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FEBRUARY 29 1980

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. 61 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 January 14, 1980.

This amendment changes the Technical Specifications to allow the count rate in the Source Range Monitor channels to drop below 3 counts per second when the entire core is being removed or replaced.

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

Sincerely,

Original Signed By
T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 61
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

concurrence and FR notice only

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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. 61
License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District (the licensee) dated January 14, 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. 61, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

8008240275

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

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Dated: February 29, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 61

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

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

Remove	Insert
5	5
5A	5A
61	61
---	62a
86	86
94a	94a
203 - 205	203 - 205a
206 - 209a	206 - 209b

3. All automatic containment isolation valves are operable or de-activated in the isolated position.

4. All blind flanges and manways are closed.

Q. Rated Power - Rated power refers to operation at a reactor power of 2381 megawatts thermal. This is also termed 100% power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated neutron flux, and rated nuclear system pressure refer to the values of these parameters when the reactor is at rated power. Design power, the power to which the safety analysis applies, is 105% of rated power, which corresponds to 2486 megawatts thermal.

R. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup/Hot Standby" or "Run" position with the reactor critical and above 1% rated power.

S. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

T. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant after that refueling.

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

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

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

2. The standby gas treatment system is operable.

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

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

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

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

X. Spiral Reload - Pertains to the spiral reloading of the core with fuel, at least 50% of which has previously accumulated a minimum exposure of 1000 MWD/T.

Y. Surveillance Frequency -

Surveillance requirements shall be applicable during the operational conditions associated with individual LCO's unless otherwise stated in an individual Surveillance Requirement.

Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval.

b. A total maximum combined interval time for any 3 consecutive surveillance intervals not to exceed 3.25 times the specified interval.

Performance of a Surveillance Requirement within the specified time interval shall constitute compliance with operability requirements for an LCO unless otherwise required by the specification.

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

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

Function	Trip Level Setting	Minimum Number Of Operable Instrument Channels/Trip System(5)
APRM Upscale (Flow Bias)	$\leq (0.66W + 42\%)\left[\frac{FRP}{MFLPD}\right]$(2)	2(1)
APRM Upscale (Startup)	$\leq$ 12%	2(1)
APRM Downscale (9)	$\geq$ 2.5%	2(1)
APRM Inoperative	(10b)	2(1)
RBM Upscale (Flow Bias)	$\leq$ (0.66W + 39%) (2)	1
RBM Downscale (9)	$\geq$ 2.5%	1
RBM Inoperative	(10c)	1
IRM Upscale (8)	$\leq$ 108/125 of Full Scale	3(1)
IRM Downscale (3)(8)	$\geq$ 2.5%	3(1)
IRM Detector Not Full In (8)		3(1)
IRM Inoperative (8)	(10a)	3(1)
SRM Upscale (8)	$\leq 1 \times 10^5$ Counts/Second	1(1)(6)
SRM Detector Not Full In (4)(8)	($\geq$ 100 cps)	1(1)(6)
SRM Inoperative (8)	(10a)	1(1)(6)
Flow Bias Comparator	$\leq$ 10% Difference In Recirc. Flows	1
Flow Bias Upscale/Inop.	$\leq$ 110% Recirc. Flow	1
SRM Downscale (8)(7)	$\geq$ 3 Counts/Second (11)	1(1)(6)

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

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 volum high level components have only one logic channel and are not required for safety.

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 of 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 CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.3 (Cont'd)	4.3 (Cont'd)
B. Control Rods	B. Control Rods
1. Each control rod shall be coupled to its drive or completely inserted and the control rod directional control valves disarmed electrically. This requirement does not apply in the refuel condition when the reactor is vented. Two or more control rod drives may be removed as long as Specification 3.10.A.5 or 3.10.A.6 is met.	1. The coupling integrity shall be verified for each withdrawn control rod as follows:
	a. When a rod is withdrawn the first time after each refueling outage or after maintenance, observe discernible response of the nuclear instrumentation and rod position indication. However, for initial rods when response is not discernible, subsequent exercising of these rods after the reactor is above 30% power shall be performed to verify instrumentation response.
	b. When the rod is fully withdrawn the first time after each refueling outage or after maintenance, observe that the drive does not go to the overtravel position.

LIMITING CONDITIONS FOR OPERATION

3.10 CORE ALTERATIONS

Applicability

Applies to the fuel handling and core reactivity limitations.

Objective

To ensure that core reactivity is within the capability of the control rods and to prevent criticality during refueling.

Specification

A. Refueling Interlocks

1. The reactor mode switch shall be locked in the "Refuel" position during core alterations and the refueling interlocks shall be operable except as specified in 3.10.A.5 and 3.10.A.6 below.

2. Fuel shall not be loaded into the reactor core unless all control rods are fully inserted or unless the spiral unload/reload technique is used.

SURVEILLANCE REQUIREMENTS

4.10 CORE ALTERATIONS

Applicability

Applies to the periodic testing of those interlocks and instrumentation used during refueling and core alterations.

Objective

To verify the operability of instrumentation and interlocks used in refueling and core alterations.

Specification

A. Refueling Interlocks

1. Prior to any fuel handling with the head off the reactor vessel, the refueling interlocks shall be functionally tested. They shall be tested at weekly intervals thereafter until no longer required. They shall also be tested following any repair work associated with the interlocks.

2. Prior to performing control rod or control rod drive maintenance on a control cell without removing fuel assemblies, a shutdown margin demonstation shall have been performed for the existing core configuration as per Specification 4.3.A.1; additionally, the refueling interlocks shall be operable for all other control rods.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.A (Cont'd)	4.10.A (Cont'd)
3. The fuel grapple hoist load switch shall be set at ≤ 650 lbs.	3. Whenever the reactor is in the refuel mode and rod block interlocks are being bypassed for core unloading, one licensed operator and one member of the reactor engineering staff will verify that all fuel has been removed before the corresponding control rod is withdrawn.
4. If the frame-mounted auxiliary hoist, the monorail-mounted auxiliary hoist, or the service platform hoist is to be used for handling fuel with the head off the reactor vessel, the load limit switch on the hoist to be used shall be set at ≤ 400 lbs.	4. Following the withdrawal and bypassing of a control rod, two licensed operators will verify that the interlock bypassed is on the correct control rod.
5. A maximum of two nonadjacent control rods may be withdrawn from the core for the purpose of performing control rod and/or control rod drive maintenance, provided the following conditions are satisfied:	5. Prior to loading fuel in a control cell (using the spiral reload technique), the control room operator and a licensed operator and a member of the reactor engineering staff on the refueling floor shall verify that the control rod is inserted in the cell to be loaded.
a. The reactor mode switch shall be locked in the "refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed for one of the control rods on which maintenance is being performed. All other refueling interlocks shall be operable.	
b. A sufficient number of control rods shall be operable so that the core can be made subcritical with the strongest operable control rod fully withdrawn and all other operable control rods fully inserted, or all directional control valves for remaining control rods shall be disarmed electrically and sufficient margin to criticality shall be demonstrated.	
c. If maintenance is to be performed on two control rod drives, they must be separated by more than two control cells in any direction.	
d. An appropriate number of SRM's are available as defined in specification 3.10.B.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.A (Cont'd)	4.10 (Cont'd)
6. Any number of control rods may be withdrawn or removed from the reactor core providing the following conditions are satisfied:	
a. The reactor mode switch is locked in the "refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be operable.	
B. Core Monitoring	B. Core Monitoring
During core alterations two SRM's shall be operable, one in the core quadrant where fuel or control rods are being moved and one in an adjacent quadrant. For an SRM to be considered operable, the following conditions shall be satisfied:	Prior to making any alterations to the core, the SRM's shall be functionally tested and checked for neutron response. Thereafter, while required to be operable, the SRM's will be checked daily for response (or every 12 hours until 3 cps is attained if the spiral reload technique is being used).
1. The SRM shall be inserted to the normal operating level. (Use of special moveable, dunking type detectors during initial fuel loading and major core alterations in place of normal detectors is permissible as long as the detector is connected to the normal SRM circuit.)	
2. Operable SRM's shall have a minimum of 3 cps except as specified in 3 and 4 below.	
3. Prior to spiral unloading, the SRM's shall have an initial count rate of 3 cps. During spiral unloading, the count rate on the SRM's may drop below 3 cps.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.B (Cont'd)	4.10 (Cont'd)
4. During spiral reload, SRM operability will be verified by using a portable external source every 12 hours until the required amount of fuel is loaded to maintain 3 cps. As an alternative to the above, two fuel assemblies will be loaded in different cells containing control blades around each SRM to obtain the required 3 cps. Until these two assemblies have been loaded, the 3 cps requirement is not necessary.	
C. Spent Fuel Pool Water Level	C. Spent Fuel Pool Water Level
Whenever irradiated fuel is stored in the spent fuel pool, the pool water level shall be maintained at or above 8½' above the top of the fuel.	When irradiated fuel is stored in the spent fuel pool, the water level shall be recorded daily.
D. Time Limitation	
Irradiated fuel shall not be handled in or above the reactor prior to 24 hours after reactor shutdown.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10 (Cont'd)	4.10 (Cont'd)
E. Spent Fuel Cask Handling	E. Spent Fuel Cask Handling
1. Fuel cask handling above the 931' level of the Reactor Building will be done in the RESTRICTED MODE only except as specified in 3.10.E.2.	1. Prior to fuel cask handling operations the redundant crane including the rope, hooks, slings, shackles and other operating mechanisms will be inspected. The rope will be replaced if any of the following conditions exist: a. Twelve (12) randomly distributed broken wires in one lay or four (4) broken wires in one strand of one rope lay. b. Wear of one-third the original diameter of outside individual wire. c. Kinking, crushing, or any other damage resulting in distortion of the rope. d. Evidence of any type of heat damage. e. Reductions from nominal diameter of more than 1/16 inch for a rope diameter from 7/8" to 1¼" inclusive.
2. Fuel cask handling in other than the RESTRICTED MODE will be permitted in emergency or equipment failure situations only to the extent necessary to get the cask to the closest acceptable stable location.	2. Prior to operations in the RESTRICTED MODE: a. the controlled area limit switches will be tested; b. the "two-block" limit switches will be tested; c. the "inching hoist" controls will be tested.
3. Operation with a failed controlled area limit switch is permissible for 48 hours providing an operator is on the refueling floor to assure the crane is operated within the restricted zone painted on the floor.	3. The empty spent fuel cask will be lifted free of all support by a maximum of 1 foot and left hanging for 5 minutes prior to any series of fuel cask handling operations.
4. Spent fuel casks weighing in excess of 140,000 lbs. shall not be handled.	

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3.10 BASES

A. Refueling Interlocks

The refueling interlocks are designed to back up procedural core reactivity controls during refueling operations. The interlocks prevent an inadvertent criticality during refueling operations (when the reactivity potential of the core is being altered) by restricting the movement of control rods and the operation of refueling equipment. The interlocks include circuitry which senses the condition of the refueling equipment and the control rods. Depending on the sensed condition, interlocks are actuated which prevent the movement of the refueling equipment or withdrawal of control rods (rod block). Circuitry is provided which senses the following conditions:

1. All rods inserted.

2. Refueling platform positioned near or over the core.

3. Refueling platform hoists are fuel-loaded (fuel grapple, frame-mounted hoist, monorail-mounted hoist).

4. Fuel grapple not full up.

5. Service platform hoist fuel-loaded.

6. One rod withdrawn.

When the mode switch is in the "Refuel" position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. When the mode switch is in the refuel position, only one control rod can be withdrawn. The refueling interlocks, in combination with core nuclear design and refueling procedures, limit the probability of an inadvertent criticality. The nuclear characteristics of the core assure that the reactor is subcritical even when the highest worth control rod is fully withdrawn. The combination of refueling interlocks for control rods and the refueling platform provide redundant methods of preventing inadvertent criticality even after procedural violations. The interlocks on hoists provide yet another method of avoiding inadvertent criticality.

Fuel handling is normally conducted with the fuel grapple hoist. The total load on this hoist when the interlock is required consists of the weight of the fuel grapple and the fuel assembly. This total is approximately 980 lbs., in comparison to the load-trip setting of 650 lbs. Provisions have also been made to allow fuel handling with either of the three auxiliary hoists and still maintain the refueling interlocks. The 400 lb. load-trip setting on these hoists is adequate to trip the interlock when one of the more than 600 lb. fuel bundles is being handled.

During certain periods, it is desirable to perform maintenance on two control rods and/or control rod drives at the same time. The maintenance is performed with the mode switch in the "refuel" position to provide the refueling interlocks normally available during refueling operations. In order to withdraw a second control rod after withdrawal of the first rod, it is necessary to bypass the refueling interlock on the first control rod which prevents more than one control rod from being withdrawn at the same time. The requirement that an adequate shutdown margin be demonstrated or that all remaining control rods have their directional control valves electrically disarmed ensures that inadvertent criticality cannot occur during this maintenance. The adequacy of the shutdown margin is verified by demonstrating that the core is shut down by a margin of 0.38 percent Δk with the strongest operable control rod fully withdrawn, or that at least 0.38 percent Δk shutdown margin is available if the remaining control rods have had their directional control valves disarmed. Disarming the directional control valves does not inhibit control rod scram capability.

Specification 3.10.A.6 allows unloading of a significant portion of the reactor core. This operation is performed with the mode switch in the "refuel" position to provide the refueling interlocks normally available during refueling operations. In order to withdraw more than one control rod, it is necessary to bypass the refueling interlock on each withdrawn control rod which prevents more than one control rod from being withdrawn at a time. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core before the interlock can be bypassed ensures that withdrawal of another control rod does not result in inadvertent criticality. Prior to removal of the last two diagonal fuel assemblies, a double blade guide shall be inserted to properly support the control rod and fuel assemblies. After removal of the last two fuel assemblies and withdrawal of the control rod, the double blade guide may be removed.

Each control rod provides primary reactivity control for the fuel assemblies in the cell associated with that control rod. Thus, removal of an entire cell (fuel assemblies plus control rod) results in a lower reactivity potential of the core. The requirements for SRM operability during these core alterations assure sufficient core monitoring.

To minimize the possibility of loading fuel into a cell containing no control rod, when refueling interlock input signals are bypassed, it is required that the control room operator and a licensed operator and a member of the reactor engineering staff on the refueling floor verify that the control rod is inserted in the cell to be loaded. Prior to insertion of the control rod, it shall be verified that a double blade guide was placed in the cell to be loaded to properly support the control rod and fuel assemblies.

B. Core Monitoring

The SRM's are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRM's in or adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. The requirement of ≥ 3 counts per second provides assurance that neutron flux is being monitored and insures that startup is conducted only if the source range flux level is above the minimum assumed in the control rod drop accident.

A spiral unloading pattern is one by which the fuel in the outermost cells (four fuel bundles surrounding a control blade) is removed first. Unloading continues by removing the remaining outermost fuel cell by cell. The center cell will be the last removed. Spiral reloading is the reverse of unloading. Spiral unloading and reloading will preclude the creation of flux traps (moderator filled cavities surrounded on all sides by fuel).

During spiral unloading, the SRM's shall have an initial count rate of >3 cps with all rods fully inserted. The count rate will diminish during fuel removal. After all the fuel is removed from a cell, the control rod may be withdrawn in that cell. After the control rod is withdrawn, the refueling interlock will be bypassed on that control rod. Following the withdrawal and bypassing of the control rod, two licensed operators will verify that the interlock bypassed is on the correct control rod. Once the control rod is withdrawn, it will be valved out of service. The refueling interlocks will prevent the withdrawal of another control rod unless the control rod just withdrawn from the unloaded cell is bypassed.

Under this special condition of complete spiral core unloading, it is expected that the count rate of the SRM's will drop below 3 cps before all of the fuel is unloaded. Since there will be no reactivity additions, a lower number of counts will not present a hazard. When all of the fuel has been removed to the spent fuel storage pool, the SRM's will no longer be required. Requiring the SRM's to be operational prior to fuel removal assures that the SRM's are operable and can be relied on even when the count rate may go below 3 cps.

During spiral reload, SRM operability will be verified by using a portable external source every 12 hours until the required amount of fuel is loaded to maintain 3 cps. As an alternative to the above, two fuel assemblies will be loaded in different cells containing control blades around each SRM to obtain the required 3 cps. Until these two assemblies have been loaded, the 3 cps requirement is not necessary.

C. Spent Fuel Pool Water Level

To assure that there is adequate water to shield and cool the irradiated fuel assemblies stored in the pool, a minimum pool water level is established. The minimum water level of 8½' above the top of the fuel is established because it provides adequate shielding and is well above the level to assure adequate cooling.

3.10 BASES (Cont'd)

D. Time Limitation

The radiological consequences of a fuel handling accident are based upon the accident occurring at least 24 hours after reactor shutdown.

E. Spent Fuel Cask Handling

The operation of the redundant crane in the Restricted Mode during fuel cask handling operations assures that the cask remains within the controlled area once it has been removed from its transport vehicle (i.e., once it is above the 931' elevation). Handling of the cask on the Refueling Floor in the Unrestricted Mode is allowed only in the case of equipment failures or emergency conditions when the cask is already suspended. The Unrestricted Mode of operation is allowed only to the extent necessary to get the cask to a suitable stationary position so the required repairs can be made. Operation with a failed controlled area microswitch will be allowed for a 48-hour period providing an Operator is on the floor in addition to the crane operator to assure that the cask handling is limited to the controlled area as marked on the floor. This will allow adequate time to make repairs but still will not restrict cask handling operations unduly.

4.10 BASES

A. Refueling Interlocks

Complete functional testing of all refueling interlocks before any refueling outage will provide positive indication that the interlocks operate in the situations for which they were designed. By loading each hoist with a weight equal to the fuel assembly, positioning the refueling platform and withdrawing control rods, the interlocks can be subjected to valid operational tests. Where redundancy is provided in the logic circuitry, tests can be performed to assure that each redundant logic element can independently perform its functions.

B. Core Monitoring

Requiring the SRM's to be functionally tested prior to any core alteration assures that the SRM's will be operable at the start of that alteration. The daily response check (or 12-hour check for spiral reload) of the SRM's ensures their continued operability.

F. Spent Fuel Cask Handling

The Surveillance Requirements specified assure that the redundant crane is adequately inspected in accordance with the accepted ANSI Standard (B.30.2.0) and manufacturer's recommendations to determine that the equipment is in satisfactory condition. The testing of the controlled area limit switches assures that the crane operation will be limited to the designated area in the Restricted Mode of operation. The test of the "two-block" limit switch assures the power to the hoisting motor will be interrupted before an actual "two-blocking" incident can occur. The test of the inching hoist assures that this mode of load control is available when required.

4.10 BASES (Cont'd)

Requiring the lifting and holding of the cask for 5 minutes during the initial lift of each series of cask handling operations puts a load test on the entire crane lifting mechanism as well as the braking system.

Performing this test when the cask is being lifted initially from the cask car assures that the system is operable prior to lifting the load to an excessive height.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 61 TO LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

1.0 Introduction

By letter dated January 14, 1980,[1] Nebraska Public Power District (the licensee) has requested an amendment to the Technical Specifications for the Cooper Nuclear Station. The effect of the amendment would be to allow the count rate in the Source Range Monitor (SRM) channels to drop below 3 counts per second (cps) when the entire reactor core is being removed or replaced. The present Technical Specifications require that a count rate of at least 3 cps be maintained whenever one or more fuel assemblies are present in the core.

2.0 Discussion

Cooper Technical Specifications require that all control blades be inserted into the core during fuel loading. This is no problem during normal refueling and control blade drive maintenance since only one core cell (defined as a control blade plus the four adjacent fuel assemblies) is worked on at any given time. However, a removal of the entire core would require all the fuel to be removed before any control blade was removed. This is not possible unless the plant has a full complement of control blade guides. These guides are needed to provide lateral support to control blades in defueled cells.

Cooper Technical Specifications, in addition to the control blade insertion requirement, require that the SRM count rates be above 3 cps whenever more than one control blade is removed.

The safety issues are two-fold if control blades are to be removed with fuel still in the core. First, the intermediate fuel and control blade arrays must be subcritical at all times, even if the highest worth blade is withdrawn. Second, there must be adequate monitoring of neutron flux levels during the core alterations.

During any core alteration, and especially during core loading, it is necessary to monitor flux levels. In this manner, even in the highly unlikely event of multiple operator errors, there is reasonable assurance that any approach to criticality would be detected in time to halt operations.

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The minimum count rate requirement in the Technical Specifications accomplishes three safety functions: (1) it assures the presence of some neutrons in the core, (2) it provides assurance that the analog portion of the SRM channels is operable, and (3) it provides assurance that the SRM detectors are close enough to the array of fuel assemblies to monitor core flux levels.

Unloading and reloading of the entire core leads to some difficulty with this minimum count rate requirement. When only a small number of assemblies are present within the core, the SRM count rate will drop below the minimum due to the small number of neutrons being produced, and due to attenuation of these neutrons in the water and control blades separating the fuel from the SRM detectors. Past practice has been to connect temporary "dunking" chambers to the SRM channels in place of the normal detectors, and to locate these detectors near the fuel.

Besides being operationally inconvenient, dunking chambers suffer from signal variations due to their lack of fixed geometry. Moreover, the use of dunking chambers increases the risk of loose objects being dropped into the vessel.

3.0 Evaluation

3.1 Subcriticality of the Intermediate Arrays

The proposed Technical Specifications would only allow spiral unloading and reloading of the core. In such a program, only cells on the edge of the array are unloaded or reloaded. No imbedded cavities or major peripheral concavities would be permitted. In such a case, the neutron multiplication factor of the intermediate arrays must be less than or equal to that of the fully loaded core. Since the shutdown margin of the fully loaded core is well assured by other specifications, we find the proposed change to be acceptable from the point of view of shutdown margin.

By telephone discussion, the licensee has modified his request to include changing Specification 4.10.A.2 to facilitate maintenance of single control rods. The change in Specification 4.10.A.2 allows maintenance of a single control rod to be performed provided (a) all other rods are fully inserted with their refueling interlocks operable and (b) the required shutdown margin of 0.38 percent has been demonstrated for the current core configuration. This restriction assures that no more than one control rod will be withdrawn during single control rod maintenance and that the core will remain subcritical by a sufficient margin during such maintenance. We find this change, also, to be acceptable from the point of view of shutdown margin.

3.2 Flux Monitoring

3.2.1 Minimum Flux in the Core

A multiplying medium with no neutrons present forms the basis for an accident scenario in which reactivity is gradually but inadvertently added until the medium is highly supercritical. No neutron flux will be evident since there are no neutrons present to be multiplied. The introduction of some neutrons at this point would cause the core to undergo a sudden power burst, rather than a gradual startup, with no warning from the nuclear instrumentation.

This scenario is of great concern when loading fresh fuel, but is of lesser concern for exposed fuel. Exposed fuel continuously produces neutrons by spontaneous fission of certain plutonium isotopes, photofission, and photodisintegration of deuterium in the moderator. This neutron production in exposed fuel is normally great enough to meet the 3 cps minimum for a full core after a refueling outage with the lumped neutron sources removed.

Thus, there is assurance that a minimum flux level will be present as long as some exposed fuel is present. We therefore find the proposed amendment to be acceptable from the point of view of minimum flux provided the words "spiral reload" are interpreted to mean "reload of the core with fuel, at least 50% of which has previously accumulated a minimum exposure of 1000 MWD/T." We have modified the licensee's Technical Specification to include this definition in this amendment, and he has agreed to this change. We do not find the amendment to be applicable to the loading of a new core containing only fresh fuel. Such a loading must use lumped neutron sources and dunking chambers to meet the normal 3 cps minimum count rate.

3.2.2 SRM Operability

The Technical Specifications normally require a functional check of the SRM channels, including a check of neutron response, prior to making any alteration to the core and daily thereafter. This would be sufficient for core unloading and reloading, except that the more extensive fuel handling operations involved imply a greater possibility of SRM mechanical failure. The Cooper Technical Specifications have been modified to increase the functional check frequency from once/day to once/12 hours while using the spiral reload technique to compensate. Testing the neutron response is accomplished by using an external source or by loading two fuel assemblies in different cells containing control blades around each SRM to obtain the required 3 cps. This should being the count rate up to 3 cps and thus continuously verify operability. Therefore, we find this alternative to be acceptable.

3.2.3 Flux Attenuation

The four SRM detectors are located, one per quadrant, roughly half a core radius from the center. Although these are incore detectors and thus very sensitive when the reactor is fully loaded, they lose some of their effectiveness when the reactor is partially defueled and the detectors are located some distance from the array of remaining fuel.

GE's spent fuel pool studies have shown[2] that 16 or more fuel assemblies (i.e., four or more control cells) must be loaded together before criticality is possible. Moreover, in a spiral reloading, the cell centered on the central blade must be loaded first. Given this, we have examined the Cooper core layout and determined that the worst case configuration for monitoring purposes consists of four loaded cells, centered on blades 22-27, 22-31, 26-27 and 26-31. This array is two control cells (i.e., about two feet) away from the nearest SRM detectors. (All other arrays containing 16 or more assemblies are closer to a detector.) We have previously examined the sensitivity loss in such a case on another docket,[3] and found it to be at most one decade of sensitivity (i.e., about one fifth of the SRMs logarithmic scale). As in Reference 3, we find this to be acceptable.

4.0 Summary

We have examined the safety issues and found the proposed amendment to be acceptable provided that the spiral reload consists of 50% previously exposed fuel as discussed herein (Section 3.2.1).

5.0 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.

6.0 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 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: February 29, 1980

References

1. Letter, J. M. Pilant (Nebraska Public Power District) to T. A. Ippolito (NRC), dated January 14, 1980.

2. General Electric Standard Safety Analysis Report, 251-GESSAR, Section 4.3.2.7, pg. 4.3-27.

3. "Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 27 to Facility Operating License No. DPR-63," Docket No. 50-220, enclosed with letter, T. A. Ippolito (NRC) to D. P. Dise (Niagara Mohawk Power Corporation), dated March 2, 1979.

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. 61 to Facility Operating License No. DPR-46, issued to Nebraska Public Power District, 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.

The amendment changes the Technical Specifications to allow the count rate in the Source Range Monitor channels to drop below 3 counts per second when the entire reactor core is being removed or replaced.

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 issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated January 14, 1980, (2) Amendment No. 61 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 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 Operating Reactors.

Dated at Bethesda, Maryland, this 29th day of February 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors