FEB 23 1981

Docket No. 50-298

Distribution:
Docket File
NRC PDR
Local PDR
ORB #2 Rdg
D. Eisenhut
R. Purple
T. Novak
R. Tedesco
G. Lainas
S. Norris
B. Siegel
OELD
OI&E (5)
~~B. Jones~~ (4) F. Baumlart
J. Wetmore
ACRS (16)
OPA
R. Diggs
H. Denton
J. Heltemes, AEOD
NSIC
TERA
A. Rosenthal
ASLABP

Mr. J. M. Pilant, Director
Licensing & Quality Assurance
Nebraska Public Power District
P. O. Box 499
Columbus, Nebraska 68501

Dear Mr. Pilant:

The Commission has issued the enclosed Amendment No. 68 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 letters of January 30, 1980, October 15, 1980 and January 6, 1981.

These changes to the Technical Specifications involve administrative changes and incorporation of certain of the TMI-2 Lessons Learned Category "A" requirements. The Category "A" requirements concern (1) Emergency Power Supply/Inadequate Core Cooling, (2) Valve Position Indication, (3) Containment Isolation, (4) Shift Technical Advisor Augmentation, (5) Integrity of Systems Outside Containment, and (6) Iodine Monitoring.

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

Sincerely,
Original Signed by
T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 68
2. Safety Evaluation
3. Notice

cc w/encls:
See next page

RECEIVED
MAR 04 1981
U.S. NUCLEAR REGULATORY COMMISSION

8103100409

P

OFFICE	DL:ORB#2	DL:ORB#2	DL:ORB#2	DL:OR	OELD		
SURNAME	SNorris	BSiegel:ms	TAIppolito	TMNovak	SThiby		
DATE	2/11/81	2/5/81	2/11/81	2/12/81	2/9/81		

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

February 23, 1981

Docket No. 50-298

Mr. J. M. Pilant, Director
Licensing & Quality Assurance
Nebraska Public Power District
P. O. Box 499
Columbus, Nebraska 68501

Dear Mr. Pilant:

The Commission has issued the enclosed Amendment No.68 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 letters of January 30, 1980, October 15, 1980 and January 6, 1981.

These changes to the Technical Specifications involve administrative changes and incorporation of certain of the TMI-2 Lessons Learned Category "A" requirements. The Category "A" requirements concern (1) Emergency Power Supply/Inadequate Core Cooling, (2) Valve Position Indication, (3) Containment Isolation, (4) Shift Technical Advisor Augmentation, (5) Integrity of Systems Outside Containment, and (6) Iodine Monitoring.

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

Sincerely,

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No.68
2. Safety Evaluation
3. Notice

cc w/encls:
See next page

cc:

Mr. G. D. Watson, General Counsel
Nebraska Public Power District
P. O. Box 499
Columbus, Nebraska 68601

Mr. Arthur C. Gehr, Attorney
Snell & Wilmer
3100 Valley Center
Phoenix, Arizona 85073

Cooper Nuclear Station
ATTN: Mr. L. Lessor
Station Superintendent
P. O. Box 98
Brownville, Nebraska 68321

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

Director
Nebraska Dept. of Environmental Control
P. O. Box 94877, State House Station
Lincoln, Nebraska 68509

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

Director, Criteria and Standards
Division
Office of Radiation Programs (ANR-460)
U. S. Environmental Protection Agency
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region VII Office
ATTN: EIS COORDINATOR
324 East 11th Street
Kansas City, MO 64106

Mr. Dennis Dubois
USNRC
Resident Inspector
P. O. Box 446
Nebraska City, NA 68410

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.68
License No. DPR-46

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

   A. The submittals by Nebraska Public Power District dated January 30 and October 15, 1980 and January 6, 1981, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the submittals, 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.68 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

8103100414

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

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Dated: February 23, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 68

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Remove the following pages of the Appendix "A" Technical Specifications and replace with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
iii	iii
4	4
55	55
60	60
71	71
80	80
87	87
102	102
136	136
150	150
174	174
219a	219a
229	229
235a	235a

TABLE OF CONTENTS (cont'd)

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS	Page No.
3.14 Fire Detection System	4.14	216b
3.15 Fire Suppression Water System	4.15	216b
3.16 Spray and/or Sprinkler System (Fire Protection)	4.16	216e
3.17 Carbon Dioxide System	4.17	216f
3.18 Fire Hose Stations	4.18	216g
3.19 Fire Barrier Penetration Fire Seals	4.19	216h
5.0 MAJOR DESIGN FEATURES		217 - 218
6.0 ADMINISTRATIVE CONTROLS		219 - 237
6.1 Organization		219
6.2 Review and Audit		220
6.3 Station Operating Procedures		226
6.4 Actions to be Taken in the Event of Occurrences Specified in Section 6.7.2.A.		227
6.5 Actions to be Taken if a Safety Limit is Exceeded		227
6.6 Station Operating Records		228
6.7 Station Reporting Requirements		230
6.8 Environmental Qualification		235a
6.9 Systems Integrity Monitoring Program		235a
6.10 Iodine Monitoring Program		235a

Amendment No. 18, 41, 42, 68

K. Limiting Safety System Setting (LSSS) - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represent a margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.

L. Mode - The reactor mode is established by the mode selector-switch. The modes include refuel, run, shutdown and startup/hot standby which are defined as follows:

1. Refuel Mode - The reactor is in the refuel mode when the mode switch is in the refuel mode position. When the mode switch is in the refuel position, the refueling interlocks are in service.

2. Run Mode - In this mode the reactor system pressure is at or above 825 psig and the reactor protection system is energized with APRM protection (excluding the 15% high flux trip) and RSM interlocks in service.

3. Shutdown Mode - The reactor is in the shutdown mode when the reactor mode switch is in the shutdown mode position.

4. Startup/Hot Standby - In this mode the reactor protection scram trips initiated by the main steam line isolation valve closure are bypassed when reactor pressure is less than 1000 psig, the low pressure main steam line isolation valve closure trip is bypassed, the reactor protection system is energized with APRM (15% SCRAM) and IRM neutron monitoring system trips and control rod withdrawal interlocks in service.

M. Operable - A system or component shall be considered operable when it is capable of performing its intended function in its required manner.

N. Operating - Operating means that a system or component is performing its intended functions in its required manner.

O. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

P. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connected to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

2. At least one door in each air lock is closed and sealed.

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

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required When Component Operability Is Not Assured
RHR Pump Low Flow	RHR-dPIS-125 A & B	≥2500 gpm	1	A
Time Delays	RHR-TDR-K45, 1A&1B	4.25≤T≤5.75 min.	1	A
RHR Pump Start Time Delay	RHR-TDR-K75A & K70B RHR-TDR-K75B & K70A	4.5≤T≤5.5 Sec. ≤.5 sec.	1 1	A A
RHR Heat Exchanger Bypass T.D.	RHR-TDR-K93, A & B	1.8≤T≤2.2 min.	1	B
RHR Crosstie Valve Position	RHR-LMS-2	Valve Not closed	(3)	E
Bus 1A Low Volt. Aux. Relay	27 X 3/1A	Loss of Voltage	1	B
Bus 1B Low Volt. Aux. Relay	27 X 3/1B	Loss of Voltage	1	B
Bus 1F Low Volt. Aux. Relays	27 X 1/1F 27 X 2/1F	Loss of Voltage Loss of Voltage	1 1	B B
Bus 1G Low Volt. Aux. Relays	27 X 1/1G 27 X 2/1G	Loss of Voltage Loss of Voltage	1	B
Pump Discharge Line	CM-PS-266 CM-PS-270	≥5 psig ≥15 psig	(3) (3)	D D
Emergency Buses Undervoltage Relays (degraded voltage)	27/1F-2, 27/1FA-2 27/1G-2, 27/1GB-2 27/ET-2	3600 ±5% 8 second ±2 sec. time delay	2 2 1	B B B
Emergency Buses Loss of Voltage Relays	27/1F-1, 27/1FA-1, 27/1G-1, 27/1GB-1, 27/ET-1	2900V ±5% 5 second ±1 sec. delay	1	B
Emergency Buses Under-Voltage Relays Timers	27X7/1F, 27X7/1G, 27X10/1G	10 second ±2 sec.	1	B

NOTES FOR TABLE 3.2.B

1. When any ECCS system is required to be operable, there shall be two operable trip systems except as noted. If a requirement of the fourth column is reduced by one, the indicated action shall be taken. If the same function is inoperable in more than one trip system or the fourth column reduced by more than one, action B shall be taken.

   Action:

   A. Repair in 24 hours. If the function is not operable in 24 hours, take action B.

   B. Declare the system or component inoperable.

   C. Immediately take action B until power is verified on the trip system.

   D. The high point vent shall be vented weekly upon failure of PS 73A or B, PS 266, PS 268, PS 269, PS 270.

   E. Repair as soon as possible. It does not directly effect system operations.

2. In only one trip system.

3. Not considered in a trip system.

4. Requires one channel from each physical location in the steam line space.

5. One relay senses each phase of MCC-S and the LO relay is a transfer permissive relay.

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

	Item	Item I.D. No.	Functional Test Freq.	Calibration Freq.	Instrument Check
Instrumentation					
1.	Drywell High Pressure	PC-PS-101, A,B,C & D	Once/Month (1)	Once/3 Months	None
2.	Reactor Vessel Shroud Level	NBI-LITS-73, A & B #1	Once/Month (1)	Once/3 Months	Once/Day
3.	Reactor Low Pressure	RR-PS-128 A & B	Once/Month (1)	Once/3 Months	None
4.	Reactor Low Pressure	NBI-PS-52 A & C NBI-PIS-52 B & D	Once/Month (1)	Once/3 Months	None
5.	Drywell Press.-Containment Spray	PC-PS-119, A,B,C & D	Once/Month (1)	Once/3 Months	None
6.	RHR Pump Discharge Press.	RHR-PS-120, A,B,C & D	Once/Month (1)	Once/3 Months	None
7.	RHR Pump Discharge Press.	RHR-PS-105, A,B,C & D	Once/Month (1)	Once/3 Months	None
8.	RHR Pump Low Flow Switch	RHR-dPIS-125 A & B	Once/Month (1)	Once 3 Months	None
9.	RHR Pump Start Time Delay	RHR-TDR-K70, A & B	Once/Month (1)	Once/Oper. Cycle	None
10.	RHR Pump Start Time Delay	RHR-TDR-K75, A & B	Once/Month (1)	Once/Oper. Cycle	None
11.	RHR Heat Exchanger Bypass T.D.	RHR-TDR-K93, A & B	Once/Month (1)	Once/Oper. Cycle	None
12.	RHR Cross Tie Valve Position	RHR-LMS-2	Once/Month (1)	N.A.	
13.	Low Voltage Relays	27 X 3/1A	(7)		None
14.	Low Voltage Relays	27 X 3/1B	(7)		None
15.	Low Voltage Relays	27 x 2/1F, 27 X 2/1G	(7)		None
16.	Low Voltage Relays	27 X 1/1F, 27 X (1)/1G	(7)		None
17.	Pump Disch. Line Press. Low	CM-PS-266, CM-PS-270	Once/3 Months	Once/3 Months	None
18.	Emergency buses Undervoltage Relays (Degraded Voltage)	27/1F-2, 27/1FA-2, 27/1G-2, 27/1GB-2, 27/ET-2	Once/Month	Once/18 Months	Once/12 hrs.
19.	Emergency Buses Loss of Voltage Relays	27/1F-1, 17/1FA-1, 27/1G-1, 27/1GB-1, 17/ET-1	Once/Month	Once/18 Months	Once/12 hrs.
20.	Emergency Buses Undervoltage Relays Timers	27X7/1F, 27X7/1G, 27X10/1G	Once/Month	Once/18 Months	None

COOPER NUCLEAR STATION
TABLE 4.2.F
PRIMARY CONTAINMENT SURVEILLANCE INSTRUMENTATION
TEST AND CALIBRATION FREQUENCIES

Instrument	Instrument I.D. No.	Calibration Frequency	Instrument Check
Reactor Water Level	NBI-LI-85A	Once/6 Months	Each Shift
	NBI-LI-85B	Once/6 Months	Each Shift
Reactor Pressure	RFC-PI-90A	Once/6 Months	Each Shift
	RFC-PI-90B	Once/6 Months	Each Shift
Drywell Pressure	PC-PI-512A	Once/6 Months	Each Shift
	PC-PR-512B	Once/6 Months	Each Shift
Drywell Temperature	PC-TR-503	Once/6 Months	Each Shift
	PC-TI-505	Once/6 Months	Each Shift
Suppression Chamber Air Temperature	PC-TR-21A	Once/6 Months	Each Shift
	PC-TR-23, Ch. 1 & 2	Once/6 Months	Each Shift
Suppression Chamber Water Temperature	PC-TR-21B	Once/6 Months	Each Shift
	PC-TR-22, Ch. 1 & 2	Once/6 Months	Each Shift
Suppression Chamber Water Level	PC-LI-10	Once/6 Months	Each Shift
	PC-LR-11	Once/6 Months	Each Shift
	PC-LI-12	Once/6 Months	Each Shift
	PC-LI-13	Once/6 Months	Each Shift
Suppression Chamber Pressure	PC-PR-20	Once/6 Months	Each Shift
Control Rod Position	N.A.	N.A.	Each Shift
Neutron Monitoring (APRM)	N.A.	Once/Week	Each Shift
Torus to Drywell Differential Pressure	PC-dPR-20	Once/6 Months	Each Shift
Suppression Chamber/ Drywell Pressure (ΔP)	PC-PR-20/513 (2)	Once/6 Months	Each Shift

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.

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.

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. ble. Any deviation in readings will initiate an early recalibration, thereby maintaining the quality of the instrument readings.

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.

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 the Environmental Technical Specifications. Upon sensing a high discharge level, an isolation signal is generated which closes of radwaste discharge valve. The set point is adjustable to compensate for variable isotopic discharges and dilution flow rates.

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.

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.

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.

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 sytem 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.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.D Safety and Relief Valves	4.6.D Safety and Relief Valves
1. During reactor power operating conditions and prior to reactor startup from a Cold Condition, or whenever reactor coolant pressure is greater than atmospheric and temperature greater than 212°F, all three safety valves and the safety modes of all relief valves shall be operable, except as specified in 3.6.D.2.	1. Approximately half of the safety valves and relief valves shall be checked or replaced with bench checked valves once per operating cycle. All valves will be tested every two cycles. The set point of the safety valves shall be as specified in Specification 2.2.
2.	2. At least one of the relief valves shall be disassembled and inspected each refueling outage.
a. From and after the date that the safety valve function of one relief valve is made or found to be inoperable, continued reactor operation is permissible only during the succeeding thirty days unless such valve function is sooner made operable.	3. The integrity of the relief safety valve bellows on any three stage valve shall be continuously monitored.
b. From and after the date that the safety valve function of two relief valves is made or found to be inoperable, continued reactor operation is permissible only during the succeeding seven days unless such valve function is sooner made operable.	4. The operability of the bellows monitoring system shall be demonstrated once every three months when three stage valves are installed.
3. If Specification 3.6.D.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure shall be reduced to a cool shutdown condition within 24 hours.	5. Once per operating cycle, with the reactor pressure ≥ 100 psig, each relief valve shall be manually opened until the main turbine bypass valves have closed to compensate for relief valve opening.
4. From and after the date that position indication on any one relief valve is made or found to be inoperable, continued reactor operation is permissible only during the succeeding thirty days unless such valve position indication is sooner made operable.	6. a. Operability of the relief valve position indicating pressure switches and the safety valve position indicating thermocouples shall be demonstrated once per operating cycle. b. An Instrument Check of the safety and relief valve position indicating devices shall be performed monthly.

The relief and safety valves are bench tested every second operating cycle to ensure that their set points are within the ± 1 percent tolerance. Additionally, once per operating cycle, each relief valve is tested manually with reactor pressure above 100 psig to demonstrate its ability to pass steam.

The requirements established above apply when the nuclear system can be pressurized above ambient conditions. These requirements are applicable at nuclear system pressures below normal operating pressures because abnormal operational transients could possibly start at these conditions such that eventual overpressure relief would be needed. However, these transients are much less severe, in terms of pressure, than starting at rated conditions. The valves need not be functional when the vessel head is removed, since the nuclear system cannot be pressurized.

The position indicating pressure switches for the relief valves and the thermocouples for the safety valves serve as a diagnostic aid to the operator in the event of a safety/relief valve failure. If position indication is lost, alternate means are available to the operator to determine if a safety valve is leaking.

E. Jet Pumps

Failure of a jet pump nozzle assembly hold down mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design bases double-ended line break. Therefore, if a failure occurs, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ± 5%, the flow rates in both recirculation loops will be verified by Control Room monitoring instruments. If the two flow rate values do not differ by more than 10%, riser and nozzle assembly integrity has been verified. If they do differ by 10% or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10% or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the plant shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115% to 120% for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3% to 6%) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

A nozzle-riser system failure could also generate the coincident failure of a

TABLE 3.7.4 (Page 2)

PRIMARY CONTAINMENT TESTABLE ISOLATION VALVES

PEN. NO.	VALVE NUMBERS	TEST MEDIA
X-39A	RHR-MO-26A and RHR-MO-31A, Drywell Spray Header Supply	Air
X-39B	RHR-MO-26B and RHR-MO-31B, Drywell Spray Header Supply	Air
X-41	RRV-740AV and RRV-741AV, Reactor Water Sample Line	Air
X-42	SLC-12CV and SLC-13CV, Standby Liquid Control	Air
X-205	PC-233MV and PC-237AV, Purge and Vent Supply to Torus	Air
X-205	PC-13CV and PC-243AV, Torus Vacuum Relief	Air
X-205	PC-14CV and PC-244AV, Torus Vacuum Relief	Air
X-205	MV-1303 and MV-1304, ACAD Supply to Torus	Air
X-210A	RCIC-MO-27 and RCIC-13CV, RCIC Minimum Flow Line	Air
X-210A	RHR-MO-21A, RHR to Torus	Air
X-210A	RHR-MO-16A, RHR-10CV, and RHR-12CV, RHR Minimum Flow Line	Air
X-210B	RHR-MO-21B, RHR to Torus	Air
X-210B	HPCI-17CV and HPCI-MO-25, HPCI Minimum Flow Line	Air
X-210B	RHR-MO-16B, RHR-11CV, and RHR-13CV, RHR Minimum Flow Line	Air
X-210A and 211A	RHR-MO-34A, RHR-MO-38A, and RHR-MO-39A, RHR to Torus	Air
X-210B and 211B	RHR-MO-34B, RHR-MO-38B, and RHR-MO-39B, RHR to Torus	Air
X-212	RCIC-15CV and RCIC-37, RCIC Turbine Exhaust	Air
X-214	HPCI-15CV and HPCI-44, HPCI Turbine Exhaust	Air
X-214	HPCI-AO-70 and HPCI-AO-71, HPCI Turbine Exhaust Drain	Air
X-214	RHR-MO-166A and RHR-MO-167A RHR Heat Exch. Vent	Air
X-214	RHR-MO-166B and RHR-MO-167B RHR Heat Exch. Vent	Air
X-220	PC-230MV and PC-245AV, Purge and Vent Exhaust from Torus	Air
X-221	RCIC-12CV and RCIC-42, RCIC Vacuum Line	Air
X-222	HPCI-50 and HPCI-16CV, HPCI Turbine Drain	Air

G. A Fire Brigade of at least 5 members shall be maintained at all times. This excludes the 3 members of the minimum shift crew necessary for safe shutdowns, and other personnel required for other essential functions during a fire emergency. Three fire Brigade members shall be from the Operations Department and 2 support members may be from other departments inclusive of Security personnel.

Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of Fire Brigade members provided immediate action is taken to restore the Fire Brigade to within the minimum requirements.

H. In order to perform the function of accident assessment an engineer from the normal plant engineering staff shall be assigned to each shift during reactor operation. If the lack of qualified engineers necessitates, an additional senior reactor operator assigned to each shift may substitute in the performance of the accident assessment function. This requirement is effective until January 1, 1981.

6.1.4 The minimum qualifications, training, replacement training, and retraining of plant personnel at the time of fuel loading or appointment to the active position shall meet the requirements as described in the American National Standards Institute N-18.1-1971, "Selection and Training of Personnel for Nuclear Power Plants". The Assistant to Station Superintendent qualifications shall comply with Section 4.2 of ANSI-N18.1-1971. The Chemistry and Health Physics Supervisor shall meet or exceed the qualifications of Regulatory Guide 1.8, Sept. 1975; personnel qualification equivalency as stated in the Regulatory Guide may be proposed in selected cases. The minimum frequency of the retraining program shall be every two years. The training program shall be under the direction of a designated member of the plant staff.

A. A training program for the fire brigade will be maintained under the direction of the plant training coordinator and shall meet or exceed the requirements of Section 27 of the NFPA Code 1976, except for Fire Brigade training sessions which shall be held at least quarterly.

The training program requirements will be provided by a qualified fire protection engineer, thru the Risk Manager.

usage evaluation per the ASME Boiler and Pressure Vessel Code Section III was performed[1] for the conditions defined in the design specificaton. The locations to be monitored shall be:

a. The feedwater nozzles

b. The shell at or near the waterline

c. The flange studs

2. Monitoring, Recording, Evaluating, and Reporting

a. Operational transients that occur during plant operations will, at least annually, be reviewed and compared to the transient conditions defined in the component stress report for the locations listed in 1 above, and used as a basis for the existing fatigue analysis.

b. The number of transients which are comparable to or more severe than the transient evaluated in the stress report Code fatigue usage calculations will be recorded in an operating log book. For those transients which are more severe, available data, such as the metal and fluid temperatures, pressures, flow rates, and other conditions will be recorded in the log book.

c. The number of transient events that exceed the design specification quantity and the number of transient events with a severity greater than that included in the existing Code fatigue usage calculations shall be added. When this sum exceeds the predicated number of design condition events by twenty-five[2], a fatigue usage evaluation of such events will be performed for the affected portion of the RCPB.

H. Records of individual plant staff members showing qualifications, training and retraining.

6.6.3 Records and logs relating to the following items shall be kept for two years.

A. The test results, in units of microcuries, for leak tests of sources performed pursuant to Specification 3.8.A.

B. Records of annual physical inventories verifying accountability of the sources on record.

---

1. See paragraph N-415.2, ASME Section III, 1965 Edition.

2. The Code rules permit exclusion of twenty-five (25) stress cycles from secondary stress and fatigue usage evaluation. (See paragraphs N-412(t)(3) and N-417.10(f) of the Summer 1968 Addenda to ASME Section III, 1968 Edition.)

6.8 Environmental Qualification

A. By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of License DPR-46 dated October 24, 1980.

B. By no later than December 1, 1980, complete and auditible records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.9 Systems Integrity Monitoring Program

A program shall be established to reduce leakage from systems outside the primary containment that would or could contain highly radioactive fluids during a serious accident to as low as practical levels. This program shall include provisions establishing preventive maintenance and periodic visual inspection requirements, and leak testing requirements for each system at a frequency not to exceed refueling cycle intervals.

6.10 Iodine Monitoring Program

A program shall be established to ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include training of personnel, procedures for monitoring and provisions for maintenance of sampling and analysis equipment.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT NO. 68 TO FACILITY OPERATING LICENSE NO. DPR-46

COOPER NUCLEAR STATION

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

I. INTRODUCTION

By letters dated January 30, 1980, October 15, 1980, and January 6, 1981, Nebraska Public Power District (the licensee) has requested changes to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. The changes involve administrative changes and the incorporation of certain of the TMI-2 Lessons Learned Category "A" requirements in direct response to the NRC staff's letter dated July 2, 1980.

II. BACKGROUND INFORMATION

The changes 1 thru 4, 6 and 7 in the January 30, 1980 letter addressed by this amendment involve administrative changes to the technical specifications to correct oversights in previous amendments and inconsistencies resulting from previously approved station modifications and changes in station reporting requirements and therefore require no further evaluation. Item 5, relating to MAPLHGR, has been addressed in Amendment 67 issued January 30, 1981.

By our letter dated September 13, 1979, we issued to all operating nuclear power plants requirements established as a result of our review of the TMI-2 accident. Certain of these requirements, designated Lessons Learned Category "A" requirements, were to have been completed by the licensee prior to any operation subsequent to January 1, 1980. Our evaluation of the licensee's compliance with these Category "A" items was attached to our letter to Nebraska Public Power District dated April 10, 1980.

In order to provide reasonable assurance that operating reactor facilities are maintained within the limits determined acceptable following the implementation of the TMI-2 Lessons Learned Category "A" items, we requested that licensees amend their TSs to incorporate additional Limiting Conditions of Operation and Surveillance Requirements, as appropriate. This request was transmitted to all licensees on July 2, 1980. Included therein were model specifications that we had determined to be acceptable. The licensee's October 15, 1980 application is in direct response to our request. However, subsequent discussions with the NRC staff necessitated

8103100417

TS revisions in the areas of Valve Position Indication, Shift Technical Advisor, System Integrity Program, and Improved Iodine Measurements Program which were provided by the licensee in their January 6, 1981 amended submittal letter. Each of the issues identified by NRC staff and the licensee's response is discussed in the evaluation below.

III. EVALUATION

1. Emergency Power Supply/Inadequate Core Cooling

As we have indicated, water level instrumentation is important to post-accident monitoring in Boiling Water Reactors (BWRs) and surveillance of this instrumentation should be performed. The licensee's response to this request stated that the current surveillance requirements for the reactor water level instrumentation at Cooper Station meet or exceed our guidance.

We have reviewed the current specifications (contained in Tables 3.2F (page 65) and 4.2F (page 80) and determined that water level instrumentation is included. The specifications provide ACTION statements for inoperable instrument channels. Surveillance requirements for instrument checks and calibration are also included. The frequency of surveillance meets or exceeds our guidelines. Based on this review, we conclude that no changes are required to satisfy our request.

2. Valve Position Indication

Our requirements for installation of a reliable position indicating system for relief and safety valves was based on the need to provide the operator with a diagnostic aid to reduce the ambiguity between indications that might indicate either an open relief/safety valve or a small line break. Such a system did not need to be safety grade provided that backup methods of determining valve position are available.

The licensee's request would add to the TSs for the relief valves both a primary indicating system (pressure switches) and a secondary indicating system (thermocouples); and to the safety valves a thermocouple indicating system for primary and backup indication. Actions for the condition of inoperability of both the primary and backup detector channels for the relief valves have been specified. The safety valves discharge into the drywell. If the position indicators for these valves are inoperative drywell pressure and temperature instrumentation, which are sensitive to safety valve leakage, provide a backup means of monitoring the position of these valves. These instruments are covered by a LCO which insures operability. Therefore, no additional actions

for the condition of inoperability of the safety valve thermocouple indicating system has to be included in the TSs. Additionally surveillance requirements have been included for both the safety and relief valves. Based on our review we find the licensee's recommended changes satisfy our guidelines and are acceptable.

3. Containment Isolation

Our request indicated that the specifications should include a Table of Containment Isolation Valves which reflect the diverse isolation signal requirement of this Lessons Learned issue.

The licensee's request stated that the current specifications include a requirement for diverse isolation signals and that no changes are required.

We have reviewed the current specifications Table 3.2A (pages 50, 51, 52, 52a and 52b), Table 3.2.D (page 63), Table 4.2A (pages 68 and 69), Table 4.7.D.1.a (page 166) and Table 3.7.1 (page 168). These Tables include a listing of valves, operability requirements for the containment isolation instrumentation, surveillance requirements, and actuation signals. Based on this review we have determined that the current specifications satisfy our request and that no changes are necessary.

4. Shift Technical Advisor (STA)

Our request indicated that the TSs related to minimum shift manning should be revised to reflect the augmentation of an STA. The STA function includes both accident and operating experience assessment.

The licensee proposed in a letter from Nebraska Public Power District (J. Pilant) to NRC (H. Denton) dated January 11, 1980, that shift manning will be augmented by assigning an engineer from the normal plant engineering staff to each shift to satisfy the accident assessment function of the STA. At the NRR/IE site visit of March 26 and 27, 1980, the licensee indicated that he would like to exercise the option of assigning an additional SRO to the minimum shift manning to perform this function should the lack of availability of qualified engineers dictate.

Our evaluation of the adequacy of the licensee's actions to provide STAs, including the interim period from January 1, 1980 to January 1, 1981, was contained in our letter dated April 10, 1980. That evaluation concluded that it was acceptable for the interim period to use, as STAs, qualified normal plant engineering staff or SRO's if necessary.

5. Integrity of Systems Outside Containment

Our letter dated July 2, 1980, indicated that the license should be amended by adding a license condition related to a Systems Integrity Measurements Program. Such a condition would require the licensee to effect an appropriate program to eliminate or prevent the release of significant amounts of radioactivity to the environment via leakage from engineered safety systems and auxiliary systems, which are located outside reactor containment.

The licensee's October 15, 1980, application did not address this issue. However, in discussions between members of our staff and the licensee's representatives on December 16, 1980 it was indicated that they did not object to including such provisions. They suggested that they be incorporated into the Administrative Controls Section of the specifications. These were included in the licensees January 6, 1981 amended submittal. Accordingly, we have determined that our request has been satisfied.

6. Iodine Monitoring

Our letter dated July 2, 1980, indicated that the license should be amended by adding a license condition related to iodine monitoring. Such a condition would require the licensee to effect a program which would ensure the capability to determine the airborne iodine concentration in areas requiring personnel access under accident conditions.

The licensee's October 15, 1980 application did not address this issue. However, in discussions between members of our staff and the licensee's representatives on December 16, 1980, it was indicated that they did not object to including such provisions as part of the Administrative Controls Section of the TSs. These were included in the licensees January 6, 1981 amended submittal. Accordingly, we have determined that our request has been satisfied.

IV. ENVIRONMENTAL CONSIDERATIONS

We have determined that the amendment does not involve 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.

V. CONCLUSIONS

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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: February 23, 1981

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. 68 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 its date of issuance.

These changes to the Technical Specifications involve administrative changes and incorporation of certain of the TMI-2 Lessons Learned Category "A" requirements. The Category "A" requirements concern (1) Emergency Power Supply/Inadequate Core Cooling, (2) Valve Position Indication, (3) Containment Isolation, (4) Shift Technical Advisor Augmentation, (5) Integrity of Systems Outside Containment, and (6) Iodine Monitoring.

The licensee's submittals comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. 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

8103100423

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 licensee's submittals dated January 30 and October 15, 1980 and January 6, 1981, (2) Amendment No. 68 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 68305. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this February 23, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing