Docket File
NRC PDR
Local PDR
ORB#2 Rdg.
DEisenhut
SNorris
BSiegel
OELD
SECY
LJHarmon
EJordan
JTaylor
TBarnhart 4
WJones
DBrinkman
ACRS 10
OPA, CMiles
RDiggs
NSIC
Gray File
XTRA-5

May 4, 1983

Docket No. 50-298

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

Dear Mr. Pilant:

The Commission has issued the enclosed Amendment No. to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. This amendment consists of changes to the Technical Specifications in response to your application dated December 17, 1982 and supplemented by your February 15, 1983 submittal.

This amendment to the Technical Specifications revises the following; (1) lowers the MSIV low reactor water level isolation setpoint from the current level 2 to level 1, (2) lowers the opening and closing setpoints of two preselected SRVs upon opening of any SRV, and (3) reorganizes the bases of Section 3.2, Protective Instrumentation to categorize the systems covered and incorporates the bases for the Technical Specification changes addressed by this amendment.

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

Sincerely,

ORIGINAL SIGNED BY

Byron Siegel, Project Manager
Operating Reactors Branch #2
Division of Licensing

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

cc w/enclosures
See next page

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OFFICE	DL:ORB#2	DL:ORB#2	DL:ORB#2	AD OR:DL	OELD		
SURNAME	SNorris	BSiegel:pg	DVassallo	GLainas			
DATE	5/2/83	5/2/83	5/2/83	5/2/83	5/3/83		

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

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 83
License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District dated December 17, 1982, supplemented February 15, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 83 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGUALTORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 4. 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 83

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following pages of the Appendix "A" Technical Specifications as noted below. The revised pages are identified by Amendment number.

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iiia	-
iv	iv
-	v
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35	35
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COOPER NUCLEAR STATION
TABLE 3.1.1 (Page 2)
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 Systems (1)	Action Required When Equipment Operability is Not Assured (1)
Main Steam Line High Radiation RMP-RM-251, A,B,C, & D		X(9)		X	≤ 3 Times normal full power back ground	2	A or D
Main Steam Line Isolation Valve Closure MS-LMS-86 A,B,C, & D MS-LMS-80 A,B,C, & D		X(3)(6)(9)		X(6)	≤ 10% of valve closure	4 4	A or C A or C
Turbine Control Valve Fast Closure TGF-63/OPC-1,2,3,4				X(4)	≥ 1000 psi turbine control fluid	2	A or B
Turbine Stop Valve Closure SVOS-1(1), SVOS-1(2) SVOS-2(1), SVOS-2(2)				X(4)	≤10% of valve Closure	2	A or B
Turbine First Stage Permissive MS-PS-14 A,B,C, & D		X(9)		X	≤ 30% first stage press.	2	A or B

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TABLE 4.1.1 (Page 2)
REACTOR PROTECTION SYSTEM (SCRAM INSTRUMENTATION) FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

Instrument Channel	Group (2)	Functional Test	Minimum Frequency (3)
High Water Level in Scram Discharge Volume CRD-LS-231 A & B CRD-LS-234 A & B CRD-LT-231 C & D CRD-LT-234 C & D	A	Trip Channel and Alarm	Once/3 Months
Main Steam Line High Radiation RMP-RM-251 A,B,C, & D	B	Trip Channel and Alarm (4)	Once/Week
Main Steam Line Isolation Valve Closure MS-LMS-86 A,B,C, & D MS-LMS-80 A,B,C, & D	A	Trip Channel and Alarm	Once/Month (1)
Turbine Control Valve Fast Closure TGF-63/OPC -1,2,3,4	A	Trip Channel and Alarm	Once/Month (1)
Turbine First Stage Pressure Permissive MS-PS-14 A,B,C, & D	A	Trip Channel and Alarm	Once/3 Months
Turbine Stop Valve Closure SVOS-1 (1), SVOS-1 (2) SVOS-2 (1), SVOS-2 (2)	A	Trip Channel and Alarm	Once/Month (1)

COOPER NUCLEAR STATION
TABLE 4.1.2
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration Test (5)	Minimum Frequency (2)
IRM High Flux	C	Comparison to APRM on Controlled Shutdowns (7)	Note (4)
APRM High Flux			
Output Signal	B	Heat Balance	Once/Week
Flow Bias Signal	B	Internal Power and Flow Test with Standard Pressure Source (8)	Once/Refueling Outage
LPRM Signal	B	TIP System Traverse	Note (9)
High Reactor Pressure	A	Standard Pressure Source	Once/3 Months
High Drywell Pressure	A	Standard Pressure Source	Once/3 Months
Reactor Low Water Level	A	Pressure Standard	Once/3 Months
High Water Level in Scram Discharge Volume	A	Note (6)	Note (6)
Main Steam Line Isolation Valve Closure	A	Note (6)	Note (6)
Main Steam Line High Radiation	B	Standard Current Source (3)	Once/3 Months
Turbine First Stage Pressure Permissive	A	Standard Pressure Source	Once/6 Months
Turbine Control Valve Fast Closure	A	Standard Pressure Source	Once/3 Months
Turbine Stop Valve Closure	A	Note (6)	Note (6)

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 Water Level ($\geq$-145.5 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.B (Page 7)
AUTOMATIC DEPRESSURIZATION SYSTEM (ADS) CIRCUITRY REQUIREMENTS

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required When Component Operability Is Not Assured
Drywell High Pressure	PC-PS-100, A,B,C & D	≤ 2 psig	2	A
Reactor Low Water Level	NBI-LIS-83, A & B	≥ +12.5" Indicated Level	1	B
	NBI-LIS-72, A,B,C & D	≥ -145.5" Indicated Level	2	A
ADS Timer	MS-TDR-K5, A & B	≤ 120 sec.	1	B
Low-Low Set	NBI-PS-51, A,B,C & D	51-A Open Low Valve 1015±10 psig (Increasing) 51-B Close Low Valve 875±10 psig (Decreasing) 51-C Open High Valve 1025±10 psig (Increasing) 51-D Close High Valve 875±10 psig (Decreasing)	2	B

COOPER NUCLEAR STATION
TABLE 4.2.B (Page 7)
ADS SYSTEM TEST & CALIBRATION FREQUENCIES

Item	Item I.D. No.	Functional Test Freq.	Calibration Freq.	Instrument Check
Instruments				
1. Drywell High Pressure	PC-PS-100 A,B,C, & D	Once/Month (1)	Once/3 Months	None
2. Reactor Low Water Level	NBI-LIS-83, A & B	Once/Month (1)	Once/3 Months	Once/Day
	NBI-LIS-72, A,B,C, & D	Once/Month (1)	Once/3 Months	Once/Day
3. ADS Timer	MS-TDR-K5 A & B	Once/Month (1)	Once/Oper. Cycle	None
4. Low-Low Set	NBI-PS-51, A,B,C, & D	Once/Month (1)	Once/Oper. Cycle	None
Logic (4)(6)				
1. ADS Control Power Monitor		Once/6 Months	N.A.	
2. ADS Actuation		Once/6 Months	N.A.	
3. Low-Low Set Logic		Once/6 Months	N.A.	

3.2 BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The set points of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

A. Primary Containment Isolation Functions

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation,n, set to trip at 176.5" (+12.5") above the top of the active fuel, closes all isolation valves except those in Groups 1, 4, and 5. Details of valve grouping and required closing times are given in Specification 3.7. For valves which isolate at this level this trip setting is adequate to prevent core uncovery in the case of a break in the largest line assuming a 60 second valve closing time. Required closing times are less than this.

The low reactor water level instrumentation is set to trip when reactor water level is 127" (-37") above the top of the active fuel. This trip closes Main Steam Drain Valves, Recirc Sample Valves (Groups 1 and 7), initiates the HPCI and RCIC. The low reactor water level instrumentation is set to trip when the water level is 19" (-145") above the top of the active fuel. This trip closes Main Steam Line Isolation Valves (Reference 1), and activates the remainder of the CSCS subsystems, and starts the emergency diesel generators. These trip level settings were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation and primary system isolation so that post accident cooling can be accomplished,

and the guidelines of 10CFR100 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 XIV.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 3 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10CFR100 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 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.

B. Core and Containment Cooling Systems Initiation and Control

The instrumentation which initiates Core Standby Cooling System (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.

CORE SPRAY - No Basis

LPCI MODE - No Basis

HPCI

The HPCI high flow and temperature instrumentation are provided to detect a 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°F for high temperature are such that core uncovery is prevented and fission product release is within limits.

RCIC

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°F for temperature are based on the same criteria as the HPCI.

ADS

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

The low-low set relief logic is an automatic safety relief valve (SRV) control system designed to mitigate the postulated thrust load concern of subsequent actuations of SRV's during small break loss-of-coolant accident (LOCA) events. The setpoints used in Table 3.2.B are based upon a minimum blowdown range to provide adequate time between valve actuations to allow the SRV discharge line high water leg to clear, coupled with consideration of instrument inaccuracy and the main steam isolation valve isolation setpoint.

C. Control Rod Block Actuation

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to the safety limit CPR. 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 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 the safety limit CPR.

The RBM rod block function provides local protection of the core; i.e., the prevention of the MCPR reaching the safety limit CPR in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

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

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

The flow comparator and scram discharge volume high level components have only one logic channel and are not required for safety. The SDV high level rod block does provide adequate time to determine the cause of the level increase and take corrective action prior to automatic scram.

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

D. Radiation Monitoring Systems - Isolation and Initiation Functions

1. Steam Jet Air Ejector Off-Gas System

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

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

2. Reactor Building Isolation and Standby Gas Treatment Initiation

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

Trip settings of <100 mr/hr for the monitors in the ventilation exhaust ducts are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leaves the Reactor Building via the normal ventilation path but rather all the activity is processed by the standby gas treatment system.

3. Liquid Radwaste Discharge Isolation

The liquid radwaste monitor assures that all liquid discharged to the discharge canal does not exceed the limits of Section 2.4.1.b of Environmental Technical Specifications. Upon sensing a high discharge level, an isolation signal is generated which closes the radwaste discharge valve. The set point is adjustable to compensate for variable isotopic discharges and dilution flow rates.

4. Main Control Room Ventilation

The main control room ventilation isolation is provided by a detector monitoring the intake of the control room ventilation system. Automatic isolation of the normal supply and exhaust and the activation of the emergency filter system is provided by the radiation detector trip function at the predetermined trip level.

5. Mechanical Vacuum Pump

The mechanical vacuum pump isolation prevents the exhausting of radioactive gas thru the 1 minute holdup line upon receipt of a main steam line high radiation signal.

E. Drywell Leak Detection

Flow transmitters are used to record the flow of liquid from the drywell sumps. An air sampling system is also provided to detect leakage inside the primary containment.

F. Primary Containment Surveillance Information Readouts

For each parameter monitored, as listed in Table 3.2.F, there are two (2) channels of instrumentation. By comparing readings between the two (2) channels, a near continuous surveillance of instrument performance is available. Any deviation in readings will initiate an early recalibration, thereby maintaining the quality of the instrument readings.

The operability of the reactor water level instrumentation in Tables 3/4.2.F ensures that sufficient information is available to monitor and assess accident situations.

G. Recirculation Pump Trip

The recirculation pump trip has been added as a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events given in General Electric Company Topical Report, NEDO-10349, dated March, 1971.

REFERENCES

1. "Low-Low Set Relief Logic System and Lower MSIV Water Level Trip for Cooper Nuclear Station", NEDE 22197, December 1982, General Electric Company.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 83 TO LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

1.0 Introduction

The low-low set (LLS) relief logic modification for BWRs with Mark 1 Containments is designed to prevent multiple subsequent actuations of safety relief valves (SRV) which might normally be expected during a transient. This in turn will reduce or prevent the discharge loads on the containment and suppression pool structures resulting from subsequent SRV actuations. The discharge loads from subsequent actuations tend to be higher due to the condensation of trapped steam in the safety relief valve discharge line (SRVDL), which results in a higher water leg in the SRVDL, and hence, larger thrust loads on subsequent actuations.

The LLS design modification is an automatic SRV actuation system which, upon initiation, will assign preset opening and closing setpoints to two preselected SRVs. These setpoints are selected such that the LLS controlled SRVs will stay open longer, thus releasing more steam (energy) to the suppression pool, and hence more energy (and time) will be required for repressurization and subsequent SRV openings. The LLS increases the time between (or prevents) subsequent actuations sufficiently to allow the high water leg created from the initial SRV opening to return to or below its normal water level, thus, reducing thrust loads from subsequent actuations to within their design limits. In addition, since the LLS is designed to limit SRV subsequent actuations to one valve, torus loads will also be reduced.

The lower MSIV water level trip causes the MSIV closure actuation to be changed from a reactor water level two signal to a reactor water level one signal. This design modification maintains the main condenser availability for a longer time which allows more energy to be released to the main condenser and will result in a slower repressurization rate. The lower MSIV water level trip reduces isolations, SRV challenges and provides some benefit to SRV subsequent actuations.

In a letter dated December 17, 1982 from Nebraska Public Power District (licensee) Technical Specification changes were requested that incorporate these logic changes into the CNS Mark I Containment design. This was supplemented by information provided in a February 15, 1983 letter.

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2.0 Evaluation

2.1 System Transient and Accident Performance and-Overall Plant Safety Aspects

The General Electric (GE) generic evaluation submittals (Refs. 1 & 2) considered abnormal operational transients, design basis accidents and the anticipated transient without scram (ATWS) events to determine the impact of these design modifications on overall plant safety margins. The safety evaluation for abnormal operational transients included the following considerations to determine that the design modifications will not produce any adverse effects on safe plant systems operation and plant safety margins: (1) Reduction in Minimum Critical Power Ratio (MCPR); (2) Increase in Peak Pressure; (3) Increase in Radiation Release; (4) Cause for Equipment Damage; (5) Reduction in Plant Shutdown Capability; and (6) Decrease in Core Cooling Capability.

The limiting transient events, such as MSIV trip with flux scram, and turbine trip from high power without bypass were evaluated. General Electric concluded that the LLS will not affect the MCPR or peak pressure, because these conditions occur early in the transients before the reactor pressure response is affected by the LLS SRVs subsequent actuations.

The effects of the LLS on LOCAs were evaluated using the approved GE Appendix K evaluation models for the entire break spectrum. The evaluation showed that the LLS logic has no effect on the limiting Maximum Average Planar Linear Heat Generation Rate (MAPLHGR), because the rapid reactor depressurization precludes the actuation of LLS SRVs during the design basis accident.

For ATWS events, General Electric determined that the LLS logic has no effects because this logic would not affect the reactor pressure until after the ATWS-associated short-term pressure transient is over.

It was also concluded that lowering the MSIV water level trip will not have any effects on the limiting MCPR,peak pressure and MAPLHGR during abnormal operational transients, LOCAs and ATWS events.

The impact of these design modifications on other effects such as, increase in radiation release, cause for equipment damage, reduction in plant shutdown capability, pool heat-up and decrease in core cooling capability were also considered in this generic evaluation.

The General Electric plant specific submittal (Ref. 3) identified that only non ADS-SRVs are used in the LLS logic system to reduce subsequent plant transients and that the LLS logic would extend SRV subsequent actuation time sufficiently to clear the water columns in the SRV discharge line.

Based on our review of both the General Electric generic and plant specific submittals, we have concluded that these design modifications are acceptable, since they will not adversely affect overall plant performance and safety considerations.

2.2 Low-Low Set (LLS) Circuitry

The LLS initiation circuitry consists of two redundant channels each of which controls power to a different SRV solenoid. There are eight SRVs at CNS, six of which are actuated by the Automatic Depressurization System (ADS). The two non-ADS SRVs (71D and 71F) will be used for the LLS function. Each of the two LLS controlled SRVs will open when their respective solenoid becomes energized. In order for either LLS channel to energize its solenoid, both an arming logic and an initiation logic must be satisfied. The arming logic is satisfied when any SRV has opened and reactor pressure (two-out-of-two logic for each LLS channel) has exceeded the high pressure setpoint (this setpoint is selected above the reactor protection system high reactor pressure scram setting to assure that a scram has occurred). Four separate reactor high pressure channels (two for each LLS channel) are used. These instrument channels are part of the existing nuclear boiler instrumentation that provides inputs to the reactor protection system (RPS).

Once the arming logic for either LLS channel is satisfied, it is sealed in and annunciated in the control room, and remains sealed in until manually reset by the operator. In addition, the arming logic in either LLS channel will seal in the arming logic in the other LLS channel provided the reactor high pressure permissive in that channel is satisfied. Separation between LLS channels for this arming signal is provided by coil-contact separation. Initial SRV actuations are detected by pressure switches located in the SRV discharge lines. These pressure switches are set slightly above the normal pressure expected in the discharge line (30 psig). Once armed, the LLS actuation/control logic uses nuclear boiler system reactor pressure instrumentation to control the LLS SRV solenoids, thus opening and closing these SRVs at their assigned LLS setpoints. This control logic remains in effect as long as the arming logic is sealed in. Both LLS logic channels can be tested at power. Test status lights in the control room indicate when the arming logic and control logic relays have operated satisfactorily during testing. These test lights can also be used to verify proper operation of LLS seal in and reset circuits. The SRV discharge line pressure switches (one switch per SRV) and reactor pressure instrument channels (used both for the arming logic permissive and LLS SRV control) must be tested separately. Each LLS channel provides annunciation in the control room upon loss of power. Test switches are provided to verify operability of this power monitor function. Additional status lights are provided in the control room to indicate that the two-out-of-two reactor high pressure permissives have been satisfied. The licensee has included in their proposed plant Technical Specifications changes monthly testing of the pressure instrumentation used for LLS SRV control, and semiannual testing of the LLS logic. Technical Specification surveillance requirements already exist for the SRV discharge line pressure switches and the reactor pressure channels used for the LLS arming logic permissives. These test frequencies are acceptable to the staff.

Power to each of the two LLS channels may be provided by one of two separate sources. Normally, both LLS channels will be powered from the Division 1 125 Vdc supply. If this supply should fail, power will automatically be supplied to both LLS channels from the Division 2 125 Vdc supply. Each LLS channel contains circuitry (consisting of one relay and four associated contacts to disconnect the Division 1 supply if it fails and connect the Division 2 supply in its place) to perform this automatic transfer function. Since this automatic transfer feature is part of the existing protection system (ADS) circuitry, and since redundant protective devices (fuses and circuit breakers) are located between the automatic transfer and the safety buses, the staff finds this design to be acceptable.

The LLS circuitry contains no channel or operating bypasses. The circuitry added for this LLS function is separated in accordance with IEEE 384-1974. The components of the LLS system (including power supplies) are classified as Class 1E. The LLS will remain operable in the event of loss of offsite power. LLS components located inside the drywell are qualified for the environmental conditions associated with a small break LOCA. Based on our review, we have determined that the proposed LLS modification to be installed at CNS is designed to perform its intended function given a single failure. In addition, no single failure in the electrical circuits could be found which would cause a spurious SRV actuation. The LLS is designed in accordance with the requirements of IEEE standard 279-1971 and therefore, is acceptable.

2.3 Evaluation Summary

System Transient and Accident Performance Overall Plant Safety Aspects.

We find these design modifications, LLS logic and lowered MSIV water level trip, acceptable because they will not adversely affect plant performance or safety margins. These modifications are compatible with normal plant operation and other safety systems.

LLS Circuitry

Based on our review, we have determined that the LLS modification installed at Cooper Nuclear Station is designed to perform its intended function given a single failure. In addition, no single failure in the electrical circuits could be found which would cause a spurious SRV actuation. The LLS is designed in accordance with the requirements of IEEE Standard 279-1971 and therefore, is acceptable.

3.0 Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different

from any evaluated previously, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 References

1. General Electric letter MFN-176-82 dated November 19, 1982.
2. General Electric Report NEDE-22223 dated September 1982.
3. General Electric Report NEDE-22197 dated December 1982.

Dated: May 4, 1983

Principal Contributor: K. Desai
R. Kendall

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-298

NEBRASKA PUBLIC POWER DISTRICT

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 83 to Facility Operating License No. DPR-46 issued to Nebraska Public Power District (the licensee) which revised the Technical Specifications for operation of the Cooper Nuclear Station located in Nemaha County, Nebraska. The amendment is effective as of the date of its issuance.

The amendment to the Technical Specifications revises the following: (1) lowers the MSIV low reactor water level isolation setpoint from the current level 2 to level 1, (2) lowers the opening and closing setpoints of two pre-selected SRVs upon opening of any SRV, and (3) reorganizes the bases of Section 3.2, Protective Instrumentation to categorize the systems covered and incorporates the bases for the Technical Specification changes addressed by this amendment.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

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For further details with respect to this action, see (1) the application for amendment dated December 17, 1982, as supplemented February 15, 1983, (2) Amendment No. 83 to License No. DPR-46 and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W. Washington, D. C., and at the Auburn Public Library, 118 - 15th Street, Auburn, Nebraska 68304. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 4th day of May 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing