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Extra-5	Gray	NSIC			

Docket No. 50-298

MAR 4 1982

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. 77 to Facility Operating License No. DPR-46 for Cooper Nuclear Station. This amendment consists of changes to the Technical Specifications in response to your application dated October 21, 1980 and subsequent discussions between the NRC staff and the licensee.

These changes to the Technical Specifications pertain to the scram discharge volume (SDV) and include: SDV vent and drain valve surveillance, the addition of a SDV control rod block limiting condition for operation and surveillance requirement, and clarifications of an administrative nature pertaining to the minimum number of APRM operable instrument changes required for the rod block monitors.

The approval of these technical specifications was based on your October 21, 1980 submittal. This submittal was reviewed by our consultant under a technical assistance contract. A copy of their Technical Evaluation Report (TER-C5506-69) is enclosed. We have reviewed the Franklin Research Center TER and agree with its conclusion that your response is acceptable. The enclosed safety evaluation documents this finding.

cp
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A copy of this Notice of Issuance is also enclosed.

Sincerely,

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

Enclosures:
1. Amendment No. 77 to DPR-46
2. Safety Evaluation
3. Technical Evaluation Report - previously provided
4. Notice of Issuance

cc w/enclosures
See next page

I have no legal objection subject to my comment in my note to Siegel 3/1/82.
D. Hassell 3/1/82

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DATE			2/11/82	2/ /82	2/21/82	2/4/82

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Mr. J. M. Pilant
Nebraska Public Power District

cc:

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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. 77
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 October 21, 1980 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. 77 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

DB Vassallo

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

Attachment:
Changes to the Technical
Specifications

Dated: March 4, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 77

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Revise the Appendix A Technical Specifications as indicated below. Marginal lines on the revised pages indicate the area of change.

Remove	Insert
61	61
62	62
62a	62a
77	77
86	86
98	98
104	104
105	105

Amendment No. 82 77

TABLE 3.2.C
CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

Function	Trip Level Setting	Minimum Number Of Operable Instrument Channels/Trip System(5)
APRM Upscale (Flow Bias)	$\leq (0.66W + 42\%)\ \frac{FRP}{MFLPD}$ (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 + 41%) (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 (11)	1(1)(6)
SDV Water Level High	$\leq$ 18 gallons	1(12)

NOTES FOR TABLE 3.2.C

1. For the startup and run positions of the Reactor Mode Selector Switch, the Control Rod Withdrawal Block Instrumentation trip system shall be operable for each function. The SRM and IRM blocks need not be operable in "Run" mode, and the APRM (flow biased) and RBM rod blocks need not be operable in "Startup" mode. The Control Rod Withdrawal Block Instrumentation trip system is a one out of "n" trip system, and as such requires that only one instrument channel specified in the function column must exceed the Trip Level Setting to cause a rod block. By utilizing the RPS bypass logic (see note 5 below and note 1 of Table 3.1.1) for the Control Rod Withdrawal Block Instrumentation, a sufficient number of instrument channels will always be operable to provide redundant rod withdrawal block protection.

2. W is the recirculation loop flow in percent of design. Trip level setting is in percent of rated power (2381 MWt).

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

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

5. By design one instrument channel; i.e., one APRM or IRM per RPS trip system may be bypassed. For the APRM's and IRM's, the minimum number of channels specified is that minimum number required in each RPS channel and does not refer to a minimum number required by the control rod block instrumentation trip function. By design only one of two RBM's or one of four SRM's may be bypassed. For the SRM's, the minimum number of channels specified is the minimum number required in each of the two circuit loops of the Control Rod Block Instrumentation Trip System. For the RBM's, the minimum number of channels specified is the minimum number required by the Control Rod Block Instrumentation Trip System as a whole (except when a limiting control rod pattern exists and the requirements of Specification 3.3.B.5 apply).

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

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

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

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

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

    a. SRM and IRM

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

NOTES FOR TABLE 3.2.C (Continued)

b. APRM

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

c. RBM

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

11. During spiral unloading/reloading, the SRM count rate will be below 3 cps for some period of time. See Specification 3.10.B.

12. With the number of OPERABLE channels less than required by the Minimum Number of Operable Instrument Channels/Trip System requirements, place the inoperable channel in the tripped condition within one hour.

TABLE 4.2.C
SURVEILLANCE REQUIREMENTS FOR ROD WITHDRAWAL BLOCK INSTRUMENTATION

Function	Functional Test Freq.	Calibration Freq.	Instrument Check
APRM Upscale (Flow Bias)	(1) (3)	Once/3 Months	Once/Day
APRM Upscale (Startup Mode)	(1) (3)	Once/3 Months	Once/Day
APRM Downscale	(1) (3)	Once/3 Months	Once/Day
APRM Inoperative	(1) (3)	N.A.	Once/Day
RBM Upscale (Flow Bias)	(1) (3)	Once/6 Months	Once/Day
RBM Downscale	(1) (3)	Once/6 Months	Once/Day
RBM Inoperative	(1) (3)	N.A.	Once/Day
IRM Upscale	(1) (2) (3)	Once/3 Months	Once/Day
IRM Downscale	(1) (2) (3)	Once/3 Months	Once/Day
IRM Detector Not Full In	(2) (Once/oper. ating cycle)	Once/Oper. Cycle (10)	Once/Day
IRM Inoperative	(1) (2) (3)	N.A.	N.A.
SRM Upscale	(1) (2) (3)	Once/3 Months	Once/Day
SRM Downscale	(1) (2) (3)	Once/3 Months	Once/Day
SRM Detector Not Full In	(2) (Once/oper- ating cycle)	Once/Oper. Cycle (10)	N.A.
SRM Inoperative	(1) (2) (3)	N.A.	N.A.
Flow Bias Comparator	(1) (8)	Once/Oper. Cycle	N.A.
Flow Bias Upscale	(1) (8)	Once/3 Months	N.A.
Rod Block Logic	(9)	N.A.	N.A.
RSCS Rod Group C Bypass	(1) (11)	Once/3 Months	N.A.
SDV High Water Level	Quarterly	Once/Oper. Cycle	N.A.

3.2 BASES (cont'd.)

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

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

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

The flow comparator and scram discharge 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 a safety only when the mode switch is in the refueling position.

The effective emergency core cooling for small pipe breaks, the HPCI system, must function since reactor pressure does not decrease rapid enough to allow either core spray 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.

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

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

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

LIMITING CONDITIONS FOR OPERATION

3.3.C (Cont'd.)

3. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

D. Reactivity Anomalies

At a specific steady state base condition of the reactor actual control rod inventory will be periodically compared to a normalized computer prediction of the inventory. If the difference between observed and predicted rod inventory reaches the equivalent of 1% Δk reactivity, the reactor will be shut down until the cause has been determined and corrective actions have been taken as appropriate.

E. Recirculation Pumps

A recirculation pump shall not be started while the reactor is in natural circulation flow and reactor power is greater than 1% of rated thermal power.

F. If Specifications 3.3.A through D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Shutdown condition within 24 hours.

SURVEILLANCE REQUIREMENTS

4.3.C (Cont'd.)

D. Reactivity Anomalies

During the startup test program and startup following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

G. Scram Discharge Volume

1. The scram discharge volume (SDV) vent and drain valves shall be cycled and verified open at least once every 31 days and prior to reactor start-up.

2. The SDV vent and drain valves shall be verified to close within 30 seconds after receipt of a signal for control rod scram once per refueling cycle.

3. SDV vent and drain valve operability shall be verified following any maintenance or modification to any portion (electrical or mechanical) of the SDV which may affect the operation of the vent and drain valves.

3.3 and 4.3 BASES: (Cont'd)

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of a 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.

3.3 and 4.3 BASES: (Cont'd)

G. Scram Discharge Volume

To ensure the Scram Discharge Volume (SDV) does not fill with water, the vent and drain valves shall be verified open at least once every 31 days. This is to preclude establishing a water inventory, which if sufficiently large, could result in slow scram times or only a partial control rod insertion.

The vent and drain valves shut on a scram signal thus providing a contained volume (SDV) capable of receiving the full volume of water discharged by the control rod drives at any reactor vessel pressure. Following a scram the SDV is discharged into the reactor building drain system.

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," (most current approved submittal).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 77 TO LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

Author: K. T. Eccleston

Introduction

As a result of events involving common cause failures of Scram Discharge Volume (SDV) limit switches and SDV drain valve operability, the NRC staff issued IE Bulletin 80-14 on June 12, 1980. In addition, the staff sent a letter dated July 7, 1980 to all operating BWR licensees requesting that they propose Technical Specification changes to provide surveillance requirements for SDV vent and drain valves and LCO/surveillance requirements on SDV limit switches. Model Technical Specifications were enclosed with this letter to provide guidance to licensees for preparation of the requested submittals. The Nebraska Public Power District (licensee) submittal dated October 21, 1980, contained the proposed Technical Specifications requested by the staff.

Evaluation

A Technical Evaluation Report (TER-C5506-69) was prepared for us by Franklin Research Center (FRC) as part of our technical assistance contract program. This report provides their technical evaluation of the compliance of the licensee's October 21, 1980 submittal with NRC provided criteria.

FRC has concluded that the licensee's response does not meet the explicit requirements of paragraph 3.3-6 and Table 3.3.6-1 of the NRC staff's Model Technical Specifications (TS). However, the FRC report concludes that technical bases are defined on p. 50 of the staff's "Generic Safety Evaluation Report BWR Scram Discharge System," December 1, 1980 that permits consideration of this departure from the explicit requirements of the Model Technical Specifications. We conclude that these technical bases justify a deviation from the explicit requirements of the Model TSs.

In addition, FRC has also concluded that the proposed Cooper TSs do not meet the NRC staff's Model TS requirements of Paragraph 4.3.1.1 and Table 4.3.1.1-1 for channel Functional Test for SDV water level high instrumentations. However, the FRC TER concludes that the proposed surveillance requirements for SDV water level high are acceptable, since the licensee is installing a second Instrument Volume containing four additional level switches, for a total of eight limit switches for the reactor protective system. The Model TS were developed for plants which only have one instrument volume (four limit switches); therefore, the installation of a second instrument volume by the licensee significantly improves the design and reliability of the SDV. Taking this into account, we conclude that these technical bases justify a deviation from the explicit requirements of the model TSs.

Summary

Based upon our review of the contractor's report of his evaluations and discussions with the reviewer, we conclude that the licensee's proposed TSs satisfy our requirements for surveillance of SDV vent and drain valves and for Limiting Conditions of Operation and surveillance requirements for SDV limit switches. Consequently, we find the licensee's proposed TSs acceptable.

Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR Section 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 the amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: March 4, 1982

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. 77 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 issuance.

The amendment modifies the Technical Specifications regarding the Scram Discharge Volume (SDV) and include: SDV vent and drain valve surveillance, the addition of a SDV control rod block and surveillance requirement, and clarifications of an administrative nature pertaining to the minimum number of APRM operable instrument changes required for the rod block monitors.

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 Section 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.

For further details with respect to this action, see (1) the application for amendment dated October 21, 1980, (2) Amendment No. 77 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, NW., 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., Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 4th day of March 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

DBVassallo

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