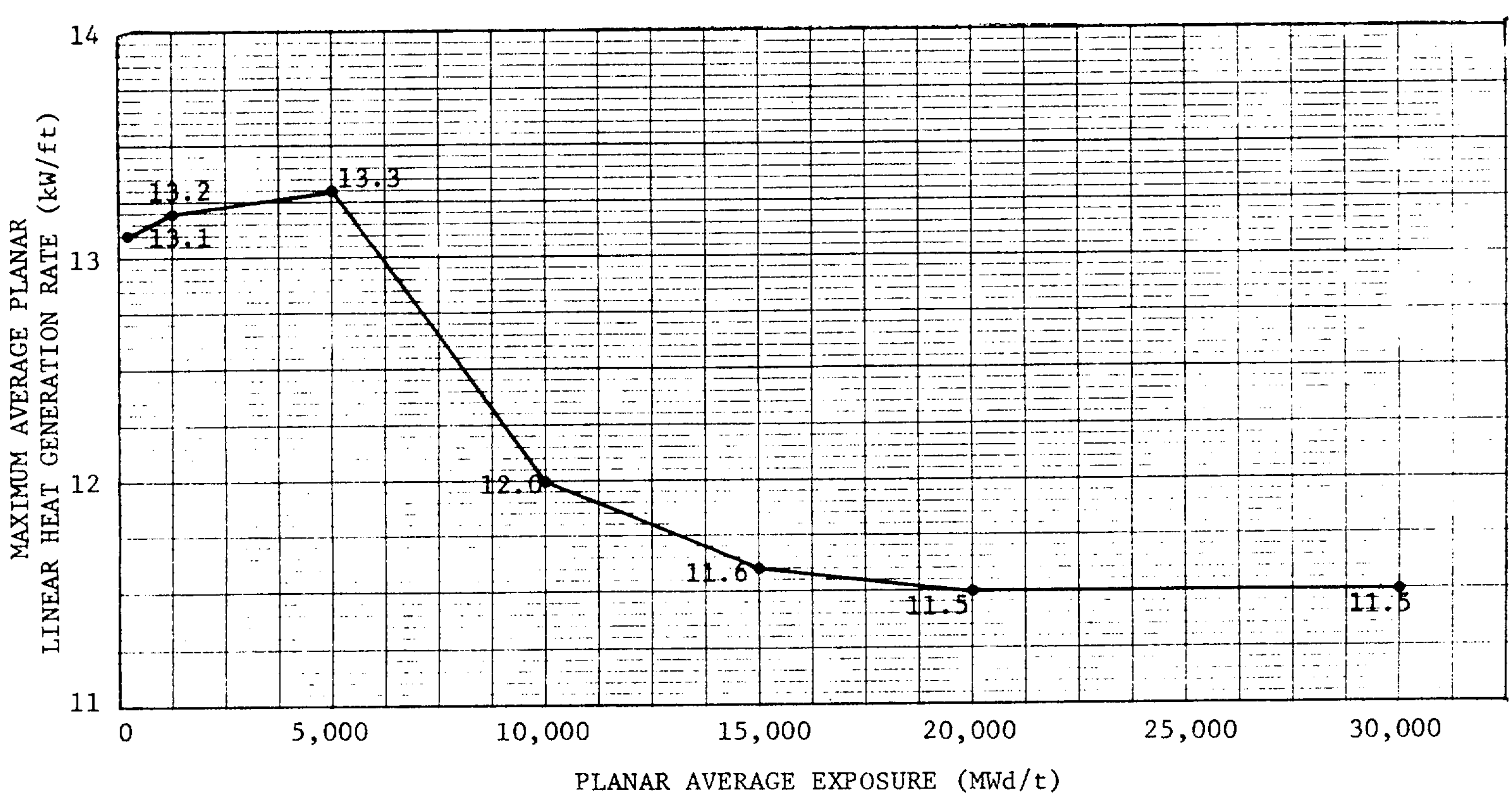


Figure 3.11-1.2. Maximum Average Linear Heat Generation Rate versus Planar Average Exposure, Initial Core, Fuel Type II

LIMITING CONDITIONS FOR OPERATION

LT = Total core length = 12 feet

L = Axial position above bottom of core

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

N = 0.038 for 7x7 fuel bundles
= 0.TBS for 8x8 fuel bundles

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

C. Minimum Critical Power Ratio (MCPR)

During steady state power operation MCPR shall be $\geq$ 1.29 for 7x7 fuel and $\geq$ (TBS) 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 to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

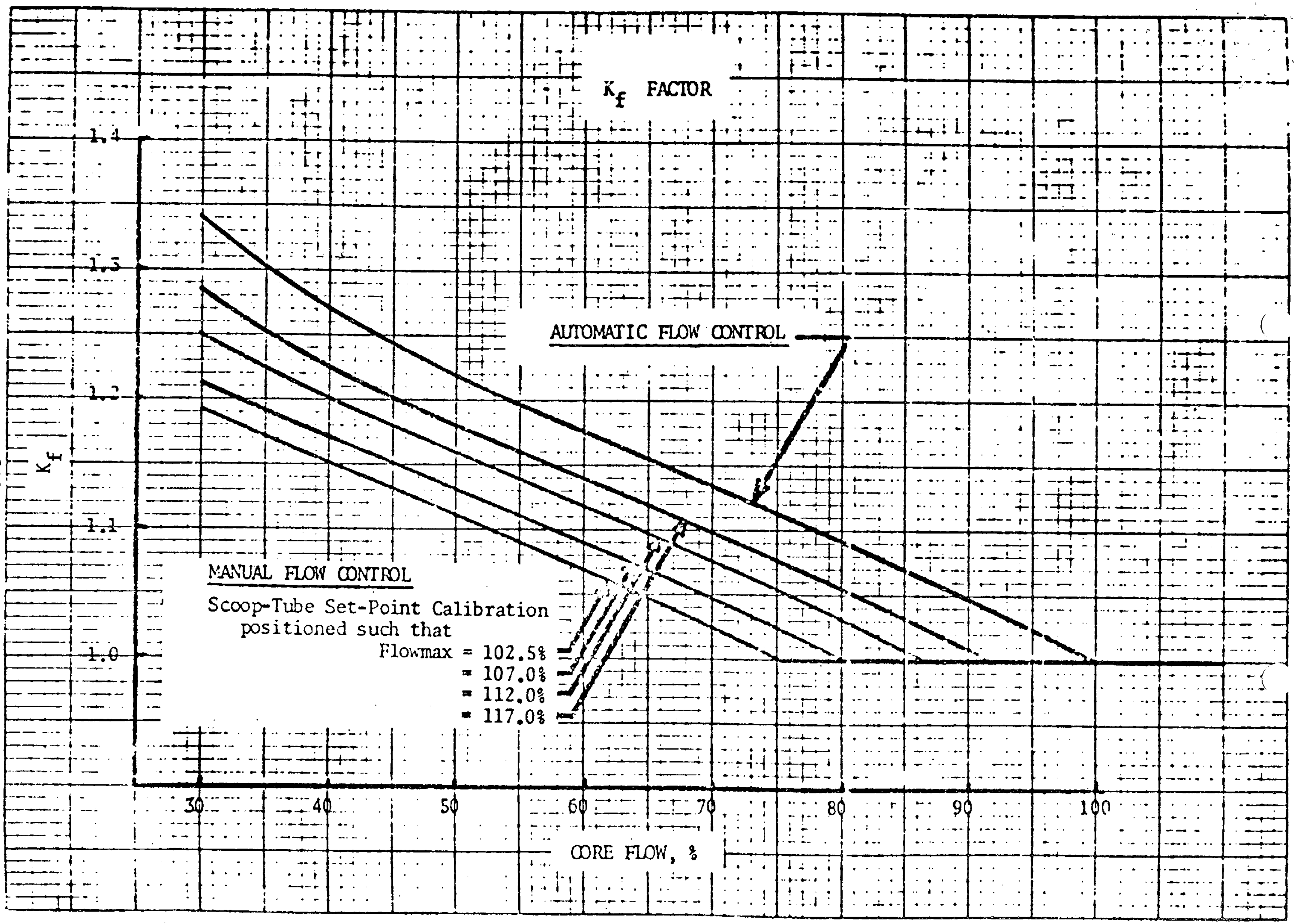
For core flows other than rated the MCPR shall be >1.29 for 7x7 fuel and < (TBS) for 8x8 fuel times K_f, where K_f is as shown in Figure 3.11-2.

SURVEILLANCE REQUIREMENTS

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at > 25% rated thermal power and following any change in power
level or distribution that would 19
cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

FIGURE 3.11-2



Note: Applicable to MG set plants only.

3.11 Bases

A. Average Planar Linear Heat Generation Rate (APLHGR)

This specifications assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR 50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod-to-rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm$ 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperature are within the 10 CFR 50, Appendix K limit. The limiting value for APLHGR is shown in Figure 3.11.1.

The calculational procedure used to establish the APLHGR shown on Figure 3.11.1 is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. 19
Differences in this analysis as compared to previous analyses performed with Reference 1 are: (1) The analyses assumes a fuel assembly planar power consistent with 102% of the MAPLHGR shown in Figure 3.11.1; (2) Fission product decay is computed assuming an energy release rate of 200 MEV/Fission; (3) Pool boiling is assumed after nucleate boiling is lost during the flow stagnation period; (4) The effects of core spray entrainment and counter-current flow limiting as described in Reference 2, are included in the reflooding calculations.

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

3.11 Bases: (Cont'd)

Table 3.11-1

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

PLANT PARAMETERS:

Core Thermal Power	2486 MWt which corresponds to 105% of rated steam flow
Vessel Steam Output	10.0×10^6 lbm/h which corresponds to 105% of rated steam flow
Vessel Steam Dome Pressure	1055 psia
Design Basis Recirculation Line Break Area	4.28* and 1.0 19
Recirculation Line Break Area for Small Breaks	1.0, 0.1 and 0.35

FUEL PARAMETERS:

Fuel Type	Fuel Bundle Geometry	Peak Technical Specification Linear Heat Generation Rate (KW/ft)	Design Axial Peaking Factor	Initial Minimum Critical Power Ratio
Type 1 and 3	7 x 7	18.5	1.4	1.18
Type 2	7 x 7	18.5	1.4	1.18

A more detailed list of input to each model and its source is presented in Section II of Reference 1.

*The DBA area includes: the area of the recirculation suction line (3.65 ft^2); plus the throat area of ten jet pumps (0.63 ft^2).

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 3.2.1 of Reference 1 and in References 2 and 3, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be 19
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[1]. 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.

3.11 Bases: (Cont'd)

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 ΔMCPR. 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 7-1 (page 7-6) of NEDO-21072.[1]

The evaluation of a given transient begins with the system
initial parameters shown in Table 7-1 (page 7-6) of NEDO-21072[1]
that are input to a GE core dynamic behavior transient computer
program described in NEDO-10802[3]. Also, the void reactivity 19
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 1.29 will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated for the same postulated transient event.

3.11 Bases (Cont'd)

The K_f factor curves shown in Figure 3.11-2 were developed generically which are applicable to all BWR/2, BWR/3, and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to rated thermal power at rated core flow.

For the manual flow control mode, the K_f factors were calculated such that at the maximum flow state (as limited by the 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 19
Cooper operation because the operating limit MCPR of 1.29 is 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, NEDO-20360, Revision 1. November 1974.

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.

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 re- 19
quirement 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.

LIMITING CONDITIONS FOR OPERATION

19 3.12 Additional Safety Related Plant Capabilities

Applicability:

Applies to the operating status of the main control room ventilation system, the reactor building closed cooling water system and the service water system.

Objective:

To assure the availability of the main control room ventilation system, the reactor building closed cooling water system and the service water system upon the conditions for which the capability is an essential response to station abnormalities.

A. Main Control Room Ventilation

1. Except as specified in Specification 3.12.A.3 below, the control room air treatment system, the diesel generators required for operation of this system and the main control room air radiation monitor shall be operable at all times when containment integrity is required.

2.a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq$99% DOP removal and $\geq$99% halogenated hydrocarbon removal.

b. The results of laboratory carbon sample analysis shall show $\geq$90% radioactive methyl iodide removal at a velocity within 20% of system design, 0.05 to 0.15 mg/m^3 inlet iodide concentration, $\geq$95% R.H. and $\geq$125^{o}F.

c. Fans shall be shown to operate within $\pm$ 10% design flow.

SURVEILLANCE REQUIREMENTS

4.12 Additional Safety Related Plant Capabilities 19

Applicability:

Applies to the surveillance requirements for the main control room ventilation system, the reactor building closed cooling water system and the service water system which are required by the corresponding Limiting Conditions for Operation.

Objective:

To verify that operability or availability under conditions for which these capabilities are an essential response to station abnormalities.

A. Main Control Room Ventilation

1. At least once per operating cycle, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate.

2.a. The tests and sample analysis of Specification 3.12.A.2 shall be performed at least once per year for standby service or after every 720 hours of system peration and following significant painting, fire or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

LIMITING CONDITIONS FOR OPERATION

19 3.12.A (cont'd)

3. From and after the date that the control room air treatment system is made or found to be inoperable for any reason, reactor or refueling operations are permissible only during the succeeding seven days unless such circuit is sooner made operable.

4. If these conditions cannot be met, reactor shutdown shall be initiated and the reactor shall be in cold shutdown within 24 hours. If refueling operations are in progress, such operations shall be terminated in an orderly manner.

SURVEILLANCE REQUIREMENTS

4.12.A (cont'd) 19

2.d. Each circuit shall be operated at least 10 hours every month.

3. At least once per operating cycle automatic initiation of the system shall be demonstrated.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
B. Reactor Building Closed Cooling Water System (REC)	B. Reactor Building Closed Cooling Water System (REC)
1. Both reactor building closed cooling water loops and their associated pumps shall be operable whenever irradiated fuel is in the vessel or the spent fuel pool, except as specified in 3.12.B.2 and 3.12.B.3 below.	1. REC System Testing Item / Frequency a. Pump Operability — Once/Month b. Motor operated Valve Operability — Once/Month c. Pump flow rate Each pump shall deliver 1175 gpm at 65 psid. — Once/3 months and after pump maintenance d. System heat tank level shall be monitored. — Daily
2. From and after the date that any component in one loop becomes inoperable continued reactor operation is permissible during the succeeding thirty days provided that during such thirty days all the components of the other loop and the active components of the engineered safeguards compartment cooling systems, the diesel generator associated with the operable loop are operable. The allowable repair time does not apply when the reactor is in the shutdown mode and reactor pressure is less than 75 psig.	2. When it is determined that any active component in an REC loop as inoperable, all components in the other loop shall be demonstrated operable immediately and weekly thereafter.
3. Both reactor building closed cooling water loops with one pump per loop shall be operable as stated in 3.12.B.1 and 3.12.B.2 above during reactor head-off operations requiring LPCI or Core Spray System availability or service water cooling shall be available.	
4. If the requirements of 3.12.B.1 through 3.12.B.3 cannot be met, the reactor shall be shutdown in an orderly manner and in the Cold Shutdown condition within 24 hours or operations requiring LPCI or core spray system availability shall be halted.	

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LIMITING CONDITIONS FOR OPERATION

19 3.12 (cont'd)

C. Service Water System

1. Both service water subsystems with both pumps in each subsystem shall be operable when ever irradiated fuel is in the vessel or spent fuel pool and prior to reactor startup except as specified in 3.12.C.2 below.

2. From and after the date that any component in the service water subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding thirty days provided that during such thirty days all active components of the other service water subsystem and its associated diesel generator are operable.

3. If the requirement of 3.12.C.1 and 3.12.C.2 cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

D. Battery Room Ventilation

1. Battery room ventilation shall be operable on a continuous basis whenever specification 3.9.A is required to be satisfied.

2. From and after the date that either of the two battery room vent fans is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding 7 days.

3. if the requirements of 3.12.D.1 & 2 cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in Cold Shutdown within 24 hours.

SURVEILLANCE REQUIREMENTS

4.12 (cont'd) 19

C. Service Water System

1. Service Water System Testing

	Item	Functional
a.	Pump Operability	Once/Month
b.	Motor Operated Valve Operability	Once/Month
c.	Pump discharge head tests	Once/3 months

2. When it is determined that any service water system component is inoperable, the operable service water subsystem components shall be demonstrated to be operable immediately and weekly thereafter.

D. Battery Room Ventilation

1. The spare battery room ventilation fan shall be checked for operability once/week.

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19 3.12 BASES

A. Main Control Room Ventilation System

The control room ventilation system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The system is designed to automatically start upon control room isolation and to maintain the control room pressure to the design positive pressure so that all leakage should be out leakage.

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential intake of radioiodine to the control room. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

If the system is found to be inoperable, there is not immediate threat to the control room and reactor operation or refueling operation may continue for a limited period of time while repairs are being made. If the system cannot be repaired within seven days, the reactor is shutdown and brought to cold shutdown within 24 hours, or refueling operations are terminated.

B. Reactor Building Closed Cooling Water System

The reactor building closed cooling water system has two pumps and one heat exchanger in each of two loops. Each loop is capable of supplying the cooling requirements of the essential services following design accident conditions with only one pump in either loop.

The system has additional flexibility provided by the capability of interconnection of the two loops and the backup water supply to the critical loop by the service water system. This flexibility and the need for only one pump in one loop to meet the design accident requirements justifies the 30 day repair time during normal operation and the reduced requirements during head-off operations requiring the availability of LPCI or the core spray systems.

C. Service Water System

The service water system consists of four vertical service water pumps located in the intake structure, and associated strainers, piping, valving and instrumentation. The pumps discharge to a common header from which independent piping supplies two Seismic Class I cooling water loops and one turbine building loop. Automatic valving is provided to shutoff all supply to the turbine building loop on drop in header pressure thus assuring supply to the Seismic Class I loops each of which feeds one diesel generator, two RHR service water booster pumps, one control room basement fan coil unit and one RBCCW

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19 3.12 BASES (cont'd)

heat exchanger. Valves are included in the common discharge header to permit the Seismic Class I service water system to be operated as two independent loops. The heat exchangers are valved such that they can be individually backwashed without interrupting system operation.

During normal operation two or three pumps will be required. Three pumps are used for a normal shutdown.

The loss of all a-c power will trip all operating service water pumps. The automatic emergency diesel generator start system and emergency equipment starting sequence will then start one selected service water pump in 30-40 seconds. In the meantime, the drop in service water header pressure will close the turbine building cooling water isolation valve guaranteeing supply to the reactor building, the control room basement, and the diesel generators from the one service water pump.

Due to the redundance of pumps and the requirement of only one to meet the accident requirements, the 30 day repair time is justified.

D. Battery Room Ventilation

The temperature rise and hydrogen buildup in the battery rooms without adequate ventilation is such that continuous safe operation of equipment in these rooms cannot be assured.

4.12 BASES 19

A. Main Control Room Ventilation System

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per operating cycle to show system performance capability.

Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant should be performed in accordance with USAEC Report DP-1082.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N101.1-1972. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

Operation of the system for 10 hours every month will demonstrate operability of the filters and adsorber system and remove excessive moisture built up on the adsorber.

19 4.12 BASES (cont'd)

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

Demonstration of the automatic initiation capability is necessary to assure system performance capability.

B. Reactor Building Closed Cooling Water System

Normal plant operation requires one heat exchanger and three pumps. Therefore, normal equipment rotation will demonstrate pump operability.

Pump rates will be demonstrated every three months as an indication of the pump condition.

C. Service Water System

The service water pumps shall be proven operable by their use during normal station operations. Since three pumps are continuously operating during normal operation and only one pump is required during accidents, the normal equipment rotation shall prove the pump operability.

Pump discharge head tests will be run every three months to verify the pumping ability.

Any silting problems caused by the service water system will be analyzed during and following the Preoperational Test Program. Any required changes in operating procedures, technical specifications or surveillance requirements will be made prior to CNS commercial operation.

D. Battery Room Ventilation

The ventilation fans will be rotated on a weekly basis to demonstrate operability.

 19

LIMITING CONDITIONS FOR OPERATION

19 3.13 RIVER LEVEL

Applicability:

Applies to the status of the river level.

Objective:

To assure the plant is protected in case of flood levels exceeding 901.2 ft. MSL.

Specification:

A. If the river level reaches 895 feet MSL or is forecast to reach 902 feet MSL the CNS Site Flood Procedure will be implemented.

B. If the river level reaches 902 feet MSL or is forecast to reach 902 feet MSL, an orderly shutdown will be initiated and the reactor vessel vented to the atmosphere.

SURVEILLANCE REQUIREMENTS

4.13 RIVER LEVEL 19

Applicability:

Applies to the surveillance requirements associated with river level.

Specification:

The river water level shall be visually observed and logged once per shift at the intake structure.

19 3.13 and 4.13 BASES

The river level of the Missouri River is controlled by the U.S. Army Corps of Engineers using dams. The closest upstream dam is approximately 350 miles upstream. The use of these dams reduces the possibility of a site flood. However, should a dam break or in case of a river level of 895 ft. MSL, the CNS Site Flood Procedure will be put into effect.

Should the level reach 902 ft. MSL or information indicates that it is reasonable to expect levels of 902 ft. MSL or greater, the reactor will be shutdown and vented.

REFERENCES

1. FSAR Question 2.34, Amendment 17 to the FSAR.

 19

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
NEBRASKA PUBLIC POWER DISTRICT	)	Docket No. 50-298
	)	
(COOPER NUCLEAR STATION)	)	

ORDER FOR MODIFICATION OF LICENSE

I.

Nebraska Public Power District (the licensee) is the holder of Facility Operating License No. DPR-46 which authorizes operation of the Cooper Nuclear Station (the facility) at steady-state reactor power levels not in excess of 2381 megawatts thermal (rated power). The facility is a boiling water reactor (BWR) located near Brownville, Nemaha County, Nebraska.

II.

1. On July 22, 1975, the Nuclear Regulatory Commission (the Commission) issued an "Order for Modification of License" (40 F.R. 31837, July 29, 1975) which confirmed a plan for limited additional operation of the facility. As detailed in the Order, the facility's channel box wear, as indicated by the noise-to-signal ratio recorded by the traversing incore probe (TIP), had exceeded the remedial action threshold. The remedial plan confirmed by the Order limited operation of the facility to not more than 40% of rated core flow rate and with a maximum fuel bundle power of 3.20 MWt. In addition, the Order permitted operation up to full flow and power for a brief period of time as necessary to collect flow vibration and water quality data. The Order further stipulated that the

licensee was to shutdown the facility following approximately 45 equivalent full flow days from April 26, 1975 unless within that period certain specified tests have been completed which demonstrated the efficacy of the 40% flow limit.

2. On August 1, 1975, the Commission issued an "Order for Modification of License" (40 F.R. 33739, August 11, 1975) which clarified the intent of the July 22, 1975 Order to explicitly allow continued operation for the balance of a period of 50 effective full flow days prior to demonstrating the efficacy of the 40% flow limit. The bases for this action were schedules for delivery of equipment needed for accelerometer tests and the desirability from a power demand standpoint to schedule the installation of the equipment during a weekend or other low load period. The Commission's staff, in its August 1, 1975 evaluation of the request, concluded that the recently obtained TIP traces did not show any accelerated channel box wear, and that operation of Cooper for the balance of a period of 50 effective full flow days prior to demonstrating the efficacy of the 40% flow limit was acceptable since no appreciable additional wear would be incurred.

3. By letter dated September 11, 1975, the licensee proposed a plan, previously discussed with the NRC staff, setting forth a course of remedial action, which would allow operation with flow rates above 40 percent of rated flow and maximum bundle power above 3.20 MWt. The plan would involve shutdown of the reactor and appropriate replacement of worn channel boxes and plugging of the core support plate bypass holes. The

reactor was shutdown on September 27, 1975, for visual inspection of the channel boxes and the necessary repairs.

4. By letter dated September 11, 1975, the licensee provided details (by reference to previous staff evaluations for the Vermont Yankee and Duane Arnold plants) relating to the installation of core bypass flow plugs in the lower core support plate and supplied analyses to demonstrate the adequacy of such plugs to minimize future channel box wear and the adequacy of the procedures for plug installation.

5. On October 8, 1975, the Commission issued an "Order for Modification of License" (40 F.R. 48554, October 16, 1975) that approved the repair program and authorized the installation of bypass hole plugs in the facility's lower core plate. As discussed in the October 8, 1975 Order, the NRC staff concluded that the plugs will reduce the vibration of the instrument thimbles caused by flow through the bypass holes. By telecon on October 29, 1975, Nebraska Public Power District confirmed that the licensee's inspection and repair program was completed. The inspection program resulted in the rejection of 125 channel boxes with unacceptable wear as defined in the repair program. These channel boxes were replaced. Sixty-six channel boxes with indications of wear, but within the criteria of the repair program, were reinstalled in the reactor in locations which are not adjacent to instrument channels. Nebraska Public Power District also confirmed that all flow bypass holes in the core plate were plugged.

6. By letters dated September 11, October 7, 17 and 24, 1975, the licensee provided analyses, including an emergency core cooling performance analysis, for reactor power operation with the plugs installed in the bypass holes. The October 7, 1975 letter supplemented letters of July 10, July 14 and September 12, 1975 related to ECCS analyses.

7. The Commission's staff has reviewed the analyses submitted by the licensee on September 11, 1975 and October 7, 1975 and supplements thereto to support operation with the bypass holes flow plugs installed. As discussed in the Commission's concurrently issued Safety Evaluation for Amendment No. 16 to the license the proposed operation with plugs will require modified limits relating to emergency core cooling system performance. The modified limits specified in the concurrently issued Amendment No. 16 would be based upon an evaluation of ECCS performance calculated in accordance with an acceptable evaluation model that conforms to the requirements of the Commission's regulations in 10 CFR Section 50.46. The amendment would modify various limits established in accordance with the Commission's Interim Acceptance Criteria, and would, with respect to the facility, terminate the further restrictions imposed by the Commission's December 27, 1974 Order for Modification of License (40 F.R. 1767, January 9, 1975), and would impose instead, limitations established in accordance with the Commission's Acceptance Criteria for Emergency Core Cooling

Systems for Light Water Nuclear Power Reactors, 10 CFR Section 50.46. The amendment would also revise the Technical Specifications to permit operation of the facility using operating limits based on the General Electric Thermal Analysis Basis (GETAB) in accordance with the licensee's application for license amendment dated July 14, 1975 as supplemented.

It also should be noted that plugs identical to those to be used in the Cooper reactor have been installed in the Vermont Yankee, Duane Arnold and Pilgrim reactors. The plugs installed in Vermont Yankee were removed during a refueling operation after ten months of successful service. No abnormalities or loose pieces were reported. Vermont Yankee has since reinstalled the plugs.

8. Based on our review of the licensee's submittals of September 11, October 7, 17 and 24, 1975, and the prior related experience at the Pilgrim and Vermont Yankee reactors, the NRC staff concluded in its concurrently issued Safety Evaluation that operation of the Cooper reactor in accordance with the additional restrictions set forth in Amendment No. 16 to the License would provide reasonable assurance that the public health and safety would not be endangered.

III.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's Rules and Regulations in 10 CFR Parts 2 and 50, IT IS ORDERED THAT Facility Operating License No. DPR-46 is hereby amended by substituting the following provisions for the provisions set out in the Commission's Orders for Modification of License dated December 27, 1974 and October 8, 1975:

1. Operation of the Cooper Nuclear Station with plugged bypass flow holes is hereby authorized subject to the conditions set forth in the concurrently issued Amendment No. 16 to the Facility License No. DPR-46 incorporating Change No. 19 to the Technical Specifications, and

2. A monitoring program using LPRM and TIP traces and available accelerometers on incore instrument guide tubes shall be performed for the purpose of detecting any instrument tube - channel box interaction.

FOR THE NUCLEAR REGULATORY COMMISSION

Ben C. Rusche, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland,
this 31st day of October, 1975.

III.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's Rules and Regulations in 10 CFR Parts 2 and 50, IT IS ORDERED THAT Facility Operating License No. DPR-46 is hereby amended by substituting the following provisions for the provisions set out in the Commission's Orders for Modification of License dated December 27, 1974 and October 8, 1975:

1. Operation of the Cooper Nuclear Station with plugged bypass flow holes is hereby authorized subject to the conditions set forth in the concurrently issued Amendment No. 16 to the Facility License No. DPR-46 incorporating Change No. 19 to the Technical Specifications, and

2. A monitoring program using LPRM and TIP traces and available accelerometers on incore instrument guide tubes shall be performed.for the purpose of detecting any instrument tube - channel box interaction.

FOR THE NUCLEAR REGULATORY COMMISSION

original signed by
Ben C. Rusche

Ben C. Rusche, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland,
this OCT 31 1975

(FOR CONCURRENCES IN LETTER AND FEDERAL REGISTER NOTICE, SEE LETTER OF TRANSMITTAL - CONCURRENCE TABE INADVERTANTLY REMOVED)

RL:ORB #2
RMDiggs

OFFICE➤	RDSilver	RL:AD/ORs	OELD	RL:A/DIR	NRR:D/DIR	NRR:DIR
SURNAME➤	DLZiemann	KRGoller		RSBoyd	EGCase	BCRusche
DATE➤	10/ /75	10/ /75	10/ /75	10/ /75	10/ /75	10/ /75

NEGATIVE DECLARATION

REGARDING PROPOSED CHANGES TO THE

TECHNICAL SPECIFICATIONS OF LICENSE DPR-46

COOPER NUCLEAR STATION

DOCKET NO. 50-298

The Nuclear Regulatory Commission (the Commission) has considered the issuance of changes to the Technical Specifications of Facility Operating License No. DPR-46. These changes would authorize the Nebraska Public Power District (the licensee) to operate the Cooper Nuclear Station (located in Nemaha County Nebraska) with changes to limiting conditions for operation resulting from application of the Acceptance Criteria for Emergency Core Cooling System (ECCS). This action is associated with a planned shutdown to effect needed alterations associated with the reactor coolant system.

The U. S. Nuclear Regulatory Commission, Division of Reactor Licensing has prepared an environmental impact appraisal for the proposed changes to the Technical Specifications of License No. DPR-46, Cooper Nuclear Station, described above. On the basis of this appraisal, the Commission has concluded that an environmental impact statement for the particular action is not warranted because there will be no environmental impact attributable to the proposed action other than that which has already been predicted and described in the Commission's Final Environmental Statement for Cooper Nuclear Station issued in February 1973. The environmental impact appraisal is 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.

Dated at Rockville, Maryland, this 29th day of October 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Wm. H. Regan, Jr., Chief
Environmental Projects Branch 4
Division of Reactor Licensing

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

ENVIRONMENTAL IMPACT APPRAISAL BY THE DIVISION OF REACTOR LICENSING

SUPPORTING AMENDMENT NO. 16 TO DPR-46

CHANGE NO. 19 TO THE TECHNICAL SPECIFICATIONS

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

ENVIRONMENTAL IMPACT APPRAISAL

1. Description of Proposed Action

By letter dated July 10, 1975, Nebraska Public Power District submitted proposed changes to the Technical Specifications, Appendix A to License DPR-46. Supplementary information was provided by letters dated July 14, September 12, October 7, October 17, and October 24, 1975. The proposed changes were requested to incorporate limiting conditions for operation resulting from the application of the Acceptance Criteria for the Emergency Core Cooling System (ECCS) in association with a shutdown to plug bypass flow holes in the core support plate.

Nebraska Public Power District is presently licensed to operate the Cooper Nuclear Station, located in the State of Nebraska, Nemaha County, at power levels up to 2381 megawatt thermal (MWt). The proposed change to incorporate the ECCS Acceptance Criteria results in an estimated 15% decrease in full power level of the unit for approximately 6 months. The restrictions on heat generation rates will require careful control of power distribution in the core. There should be no significant reduction in total burnup resulting from the revised ECCS evaluation methods.

2. Environmental Impacts of Proposed Action

Potential environmental impacts associated with the proposed action are those which might be associated with incorporation of the ECCS Acceptance Criteria and utilization of nuclear fuel for this Facility.

It is particularly noted that, in the absence of any significant long term change in power levels, there will be no change in cooling water requirements, and consequently no increase in environmental impact from radioactive effluents and thermal effluents for normal operating or post-accident conditions which in turn could not lead to significant increase in radiation doses or thermal stress to the public or to biota in the environment.

For normal operating conditions, no environmental impact other than as described in the Commission's Final Environmental Statement (FES) for Cooper Nuclear Station, Docket No. 50-298, dated February 1973, can be predicted for the proposed action. The Commission's calculated releases of radioactive effluents, both gaseous and liquid, are based on expected release rates to the environment and are quantified on the basis of the total quantity of nuclear fuel within the reactor. The estimates of radionuclides and release rates will not be affected by the proposed action, and since the total quantity of nuclear fuel is unchanged, no increase in the calculated release of radioactive effluents is predicted. Consequently, no increase in radiation doses to man or other biota are predicted.

3. Conclusion and Basis for Negative Declaration

On the basis of the foregoing analysis, it is concluded that there will be no environmental impact attributable to the proposed action other than that already predicted and described in the Commission's FES for Cooper Nuclear Station. Having made this conclusion, the Commission has further concluded that no environmental impact statement for the proposed action need be prepared and that a negative declaration to the effect is appropriate.

Date: October 29, 1975

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 16 TO FACILITY OPERATING LICENSE NO. DPR-46

(CHANGE NO. 19 TO THE TECHNICAL SPECIFICATIONS)
AND
ORDER FOR MODIFICATION OF LICENSE

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

Nebraska Public Power District has proposed to operate Cooper Nuclear Station for the remainder of this fuel cycle under the following conditions:

(1) with plugged bypass flow holes in the lower core support plate as requested in their submittal dated September 11, 1975 and supplements dated October 7, 17 and 24, 1975,

(2) using limits based on the General Electric Thermal Analysis Basis (GETAB) as requested in their submittal dated July 14, 1975 and supplements dated September 12, October 7, 17 and 24, 1975, and

(3) using modified operating limits based on an acceptable evaluation model that conforms with Section 50.46 of 10 CFR Part 50 as requested in their submittal dated July 10, 1975 and supplements dated July 14, September 12, October 7, 17 and 24, 1975.

2.0 NUCLEAR DESIGN

The primary nuclear effect caused by plugging the bypass flow holes is an increased bypass void fraction and a reduction in the average in-channel void fraction. The in- and out-of-channel void fraction changes give a net increase in the core average void fraction.

At steady state conditions, the increased bypass void fraction results in a small reduction in the maximum local peaking factor within a fuel bundle and an increase in the local bundle power calculational uncertainty. Another consequence of the reduced bypass flow is a small reduction in the infinite multiplication factor of uncontrolled fuel.

The presence of voids in the bypass region affects the relationship between the travelling incore probe (TIP) signal and the local bundle power. The TIP signal is reduced by the presence of voids and could lead to an underprediction of the peak heat flux. The relationship of the power in the four bundles surrounding a TIP instrument tube and the TIP signal as a function of bypass voids was determined by the General Electric Company (GE) by performing three group, two-dimensional diffusion theory calculations. A correction factor was developed and algorithms for computing the bypass void fraction and for making appropriate corrections in the local bundle power have been incorporated in the process computer.

The uncertainty in the local bundle power caused by bypass voids is taken into account in determining the minimum critical power ratio (MCPR) safety limit. The TIP uncertainty introduced by the bypass voids is zero in the bottom half of the core and increases from 4.12% at the core mid-plane to 5.41% at the core exit.

After the bypass flow holes are plugged, the fuel will be placed in its original core location. The following observations can be made:

(1) the control rod worths are not significantly changed and, consequently, the previous results of the control rod drop analysis remain valid,

(2) the shutdown margin will remain the same as previously analyzed,

(3) the standby liquid control system reactivity insertion rate and magnitude will not be affected.

We conclude that the analysis of the nuclear performance of the plant with plugged bypass holes is acceptable.

3.0 MECHANICAL DESIGN

The only mechanical design change in the reactor is the use of plugs to fill the bypass flow holes[1]. The plug consists of two stainless steel parts (body and shaft) which are connected by an Inconel spring. The shoulder of the body rests on the top of the core plate along the rim of a one-inch bypass hole and is pressed down by the spring. An equal and opposite force is applied on the shaft. A stainless steel latch is connected to the bottom of the shaft by means of a pin. This latch is free to rotate about the pin and latches the shaft to the core plate. The spring exerts a minimum of 35 pounds on the body and latch and a maximum of 46 pounds (with the worst tolerance combination).

Removal of a plug can be accomplished by applying about 500 pounds of force and deforming the latch plastically. More than 10 plugs were removed in tests performed at the GE test facility with consistent latch deformations without damaging other parts.

Plugs identical to those to be used in the Cooper reactor have been installed in the Vermont Yankee, Duane Arnold and Pilgrim reactors. The plugs installed in Vermont Yankee were removed during a refueling operation after 10 months of successful service. No abnormalities or loose pieces were reported. Vermont Yankee has since reinstalled the plugs.

Pressure differentials across the core plate during normal steady state operation and following a steam line break accident are expected to be on the order of 20 to 32 psi. These loads together with the spring preload will produce yielding of the latch in bending but will be significantly below about 500 pounds of force necessary for removing the plug. The 1973 GE full scale flow mockup test shows that, with up to 40 psi differential pressure, there is negligible leakage flow through the plugged holes. No plug vibration was

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

observed during the test and no apparent deformation on the latch was evident after the test. No fatigue and plastic strain ratcheting is expected since the plant power cycle during the anticipated service period will be minimal.

Stainless steel and Inconel are compatible with other reactor internals and are not expected to introduce any unusual oxidation and stress corrosion problems. The flux level at the core plate elevation is estimated to be quite low and an insignificant reduction in ductility due to irradiation is anticipated. GE has performed creep tests with both Inconel springs and stainless steel latches and found that stress relaxation or creep deformation were insignificant. The tests were performed at 550°F.

Nebraska Public Power District presented to the NRC staff a summary of channel inspections on BWR-2s and BWR-3s. These older plants have instrument tubes similar to Cooper, but no bypass flow holes in the core support plate. The bypass flow for these plants enters through clearances in the assembly end fittings, which is similar to the proposed Cooper configuration with plugged bypass holes. One hundred sixty-four channels (adjacent to instrument tubes and source tubes) were inspected during normal fuel outages in 7 plants. No significant channel wear was observed at the corners adjacent to the instrument tubes.

General Electric has a design criteria for channel box wastage of 0.010 inches for the lower 80 inches of the channel and 0.020 inches for the remaining length. All of the channels (new and old) in the core will meet this requirement. Channels with observed acceptable wear on the corner will not be reinserted in the core next to an in-core instrument where additional wear could occur during subsequent reactor operation.

Based on a review of the design, the test rig, the installation methods and primarily the previously successful operating experience at Vermont Yankee and Pilgrim, we conclude that the plugs will not fail so as to result in loose parts in the core or result in unplugging of the bypass flow holes. Also, we conclude that the installed plugs will substantially reduce the instrument tube vibration, due to flow through the bypass holes, sufficient to preclude any unacceptable wear for at least the proposed fuel cycle.

4.0 THERMAL HYDRAULIC DESIGN

The licensee's letter of July 14, 1975, and supplements dated September 12, October 7, 17 and 24, 1975, requested that the license for Cooper be amended to include operating limits based on the General Electric Thermal Analysis Basis (GETAB) described in the General Electric report NEDO-10958[(2)]. The analyses are based on a core loading with 7 x 7 fuel and with the bypass flow holes plugged.

The proposed changes involve the adoption of a new transition boiling correlation termed GEXL which would replace the Hench-Levy critical heat flux correlation as the basis for determining the thermal-hydraulic conditions which would result in a departure from nucleate boiling. One of the safety requirements for light water cooled nuclear reactors is prevention of damage to the fuel cladding. To prevent damage to the fuel cladding, light water cooled reactors must be designed and operated such that during normal operation and anticipated transients the heat transfer rate from the fuel cladding to the coolant are sufficient to prevent overheating of the fuel cladding. Although transition boiling would not necessarily result in damage to boiling water reactors (BWR) fuel rods, historically it has been used as a fuel damage limit because of the large reduction in heat transfer rate when film boiling occurs. A critical power ratio (CPR) is defined which is the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest. The MCPR is the critical power ratio corresponding to the most limiting fuel assembly in the core. The fuel assembly power at which boiling transition would be predicted to occur, using the GEXL correlation, is termed the critical power. The GEXL transition boiling correlation is more recent than the previously used Hench-Levy critical heat flux correlation and is based on an extensive data base. The methods for applying the GEXL correlation to determine thermal limits has

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(2) "General Electric BWR Thermal Analysis Basis (GETAB) Data Correlation and Design Application," NEDO-10958 and NEDE-10958 (Proprietary), November 1973.

been termed the General Electric Thermal Analysis Basis (GETAB). We have accepted the GEXL correlation and the GETAB methods in a previous report [3] as a basis for establishing the safety limit and limiting conditions for operation related to prevention of fuel damage for General Electric BWR 8 x 8 and 7 x 7 fuel. To apply GETAB to the Technical Specifications involves establishing (1) the fuel damage safety limit, (2) limiting conditions of operation such that the safety limit is not exceeded for normal operation and anticipated transients, and (3) limiting conditions for operation such that the initial conditions assumed in accident analyses are satisfied. We have evaluated the thermal margins for the Cooper Nuclear Station based on the NEDO-10958 report [2] and plant specific input information provided by the licensee. As described below, we conclude that the calculated consequences of the anticipated abnormal transients do not violate the thermal and plastic strain limits of the fuel.

4.1 FUEL CLADDING INTEGRITY SAFETY LIMIT-MCPR

The safety limit on MCPR is based on the GETAB statistical analysis which assures that more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The uncertainties in the core and system operating parameters and the GEXL correlation, Table 5-1 of the licensee submittal, [1] combined with the relative bundle power distribution in the core form the basis for the GETAB statistical determination of the safety limit MCPR. These uncertainties are the same as or more conservative than those reported in NEDO-10958 [2] and NEDO-20340 [4] with one exception. The

(3) "Review and Evaluation of GETAB (General Electric Thermal Analysis Basis) for BWRs," Division of Technical Review, Directorate of Licensing, United States Atomic Energy Commission, September, 1974.

(4) General Electric, "Process Computer Performance Evaluation Accuracy," NEDO-20340, and Amendment 1, NEDO-20340-1, dated June, 1974 and December, 1974.

exception is the uncertainty of the bypass void effect on TIP which accounts for the additional uncertainty due to the bypass void content resulting from plugging the core support plate bypass holes. The reactor core selected for the GETAB statistical analyses is a typical core (251" diameter vessel/764 fuel assemblies). This typical core is of the same reactor class as the Cooper core (218/548) but it is larger. The bundle power distribution used for the GETAB application has more high power bundles than the distribution expected during operation of the Cooper reactor. This results in a conservative value of the MCPR which meets the 99.9% criterion.

We conclude that the proposed fuel integrity safety limit, a MCPR of 1.06, is acceptable for the Cooper current fuel cycle with plugged bypass holes.

4.2 LIMITING CONDITION FOR OPERATION - MCPR

Various transient events will reduce the required operating limit 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 (ΔMCPR). The licensee has submitted the results of those transient analyses which show a significant decrease in MCPR. The types of transients evaluated were losses of flow, pressure and power increases, and coolant temperature decreases. The main factors affecting the plant transient analyses are the moderator void coefficient of reactivity, the Doppler coefficient of reactivity, and the full power scram reactivity function. The Doppler coefficient of reactivity is affected by the changes in the moderator density in the fuel channel and bypass region primarily through changes in the Dancoff-Ginsburg rod shadowing effect. This effect is small and insignificantly affects the Doppler coefficient of reactivity. The full power scram reactivity function for the end-of-cycle with plugged bypass flow holes indicates a total scram worth of -35.26 dollars. This is less total scram worth than the previously determined value and is due only to a recalculation of the Cooper end-of-cycle reactivity and not to any effects caused by changed void distributions. However, the initial scram reactivity addition rate which is important to transient analyses, is about the same as previously used.

The moderator void coefficient of reactivity used in the safety analysis of the Cooper plant with plugged bypass flow holes is more negative than used in the FSAR for two reasons. The first cause is a renormalization of the void coefficient calculations based on analyses of operating BWR data. This effect, of the order of 15 to 20 percent, is unrelated to the plugging of the bypass flow holes. The second cause is the increase in the amount of voids present in the bypass region after the bypass flow holes are plugged. The most limiting transient is a turbine trip with failure of bypass valves to open. The analyses were initiated from 105.1 percent of design power and the scram was initiated by the position switch on the turbine stop valves. The decrease in MCPR is 0.23 which is the limiting change in thermal margin. As a result, the steady state MCPR must be equal to or greater than 1.29 to satisfy the safety limit MCPR of 1.06. The calculated change in MCPR for the second most severe transient, loss of feedwater heating, is 0.15.

We conclude that the proposed MCPR value of 1.29, the limiting condition for operation, is acceptable for the Cooper current fuel cycle with plugged bypass holes.

4.3 LIMITING CONDITIONS FOR OPERATION, MCPR, AT LESS THAN RATED POWER AND FLOW

The limiting transient at less than rated power and flow condition is the recirculation pump speed control failure. The Technical Specifications would require the licensee to maintain MCPR greater than 1.29 times the K_f factor for core flows less than rated. The K_f factor curves were generically derived and assure that most limiting transient, a speed control increase, occurring at less than rated flow will not exceed the safety limit MCPR of 1.06. We conclude that the limiting conditions for operation, MCPR, at less than rated power and flow are acceptable.

5.0 OVERPRESSURE TRANSIENTS

The licensee has reanalyzed the worst case overpressure transient for operation with the bypass flow holes in the lower core support plate plugged. The transient analyzed was the closure of all main steam isolation valves with a high neutron flux scram. The assumptions used in the analysis were: Operation at 105% of power, end-of-cycle scram reactivity insertion rate curve and one safety/relief valve fails to operate. The reanalysis predicts a peak pressure at the vessel bottom of 1314 psig which is 61 psi below the code allowable pressure. The reanalysis and calculated pressure margin are acceptable.

6.0 EMERGENCY CORE COOLING SYSTEMS ANALYSIS

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License implementing the requirements of Section 50.46 of 10 CFR Part 50 of the Commission's Regulations "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that prior to any license amendment authorizing any core reloading "... the licensee shall submit a reevaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms with the provisions of Section 50.46." The order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendments as may be necessary to implement the evaluation results.

On July 10 and 14, 1975 the licensee submitted an evaluation of the ECCS performance for the design basis pipe break for Cooper along with an amendment requesting changes to the Technical Specifications for Cooper to implement the results of the evaluation.[5] [6] The licensee incorporated further information relating to the details of the ECCS evaluation, by referencing an appropriate lead plant analysis, [7] to show compliance with the Section 50.46 criteria and Appendix K to 10 CFR Part 50.

The Order for Modification of License issued December 27, 1974, stated that evaluation of ECCS cooling performance may be based on the vendor's evaluation model as modified in accordance with the changes described in the staff Safety Evaluation Report of the Cooper Nuclear Station dated December 27, 1974.

(5) Letter from J. Pilant, Nebraska Public Power District, to B. Rusche, USNRC, dated July 10, 1975.

(6) Letter from J. Pilant, Nebraska Public Power District to B. Rusche, USNRC, dated July 14, 1975.

(7) Quad Cities Station Special Report No. 15, Supplement C, Unit 2 and Attachment A (Proprietary).

The background of the staff review of the GE ECCS models and their application to Cooper is described in the Staff Safety Evaluation Report (SER) for these facilities dated December 27, 1974 issued in connection with the Order. The bases for acceptance of the principal portions of the evaluation model are set forth in the staff's Status Report of October, 1974 which are referenced in the December 27, 1974 SER. The December 27, 1974 SER and the Status Report and its Supplement describe an acceptable ECCS evaluation model and the basis for the staff's acceptance of the model. The Cooper evaluation which is covered by this SER properly conforms to the accepted model.

With respect to reflood and refill computations, the Cooper analysis was based on a modified version of the SAFE computer code, with explicit consideration of the staff recommended limitations, as described in the December 27, 1974 SER. The Cooper evaluation did not attempt to include any further credit for other potential changes which the December 27, 1974 SER indicated were under consideration by GE at that time.

During the course of our review, we concluded that additional individual break size should be analyzed to substantiate the break spectrum curves submitted in connection with the evaluation provided in August, 1974. We also requested that other break locations be studied to substantiate that the limiting break location was the recirculation line.

The additional analyses (performed on the lead plant, Quad Cities Unit No. 2 [7] and incorporated by reference) supported the earlier submittal which concluded that the worst break was the complete severence of the recirculation line. These additional calculations provided further details with regard to the limiting location and size of break as well as the worst single failure for the Cooper design. The limiting break continues to be the complete severence of the recirculation suction line assuming a failure of the LPCI injection valve.

The October 7, 1975 submittal contains the ECCS analysis for operation with the plugged bypass flow holes. The results for this ECCS Appendix K calculation show a peak cladding temperature of 2200° F; a peak local oxidation of 9.2%, and a maximum core average hydrogen generation of 0.5% for the worst large size pipe break assuming failure of the LPCI injection valve (the worst single failure). The calculations show a peak cladding temperature of 1210° F, and a peak local oxidation of less than 1.0% for the worst small size pipe break area (0.10 ft^2) assuming failure of the HPCI system (the worst concurrent single failure). We have reviewed the evaluation of ECCS performance submitted by NPPD for the Cooper plant with plugged bypass holes and conclude that the evaluation has been performed wholly in conformance with the requirements of Section 50.46. Therefore, operation of the reactor would meet the requirements of Section 50.46 provided that operation is limited to the maximum planar linear heat generation rates (MAPLHGR) of figures 8-12A, and 8-12B of the NPPD submittal dated October 7, 1975, and to a minimum critical power ratio (MCPR) greater than 1.18. The ECCS performance analysis assumed that reactor operation will be limited to a MCPR of 1.18. However, a more restrictive technical specification limits operation of the reactor to a MCPR of 1.29 based on consideration of a turbine trip transient with failure of bypass valves. A statement should be added to the bases for the limiting condition of operation indicating the MCPR used in the ECCS performance evaluation.

Certain operating conditions presently allowed are not in conformance with the analysis performed in accordance with Section 50.46. Certain changes must be made to the proposed technical specifications to conform with the evaluation of ECCS performance. An evaluation was not provided for ECCS performance during reactor operation with one recirculation loop out of service. Therefore, continuous reactor operation under such conditions should not be permitted until the necessary analyses have been performed, evaluated and determined acceptable. The reactor may, however, operate for periods up to 24 hours with one recirculation loop out-of-service. This short time period permits corrective action to be taken and minimizes unnecessary shutdowns which is consistent with other Technical Specifications. During this period of time the reactor will be operated within the restrictions of the thermal analysis and will be protected from fuel damage resulting from anticipated transients.

The LOCA analysis assumed all ADS valves operated for small line breaks with HPCI failure. Since the licensee did not provide a LOCA analysis with one ADS valve out of service for small size line breaks the Technical Specifications will not permit continuous operation with any ADS valve out of service except as with other ECCS equipment one valve may be out of service for seven days.

Based on the above, we conclude that with the Technical Specifications discussed above operation of the reactor will meet the requirements of Section 50.46 of 10 CFR Part 50 of the Commission's Regulations.

7.0 INSPECTION PROGRAM

During the October, 1975 outage Nebraska Public Power District performed an inspection of all the Cooper Station fuel bundle channel boxes from locations adjacent to in-core instrument tubes. The fuel rods in bundles with cracked channels and in-core instrument tubes adjacent to channels that exhibited cracking or high wear were inspected. The inspection revealed 125 channel boxes with an unacceptable amount of wear. These channel boxes were replaced. Sixty-six channel boxes were worn less than the amount established for replacement. These channel boxes were reinstalled in the reactor in locations which are not adjacent to instrument channels. Four of the rejected channel boxes were perforated. In three of the perforated channels, holes developed as a result of crack propagation. Inspection of these channels indicated five pieces of channel material broke off. A search was conducted and four of the five pieces were found and removed from the reactor vessel. The fifth piece was not located. However, based on the location of the pieces found, it is considered likely that the piece is resting on the core support plate. The staff evaluated the possible effects which could be caused by the small piece. The major concern associated with a small loose piece is the potential for fuel bundle flow blockage. We have concluded that it is unlikely that the piece would be carried through the primary system to a fuel bundle coolant flow orifice. However, even if the piece were transported through the system and remained a single flat piece, and even if it were arranged against an oriface to cause the maximum flow blockage, no significant fuel damage would occur. Based on our review, we have concluded that the small missing piece is not a safety concern.

The fuel rods and bundle components of the bundles with perforated channels were inspected for fuel pin damage, improper spacing of rods and rod bowing. The four bundles were found to be undamaged and were returned to the core following rechanneling. Ten core instrument tubes adjacent to damaged channels were inspected. Eight were found acceptable. One was rejected because of high wear and one was rejected because of unusual markings seen on the tube. The two rejected tubes were replaced with new tubes. Eleven additional in-core instruments were replaced because of sensor or electrical problems. Based on the results of the inspection and replacement program we have concluded that the condition of the installed channel boxes, in-core instrument tubes and fuel in the bundles subjected to channel box perforation are acceptable.

Subsection 1.1.B. would be revised to limit core thermal power to 25% or less of rated thermal power when reactor pressure is less than or equal to 800 psig or core flow is less than 10% of rated. These changes are consistent with the GETAB analyses discussed earlier in this safety evaluation.

Section 3.3.B.6. Control Rods Limiting Condition for Operation

The existing specification 3.3.B.6.c. would be revised from a MCHFR limitation to a MCPR limitation so that the specification would be consistent with the GETAB analysis.

Section 3.5.E. Automatic Pressure Relief Subsystem Limiting Condition for Operation

The existing specification allows continued operation for up to 30 days after one relief valve of the automatic depressurization system (ADS) is made or found to be inoperable. The loss-of-coolant accident analyses submitted in accordance with 10 CFR 50.46 were based on the assumption that all ADS valves operated for small line breaks with HPCI failure. Because the analyses submitted do not support extended periods of operation with one ADS valve out of service, we would reduce the time the valve can be out of service to 7 days. This is consistent with out of service times for other ECCS equipment. We have discussed this change with the NPPD staff and they did not object.

Section 3.6.F. Jet Pump Flow Mismatch

This section would be modified to limit operation to a period of twenty-four hours when one recirculation loop is out of service. The additional restriction would be consistent with the analysis discussed earlier in this report.

Section 3.11 Fuel Rods

A new section on thermal limits for fuel is being added which incorporates limits from another section of the specifications and revises limits to be consistent with the new thermal analyses.

Section 3.6.F. Jet Pump Flow Mismatch

This section would be modified to limit operation to a period of twenty-four hours when one recirculation loop is out of service. The additional restriction would be consistent with the analysis discussed earlier in this report.

Section 3.11 Fuel Rods

A new section on thermal limits for fuel is being added which incorporates limits from another section of the specifications and revises limits to be consistent with the new thermal analyses.

Section 3.11.A. Average Planar LHGR Limiting Conditions for Operation

The average planar linear heat generation limits would be revised to be consistent with the analyses performed in accordance with 10 CFR 50.46 for operation with plugged bypass holes in the lower core plate.

Section 3.11.C. Minimum Critical Power Ratio (MCPR)

Subsection 3.11.C. would be added to place operating MCPR limits on the fuel. The limits are consistent with the GETAB analyses discussed earlier in this report and require a MCPR more limiting than that needed to satisfy the requirements of the LOCA analysis.

GETAB Bases

The bases would also be changed to discuss the justification for the revised specifications itemized above. We would modify the proposed GETAB related bases to provide what we consider to be a clearer justification for the limits.

Proposed Reporting Exclusion

The specifications proposed by the licensee would exclude reporting, as an abnormal occurrence, operation in excess of the limiting MAPLHGR, local LHGR and MCPR values providing corrective action was taken upon discovery. We would not include these provisions. We believe that such events should be reported in conformity with the Technical Specifications.

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3	INITIALS	NOTE AND RETURN
	DATE	PER CONVERSATION
4	INITIALS	SEE ME
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REMARKS

The action was taken prior to completion of our review!! Consequently, we do not indicate a concurrence with the action taken.

However, we see no need to interpose a legal objection ~~to~~ requiring modification of the action taken.

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APRM Flux Trip, APRM Rod Block and RBM Rod Block Settings

The specifications proposed by the licensee would change the primary coolant flow referenced trip settings to provide greater operating margin. The licensee has not fully justified the proposed revisions. Therefore we would retain the existing flow referenced limits.

Instrument Tube-Channel Box Interaction Surveillance

Excessive instrument tube-channel interaction previously has been determined from the noise level in the LPRM signals. The plugged bypass flow holes are expected to affect the noise content of the LPRM signals. The noise content in the 1.4 to 3 Hz frequency range caused by vibration of the LPRM instrument tube should be reduced relative to the power dependent noise content. Some increase in the boiling noise, 5 to 50 Hz range, is expected because of boiling in the bypass water region.

Before the plant was shutdown in 1975, extensive LPRM time traces, TIP traces, and power spectral density (PSD) calculations were obtained for a number of combinations of power and flow. These data will provide a basis for evaluating the efficiency of plugging the bypass flow holes. After reactor startup, comparison of similar measurements with pre-shutdown data will be made to confirm that the mechanical vibration of the instrument tubes has been substantially reduced.

The licensee has agreed to provide NRC with a plan for monitoring instrument tube-channel box interaction. The monitoring would be performed on a periodic basis using the available LPRM and TIP traces and the available accelerometers on core instrument guide tubes. This monitoring program should be required by the licensee.

9.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: October 31, 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-298

NEBRASKA PUBLIC POWER DISTRICT

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE AND ORDER MODIFYING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 16 to Facility Operating License No. DPR-46, issued to the Nebraska Public Power District (the licensee), which revised 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: (1) using operating limits based on the General Electric Thermal Analysis Basis (GETAB), and (2) with modified operating limits based upon an evaluation of emergency core cooling system performance calculated in accordance with an acceptable evaluation model that conforms with the requirements of Section 50.46 of 10 CFR Part 50 of the Commission's regulations.

The Commission also has issued an Order for License Modification which authorizes operation of the facility with the bypass holes of the lower core support plate plugged, subject to the conditions set forth in the Technical Specifications, as revised by the issuance of Amendment No. 16. The Order is effective as of the date of its issuance.

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

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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 September 18, 1975 (40 F.R. 43099). The Order for Modification of License (dated October 8, 1975) which authorized plugging of the bypass holes was published in the FEDERAL REGISTER on October 16, 1975 (40 F.R. 48554). 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.

For further details with respect to this action, see (1) the applications for amendment dated July 10, 1975, July 14, 1975 and September 11, 1975 and supplements thereto dated September 12, 1975 and October 7, 17 and 24, 1975, (2) Amendment No. 16 to License No. DPR-46, with Change No. 19, (3) the concurrently issued Order for Modification of License, (4) the Commission's concurrently issued related Safety Evaluation , (5) the Commission's Negative Declaration dated October 29, 1975, (which is also being published in the FEDERAL REGISTER) and associated Environmental Impact Appraisal, and (6) the Order for Modification of License dated October 8, 1975 and the documents referenced therein. 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, 1118 - 15th Street, Auburn, Nebraska 68305. A single copy of items (2) through (5) and the October 8, 1975 Order may be obtained upon

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request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this OCT 31 1975

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Dennis L. Ziemann

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

RL:ORB #2
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SURNAME ➤	DLZiemann		KRGoller	RSBoyd	ECCase	BCRusche
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	DATE	PER CONVERSATION
4	INITIALS	SEE ME
	DATE	SIGNATURE

ACTION

REMARKS

The action was taken on Friday before we completed our review. Consequently, we have indicated no concurrence with the course of action. After considering the legal circumstances involved in these actions for Cooper, which are substantially different than those used in prior authorizations to operate with repaired channel boxes, we have concluded that there is no legal objection to this action.

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FROM (Name, office symbol or location)

Joe Scinto, ELD

DATE 11/4/75

PHONE

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

October 29, 1975

Docket No. 50-298

Dennis Ziemann, Chief, Operating Reactors Branch 2, DRL

NEGATIVE DECLARATION FOR TECHNICAL SPECIFICATION CHANGES TO COOPER NUCLEAR STATION

Attached is the Environmental Impact Appraisal and Negative Declaration associated with proposed changes in Technical Specifications appropriate to implementation of the ECCS Acceptance Criteria.

Wm. H. Regan, Jr., Chief
Environmental Projects Branch 4
Division of Reactor Licensing

Enclosures:
1. Environmental Impact Appraisal
2. Negative Declaration

Included w/ Amendment 16
Iss. 10/31/75