Nebraska Public Power District

GENERAL OFFICE
P. O. BOX 499, COLUMBUS, NEBRASKA 68601
TELEPHONE (402) 564-8561

January 31, 1979

Director, Nuclear Reactor Regulation
Attention: Mr. Thomas A. Ippolito, Chief
Operating Reactors Branch No. 3
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, DC 20555

THIS DOCUMENT CONTAINS
POOR QUALITY PAGES

Subject: Supplemental Reload Licensing Submittal and Proposed Technical Specifications for Cooper Nuclear Station Reload 4, Cycle 5, NRC Docket No. 50-298, DPR-46

Dear Mr. Ippolito:

In accordance with the provisions of the Atomic Energy Act of 1954, and the amendments thereto, the Nebraska Public Power District respectively requests that the operating license for Cooper Nuclear Station be revised to allow operation following the fourth refueling of the reactor.

Specifically, the results of the plant unique analysis are presented in the "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1, Reload 4", NEDO-24170, dated December 1978, and are submitted as an enclosure to this letter. This document has been prepared utilizing the format contained in Appendix A of General Electric Report NEDE-24011-P "Generic Reload Fuel Application". Also submitted herein (Attachment 1) are proposed changes to the Cooper Nuclear Station Technical Specifications. The proposed Technical Specification changes are as follows:

1. The Rod Block Monitor setpoint is being changed for 106% to 105% due to the results of the rod withdrawal error analysis.

2. The operating Minimum Critical Power Ratio (MCPR) limits and Overpressurization Analysis results are revised.

3. The densification power spiking penalty factor is being deleted for 8x8 fuel.

4. The low pressure main steam line isolation signal is being reduced from 850 psig to 825 psig. Justification for this change is enclosed as Attachment 2.

5. Various reference and editorial changes.

7902090350

6. Our letter of September 1, 1977 submitted the results of a reevaluation of ECCS performance at Cooper Nuclear Station; NEDO-24045 "Loss of Coolant Accident Analysis Report for Cooper Nuclear Power Station", August 1977. As stated in our letter, the results of the loss-of-coolant analysis indicated that the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) values were equal to or less restrictive than those currently contained in the Technical Specifications, but that the District did not intend to request relaxation of the MAPLHGR limits for Cycle 3 operation. Amendment 46 to the Operating License authorizing operation for Cycle 4 revised figures 3.11-1.1, 3.11-1.2, and 3.11-1.5 to incorporate the revised MAPLHGR values. Figures 3.11-1.3 and 3.11-1.4 were inadvertently overlooked and are being revised at this time based on NEDO-24045.

No changes are being made in the fuel design over those incorporated in Reload 3.

The schedule for this refueling outage currently indicates that the station will be shutdown beginning April 7, 1979 for a period of approximately four weeks. However, in the event the plant should sustain an unscheduled outage anytime after April 1, 1979, the refueling outage may begin at that time which would move the probable start-up date forward by one week. Therefore, Nebraska Public Power District respectfully requests approval of this license amendment prior to April 27, 1979.

These proposed changes have been reviewed under 10CFR170 and judged to be a Class III amendment. Payment in the amount of $4,000 is enclosed with this submittal. Additionally, the District's internal safety review committees have reviewed and approved the enclosed Cycle 5 licensing amendment.

Should you have any questions or comments regarding the enclosed, please do not hesitate to contact me.

In addition to three signed originals, 37 copies of the proposed changes are also submitted.

Sincerely yours,

Jay M. Pilant

Jay M. Pilant
Director of Licensing and
Quality Assurance

JMP/jdw:bas31/4
Enclosures

STATE OF NEBRASKA)
) ss
PLATTE COUNTY   )

Jay M. Pilant, being first duly sworn, deposes and says that he is an authorized representative of the Nebraska Public Power District, a public corporation and political subdivision of the State of Nebraska; that he is duly authorized to execute this request on behalf of Nebraska Public Power District; and that the statements in said application are true to the best of his knowledge and belief.

Jay M Pilant
____
Jay M. Pilant

Subscribed in my presence and sworn to before me this 31st day of January, 1979.

Marilyn R. Hohndorf
____
NOTARY PUBLIC

My Commission expires Oct. 14, 1980.

GENERAL NOTARY - State of Nebraska
MARILYN R. HOHNDORF
My Comm. Exp. Oct. 14, 1980

Proposed Changes to
Cooper Nuclear Station
Technical Specification Pages

9	61
12	84
13	102
17	104
18	210
19	211a
22	212
25	214b
26	214c
50	214d
52	214e

SAFETY LIMITS	LIMITING SAFETY SYSTEM SETTINGS

2.1.A (cont'd)

3. Turbine Stop Valve Closure Scram Trip Setting

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

4. Turbine Control Valve Fast Closure Scram Trip Setting

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

5. Main Steam Line Isolation Valve Closure Scram Trip Setting

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

6. Main Steam Line Isolation Valve Closure on Low Pressure

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

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

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

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

1.1 Bases: (Cont'd)

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

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

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

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

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

C. Power Transient

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

1.1 Bases: (Cont'd)

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

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

D. Reactor Water Level (Shutdown Condition)

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

References

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

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

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

2.1 Bases:

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

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

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

For analyses of the Thermal consequences of the transients, a MCPR of 1.20 for 7x7 fuel and 1.22 for 8x8 fuel is conservatively assumed to exist prior to initiation of the transients.

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

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

2.1 Bases: (Cont'd)

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

In summary:

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

ii. The licensed maximum power level is 2381 MWt.

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

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

A. Trip Settings

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

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

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

2.1 Bases: (Cont'd)

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

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of maximum fraction of limiting power density (MFLPD) and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.a.1.a, when the MFLPD is greater than the fraction of rated power (FRP). This adjustment may be accomplished by increasing the APRM gain and thus reducing the slope and intercept point of the flow referenced APRM High Flux Scram Curve by the reciprocal of the APRM gain change.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.07 when the transient is initiated from MCPR > 1.23 for 7x7 bundles, and 1.23 for 8x8 bundles.

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

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accomodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedure backed up by the rod worth minimizer, and the rod sequences control system. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This change can occur when reactor pressure is greater than Specification 2.1.A.6.

2.1 Bases: (Cont'd)

5. Main Steam Line Isolation Valve Closure on Low Pressure

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

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

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

The design for the CSCS components to meet the above guidelines was dependent upon three previously set parameters: The maximum break size, low water level scram set point and the CSCS initiation set point. To lower the set point for initiation of the CSCS may lead to a decrease in effective core cooling. To raise the CSCS initiation set point would be in a safe direction, but it would reduce the margin established to prevent actuation of the CSCS during normal operation or during normally expected transients.

Transient and accident analyses reported in Section 14 of the Final Safety Analyses Report demonstrate that these conditions result in adequate safety margins for the fuel.

C. References

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

2. Station Safety Analysis Report (Section XIV).

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

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

REFERENCES

1. Station Safety Analysis (Section XIV)
2. ASME Boiler and Pressure Vessel Code, Section III
3. USAS Piping Code, Section B31.1
4. Reactor Vessel and Appurtenances Mechanical Design (Subsection IV-2)
5. Station Nuclear Safety Operational Analysis (Appendix G)
6. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1, Reload 4", December 1978 (NEDO-24170).

2.2 BASES

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

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

Reanalysis in Reference 6 for the case of MSIV-Closure with flux scram transient results in the peak pressure of 1276 psig at the vessel bottom. This represents a 99 psi margin below the maximum of 110 percent of design pressure allowed by the Code. This is adequate margin to ensure that the 1375 psig pressure safety limit is not exceeded. A sensitivity study on peak vessel pressure to the failure to open of one of the lowest set-point safety valves was performed for a typical high power density BWR (Reference 7). The study is applicable to the Cooper reactor and shows that the sensitivity of a high power density plant to the failure of a safety valve is approximately 20 psi. A plant specific analysis for the Cooper Reload 3 overpressure transient would show results equal to or less than this value.

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

REFERENCES

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

2. Station Nuclear Safety Operational Analysis (Appendix G)

3. Station Safety Analysis (Section XIV)

4. Control and Instrumentation (Section VII)

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

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

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

COOPER NUCLEAR STATION
TABLE 3.2.A (Page 1)
PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION INSTRUMENTATION

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

NOTES FOR TABLE 3.2.A

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

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

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

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

   C. Isolate the Reactor Water Cleanup System.

   D. Isolate the Shutdown Cooling System.

3. Two required for each steam line.

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

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

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

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

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

   Group 1

   Isolation Signals:

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

   Isolations:

   1. MSIV's
   2. Main steam line drains

COOPER NUCLEAR STATION
TABLE 3.2.C
CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

Function	Trip Level Setting	Minimum Number Of Operable Instrument Channels/Trip System(5)
APRM Upscale (Flow Bias)	$\leq$ (0.66W + 42%) $\left[\frac{FRP}{MFLPD}\right]$ (2)	2(1)
APRM Upscale (Startup)	$\leq$ 12%	2(1)
APRM Downscale (9)	$\geq$ 2.5%	2(1)
APRM Inoperative	(10b)	2(1)
RBM Upscale (Flow Bias)	$\leq$ (0.66W + 39%) (2)	1
RBM Downscale (9)	$\geq$ 2.5%	1
RBM Inoperative	(10c)	1
IRM Upscale (8)	$\leq$ 108/125 of Full Scale	3(1)
IRM Downscale (3)(8)	$\geq$ 2.5%	3(1)
IRM Detector Not Full In (8)		3(1)
IRM Inoperative (8)	(10a)	3(1)
SRM Upscale (8)	$\leq 1 \times 10^5$ Counts/Second	1(1)(6)
SRM Detector Not Full In (4)(8)	($\geq$ 100 cps)	1(1)(6)
SRM Inoperative (8)	(10a)	1(1)(6)
Flow Bias Comparator	$\leq$ 10% Difference In Recirc. Flows	1
Flow Bias Upscale/Inop.	$\leq$ 110% Recirc. Flow	1
SRM Downscale (8)(7)	$\geq$ 3 Counts/Second (0.3 counts/second prior to achieving burnup of 3500 MWD/T on first core)	1(1)(6)
RSCS Rod Group C Bypass	$\geq$ 20% Core Thermal Power	(11)

3.2 BASES (cont'd.)

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

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

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

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

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

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

The HPCI high flow and temperature instrumentation are provided to detect a

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

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

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

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the safety limit. The limiting power transient is defined in Reference 3. Analysis of this transient shows that the negative reactivity rates resulting from the scram provide the required protection, and MCPR remains greater than the safety limit.

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

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

In the analytical treatment of the transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and start of motion of the control rods. This is adequate and conservative when compared to the typical time delay of about 210 milliseconds estimated from scram test results. Approximately the first 90 milliseconds of each of these time intervals result from the sensor and circuit delays; at this point, the pilot scram solenoid deenergizes. Approximately 120 milliseconds later, the control rod motion is estimated to actually begin. However, 200 milliseconds is conservatively assumed for this time interval in the transient analyses and this is also included in the allowable scram insertion times of Specification 3.3.C. The time to deenergize the pilot valve scram solenoid is measured during the calibration tests required by Spec 4.1.

D. Reactivity Anomalies

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

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

E. Recirculation Pumps

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

REFERENCES

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

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

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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.11 FUEL RODS	4.11 FUEL RODS
Applicability	Applicability
The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.	The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.
Objective	Objective
The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.	The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.
Specifications	Specifications
A. Average Planar Linear Heat Generation Rate (APLHGR)	A. Average Planar Linear Heat Generation Rate (APLHGR)
During steady state power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figure 3.11-1. If at any time during steady state operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.	The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power.
B. Linear Heat Generation Rate (LHGR)	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:	The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% rated thermal power.
$LHGR_{max} \leq LHGR_d \left[1 - \{(\Delta P/P)_{max}(L/LT)\}\right]$	
$LHGR_d$ = Design LHGR = G KW/ft.	
$(\Delta P/P)_{max}$ = Maximum power spiking penalty = N	

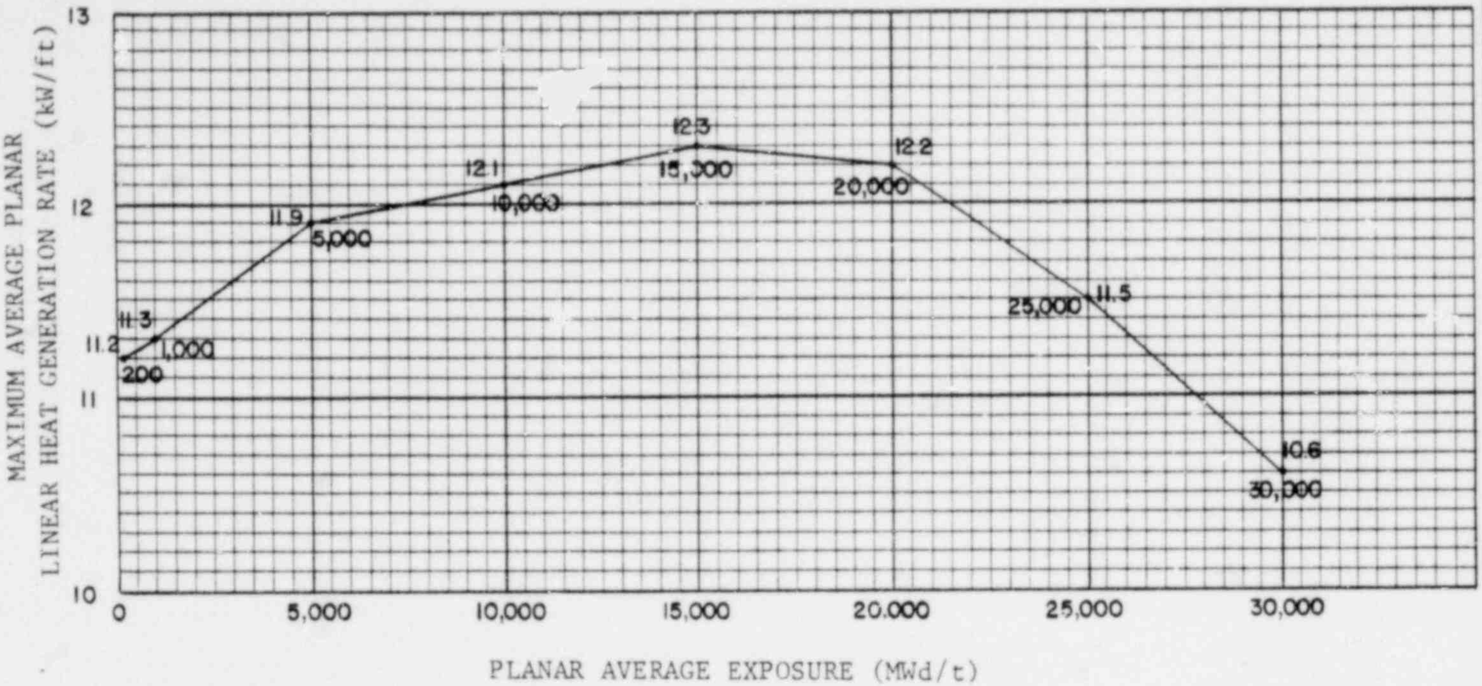

Figure 3.11-1.3. Maximum Average Planar Linear Heat Generation Rate versus Exposure with LPCI Modification and Bypass Flow Holes Plugged, 8D250 Fuel

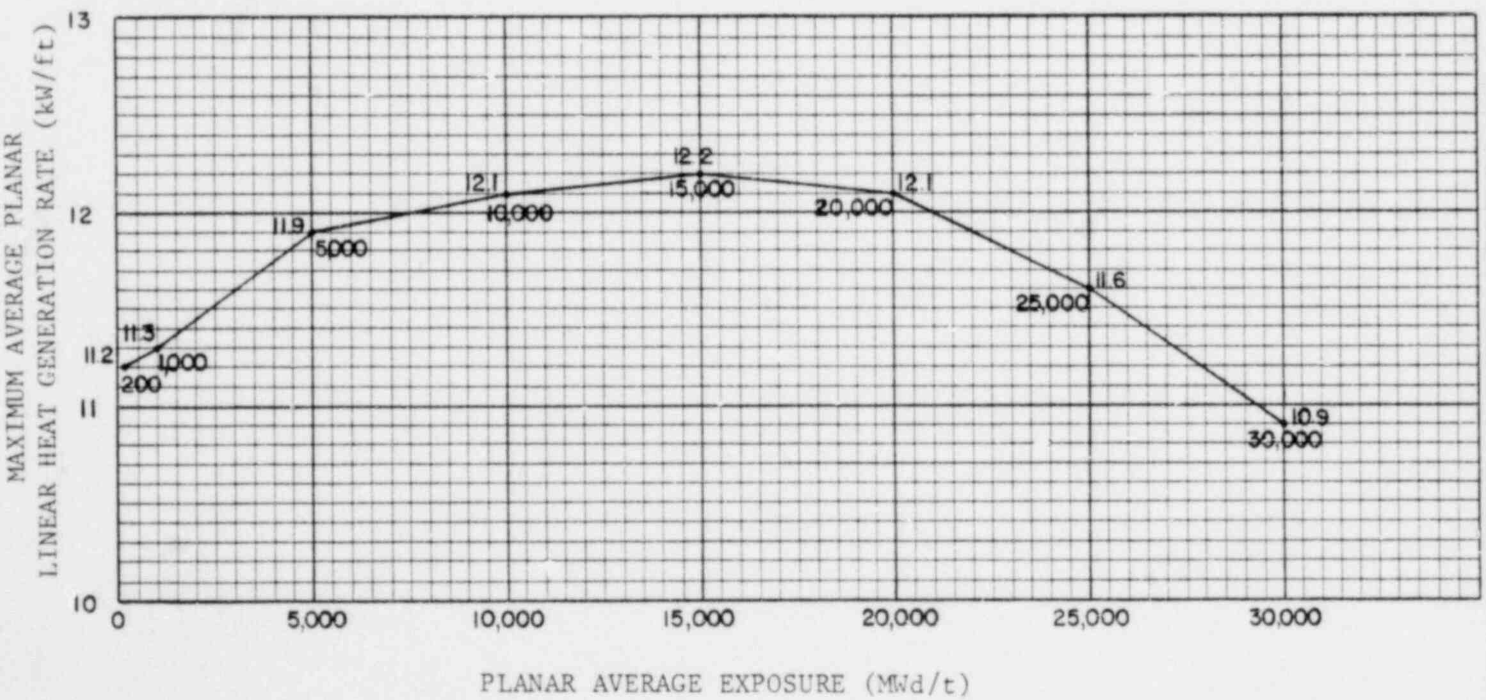

Figure 3.11-1.4. Maximum Average Planar Linear Heat Generation Rate versus Exposure with LPCI Modification and Bypass Flow Holes Plugged, 8D274L Fuel

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT

LT = Total core length - 12 feet

L = Axial position above bottom of core

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

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

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

C. Minimum Critical Power Ratio (MCPR)

During steady state power operation MCPR shall be $\geq$ 1.23 for 7x7 bundles and $\geq$ 1.23 for 8x8 bundles, at rated power and flow. If, at any time during steady state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

For core flows other than rated the MCPR shall be the operating limit at rated flow times K_f, where K_f is as shown in Figure 3.11-2.

C. Minimum Critical Power Ratio (MCPR)

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

3.11 Bases: (Cont'd)

REFERENCES

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

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

B. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 5 of Reference 2 and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern. Pellet densification power spiking in 8x8 fuel has been accounted for in the safety analysis presented in Reference 5; thus no adjustment to the LHGR limit for densification effects is required for 8x8 fuels.

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

The required operating limit MCPR's at steady state operating conditions a specified in Specification 3.11C are derived from the established fuel cladding integrity Safety Limit MCPR of 1.07, and an analysis of abnormal operational transients (Reference 5). For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.1.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the more limiting transients have been analyzed to determine which result in the largest reduction in critical power ratio (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

3.11 Bases: (Cont'd)

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

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

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

D. MCPR Limits for Core Flows Other than Rated

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

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated for the same postulated transient event.

3.11 Bases: (Cont'd)

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

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

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

The K_f factors shown in Figure 3.11-2, are conservative for Cooper operation because the operating limit MCPR's are greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

References

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

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

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

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

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

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

4.11 Bases:

A & B. Average and Local LHGR

The LHGR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

C. Minimum Critical Power Ratio (MCPR) - Surveillance Requirement

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

D. Core Stability

The calculations, regarding reactor core stability, presented in "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1, Reload 4," December 1978 (NEDO-24170), show that the reactor is in compliance with the ultimate performance criteria, including the most responsive condition at natural circulation and rod block power. However, to preclude the possibility of operation under conditions which could result in reactor core instability, the NRC requested the incorporation of a specification limit.

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

CHANGE IN MAIN STEAM LINE LOW PRESSURE ISOLATION SETTINGS

REASON FOR CHANGES

The implications of changing the low pressure Main Steam Line isolation signal from the current minimum Technical Specification limit of 850 psig have been evaluated and it has been determined that the limiting setting can be safely lowered to 825 psig. This change would help preclude spurious scrams and isolations on low pressure transients and reduce the number and magnitude of safety relief valve loadings on the Mark I Containment torus.

JUSTIFICATION FOR CHANGES

The low pressure isolation signal is provided to give protection against fast depressurization thereby limiting cooldown on the vessel and fuel duty. The 850 psig low pressure isolation was originally determined based on judgment and was chosen approximately 100 psi less than the turbine inlet pressure. The 100 psi number is not critical and a larger value would result in only small changes in the effects on saturation temperature and fuel duty (the difference in saturation temperature between 850 psig and 750 psig is approximately $15^{o}F$). Operating margins less than 100 psi could cause unwanted isolations after scram events. This can occur since pressure regulators with their built-in control time constants may not be able to limit the pressure drop via the control valves (or bypass valves) before the isolation setpoint is reached; or could cause scrams on low pressure transients.

Lowering of the Technical Specification limit to 825 psig will not invalidate the transient safety analyses reported in the "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1, Reload 4", December 1978 (NEDO-24170) and will result in a negligible added requirement in terms of fuel duty and vessel cooldown. Therefore, lowering of the existing isolation setpoint as described above will not degrade the degree of protection offered by this safety system.