MAY 2 3 1975

Docket No. 50-324

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Carolina Power & Light Company
ATTN: Mr. J. A. Jones
Executive Vice President
336 Fayetteville Street
P. O. Box 1551
Raleigh, North Carolina 27602

Amendment No. 2
Change No. 2
License No. DPR-62

Gentlemen:

By letter dated April 25, 1975, you indicated that due to the delay in achieving power operation for the Brunswick Steam Electric Plant, Unit 2, the neutron sources have decayed to a level that certain technical specifications cannot be satisfied. Specifically, the minimum count rate of ≥ 3 counts per second (cps) for the Source Range Monitor (SRM) may not be satisfied unless sufficient power operation has occurred to regenerate the neutron sources.

In lieu of replacement of the startup neutron sources, you requested a temporary change to the SRM count rate requirements of the Technical Specifications. Specifically, you requested that the minimum count rate be reduced from 3 to 0.3 cps for the first core load when the source is at low strength.

We have completed our review of your request and the associated analyses. We note that a similar request was made by the licensee and granted by the NRC for the Cooper Nuclear Station. We conclude that your requested change is acceptable and have issued this change as Amendment No. 2 to DPR-62, a copy of which is provided in Enclosure 1. This amendment authorizes the Technical Specification (Appendix A) change which follows. The underlined items represent the change.

On page 3.2-40, Table 3.2-11 item 1.c under Remarks add (<u>For initial core, when source is at low strength</u>) and under Trip Setting add (<u>≥ 0.3 cps</u>).

On page 3.3-5 add the following sentence to 3.3.B4 and 4.3.B4 "<u>The minimum count rate may be reduced to 0.3 cps for the first core load when the source is at low strength.</u>"

bcc: J.R. Buchanan, ORNL
T. B. Abernathy, DTIE
A. Rosenthal, ASLAB
N. H. Goodrich, ASLBP

OFFICE ➤						
SURNAME ➤						
DATE ➤						

On page 3.3-13 Bases 3.3.B4 and 4.3.B4 add the following sentence after the fourth sentence. "For the initial core, when the startup source strength is low, the minimum requirement will be 0.3 counts per second, which assures any transient would begin at or above 10^{-12} of rated power."

On page 3.10-2 under 3.10.B2 add the following (0.3 cps for the initial core when the source is at low strength).

On page 3.10-4 under Bases: 3.10-B. insert the following in the third sentence after the requirement of three counts per second (0.3 cps for the initial core when the source strength is low).

The requested change is authorized on the basis that the potential consequences of a control rod drop or uncontrolled rod withdrawal accident are no more severe than the currently acceptable design bases. Furthermore, the reactivity characteristics of the core are known, permitting use of conservative operating procedures to minimize the probability of too rapid an approach to criticality.

Accordingly, pursuant to 10 CFR, Part 50, Section 50-59 we have concluded that: (1) because the change does not involve a significant increase in the probability or consequence of accidents previously considered, it 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 with this change; and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this Amendment No. 2 will not be inimical to the common defense and security or to the health and safety of the public.

Our Safety Evaluation supporting this change to the Technical Specification is provided in Enclosure 2 for your information and use. A copy of a related Federal Register Notice which has been forwarded to the Office of the Federal Register for publication is provided in Enclosure 3.

Sincerely,

/s/

Walter R. Butler, Chief
Light Water Reactors Branch 1-2
Division of Reactor Licensing

Enclosures:
1. Amendment 2 - DPR-62
2. Staff Evaluation
3. Federal Register Notice

cc: See page 3

OFFICE ➤	L:LWR 1-2	OELD	L:LWR 1-2			
SURNAME ➤	RPowell/red		WButler WB			
DATE ➤	7791 5/9/75	5/22/75	5/23/75			

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

CAROLINA POWER AND LIGHT COMPANY
DOCKET NO. 50-324
BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 2
License No. DPR-62

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

   A. The application for amendment by Carolina Power and Light Company (the licensee) dated April 25, 1975 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; and

   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.

2. Accordingly, the license is amended by the changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2C(2) of Facility License No. DPR-62 is hereby amended to read as follows:

"2C(2) Technical Specifications

The Technical Specifications contained in Appendix A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 2."

AMERICAN REVOLUTION BICENTENNIAL 1776-1976

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

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Chief
Light Water Reactors Branch 1-2
Division of Reactor Licensing

Attachment:
Change No. 2
Technical Specifications

Date of Issuance: MAY 2 3 1975

MAY 23 1975

SAFETY EVALUATION OF REQUEST FOR REDUCTION OF MINIMUM SOURCE RANGE MONITOR COUNT RATE FOR THE FIRST CORE LOAD OF THE BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

Introduction

By letter dated April 25, 1975, the Carolina Power and Light Company (CP&L) requested an amendment to the Brunswick Steam Electric Plant (BSEP) Unit 2, operating license DPR-62 to reduce the Technical Specification (Appendix A) requirement for a minimum count rate of 3 cps on the source range monitor (SRM) to 0.3 cps for the first core load when the neutron source strength is low. CP&L has indicated that the startup neutron sources have decayed to a level which will not provide the 3 cps required on the SRM prior to startup of the reactor and have requested this limit of 3 cps be reduced to 0.3 cps for the first core until reactivation of the neutron sources has been accomplished.

The same problem of inadequate neutron source strength to meet the minimum count rate of 3 cps on the (SRM) occurred during the initial startup program for the Cooper Nuclear Station. The staff at that time reviewed and granted the reduction of minimum count rate from 3 cps to 0.3 cps until the neutron sources were reactivated.

Discussion

CP&L, in its April 25, 1975 request provided the analysis which considered the effect of a minimum count rate of 0.3 cps on the potential consequences of a control rod drop accident and a continuous rod withdrawal transient.

In the case of the continuous rod withdrawal transient during a reactor startup, the analysis given in the Final Safety Analysis Report (FSAR) was more severe than that which could be experienced in the planned startup, i.e., (1) the rod withdrawal would be initiated at a lower power level; (2) the maximum rod worth of an out of sequence rod is less than that assumed in the FSAR analysis, and (3) the Intermediate Range Monitor (IRM) 15% rated power trip would terminate the power transient at a lower power than that assumed in the FSAR analysis.

The rod drop accident was analized assuming that the neutron source for the core was only that core neutron level resulting from spontaneous fission. This change in the source level results in a slight increase in the control rod worth with a consequent increase in the peak fuel enthalpy of approximately 10 cal/gm for all cases analyzed. The results of the analysis show that with the rod sequence control system in operation, the control rod worth will be limited such that the 280 cal/gm design limit will never be reached.

We conclude based upon the analysis presented, that the consequences of a rod drop accident or as continuous rod withdrawal transient are less severe than those of the design basis.

In addition to the above considerations, the known operational characteristics of the reactor and procedural controls indicate that the proposed reduction of the SRM count rate limit to 0.3 cps does not involve a significant hazard consideration. The reactor has been brought critical in excess of 20 times during recent testing. Its reactivity characteristics are well known. Operating records indicate that the core becomes critical on the withdrawal of approximately 69 control rods. With an initial SRM count rate of 0.3 cps the count rate could be expected to exceed 3 cps prior to withdrawal of 53 control rods, the end of the B-3 rod sequence.

The SRMs read from 0.1 to 10^6 cps, with a noise level of less than 0.1 cps. Thus with a $\geq$ 0.3 cps limit the minimum signal to noise ratio will be $\geq$ 3.

Conclusion

We conclude, based upon the results of analyses of the control rod drop accident and the continuous rod withdrawal transient during reactor startup, that the consequences of these events are within acceptable safety limits with the initial SRM count rate $\geq$ 0.3 cps for the first core load of the BSEP, Unit 2. The experience with the core gained during startup operations and the procedural restrictions provide additional assurance. We conclude that the Technical Specification change requested in the CP&L letter of April 28, 1975, is acceptable and that the BSEP Unit 2 can be operated with this change with reasonable assurance that the health and safety of the public will not be endangered. We also conclude that the proposed change is not an unreviewed safety matter and does not involve a significant hazards consideration.

Ray Powell

Ray Powell, Project Manager
Light Water Reactors Branch 1-2
Division of Reactor Licensing

Walter R. Butler, Chief
Light Water Reactors Branch 1-2
Division of Reactor Licensing

Date MAY 2 3 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-324

CAROLINA POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 2 to Facility Operating License No. DPR-62, which was issued to Carolina Power & Light Company on December 27, 1974. Amendment No. 2 to DPR-62 revises the Technical Specifications for operation of the Brunswick Steam Electric Plant, Unit 2, located on the Cape Fear River, near Southport in Brunswick County, North Carolina. The amendment is effective as of its date of issuance.

The purpose of this amendment to the Technical Specifications is to permit a reduction in the minimum count rate (Source Range Monitor) to 0.3 cps when the neutron source strength is low.

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 is not required since the amendment does not involve a significant hazards consideration.

For further details with respect to this action, see: (1) the application for amendment, dated April 25, 1975; (2) Amendment No. 2 to License No. DPR-62, with Change No. 2; 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 Southport - Brunswick County Library, 109 W. Moore Street, Southport, North Carolina 28461.

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 Reactor Licensing.

Dated at Bethesda, Maryland, this 23nd day of May, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Chief
Light Water Reactors Branch 1-2
Division of Reactor Licensing

TABLE 4.2-10

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CSCS
AUTOMATIC DEPRESSURIZATION SYSTEM

	Trip Function	Functional Test	Calibration	Instrument Check
1.	High drywell pressure E11-PS-N010A,B,C,D	once/month	(1)	N/A
2.	Reactor low water level #3 B21-LIS-N031A,B,C,D	Same level instruments that are used for core spray initiation due to reactor low water level #3 and will be functionally tested and calibrated at same time		
3.	Time delay time B21-TDPU-K5A,B	once/operating cycle	(1)	N/A
4.	ADS trip system bus power monitor B21-K1A,B	once/month	N/A	N/A

ADS subsystem logic system functional test will be performed once/6 months cycle.

NOTES:

(1) When functional test shows the setpoints are out of specified limits, a calibration will be performed immediately.

TABLE 3.2-11

CONTROL ROD BLOCKS INITIATED FROM NEUTRON MONITORING SYSTEM

Trip Function	Minimum Number of Operable Instrument Channels(2)	Modes in Which Function Must Be Operable: Refuel	Startup	Run	Trip Setting	Remarks
1. Startup range monitor						
a. Upscale SRM channels A,B,C,D Relay C51-K4	3	X	X		$\leq 10^5$ cps	Bypass if mode switch in RUN or when IRM range switch on RANGE 8 or above.
b. Inoperative SRM channels A,B,C,D Relay C51-K1	3	X	X		(1)	Bypass if mode switch in RUN or when IRM range switch on RANGE 8 or above.
c. Downscale SRM channels A,B,C,D Relay C51-K2	3	X	X		$\geq$ 3 cps ($\geq$0.3 cps)	Bypass if mode switch in RUN or when IRM range switch on RANGES 3 or above. (For initial core when source is at low strength.)
d. Detector not in startup position SRM channels A,B,C,D Relay C51-K9A,B,C,D	3	X	X		Detector motor module unit switch LS-4 not closed (detector not full in)	Bypassed when the count rate is $\geq$ 100 cps IRM on RANGES 3 or above.
2. Intermediate range monitor						
a. Upscale IRM channels A through H, Relay C51-K52	6	X	X		$\leq$108/125 full scale	Bypassed in run mode.
b. Inoperative IRM channels A through H Relay C51-K54	6	X	X		(1)	Bypassed in run mode.

BSEP 1 & 2 Change #2

LIMITING CONDITIONS FOR OPERATION

3.3.B Control Rods (Cont'd)

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second. The minimum count rate may be reduced to 0.3 cps for the first core load when the source is at low strength.

Change #2

5. During reactor power operation with limiting control rod patterns, as determined by a Plant Engineer, either:

   a. Both RBM channels shall be operable; or

   b. Control rod withdrawal shall be blocked; or

   c. The operating power level shall be limited so that the MCHFR will remain above 1.0 assuming a single error that results in complete withdrawal of any single operable control rod.

6. In order to perform the required shutdown margin demonstrations subsequent to any fuel loading operations, to perform tests to verify shutdown margin due to inoperable control rod, or to perform control rod drive scram and/or friction testing and the initial startup test program, the relaxation of the following RSCS restraints is permitted. The sequence restraints imposed on control rod groups A_{12}, A_{34}, B_{12} or B_{34} may be removed for the test period by means of the individual rod position bypass switches.

Amendment 1

SURVEILLANCE REQUIREMENTS

4.3.B Control Rods (Cont'd)

4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second. The minimum count rate may be reduced to 0.3 cps for the first core load when the source is at low strength.

Change #2

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

6. Prior to control rod withdrawal for startup, verify the conformance to specification 3.3.B.3d before a rod may be bypassed in the RSCS. The requirements to allow use of the individual rod position bypass switches within rod groups A_{12}, A_{34}, B_{12}, or B_{34} of the RSCS during shutdown margin, scram time or friction testing and the initial startup test program are:

   (a) RWM operable as per specification 3.3.B.3C.

   (b) After the bypassing of the rods in the RSCS groups A_{12}, A_{34}, B_{12} or B_{34} for test purposes, it shall be demonstrated that movement of the rods in the 50 percent density to the preset power level range is blocked or limited to the single notch mode of withdrawal.

   (c) A second licensed operator shall verify the conformance to procedures and this Specification.

Amendment 1

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.3.C Scram Insertion Times	4.3.C Scram Insertion Times

3.3.C Scram Insertion Times

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids at time zero, of all operable control rods in the reactor power operation condition shall be no longer than:

Above 950 psig

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.0
90	3.5

2. The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two-by-two array shall be no longer than:

Above 950 psig

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.800

4.3.C Scram Insertion Times

1. After each refueling outage all operable fully withdrawn insequence rods shall be scram time tested during operational hydrostatic testing or during startup from the fully withdrawn position with the nuclear system pressure above 800 psig. This testing shall be completed prior to synchronizing the main turbine generator initially following restart of the plant. Prior to exceeding 40% of rated power, all untested operable control rods shall be tested as described above.

2. At 16 week intervals, 10 percent of the control rods capable of movement with control rod drive pressure shall be scram timed above 950 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

   If a scram occurs and scram time measurements are available from the scram timing processor, the above 16 week time interval is to start from date of scram.

   If a scheduled shutdown is planned near the midcycle period, at which time rod scram measurements will be taken for over 50 percent of the operable control rods, the above 16 week interval does not apply.

Amendment 1

BASES:

3.3.B and 4.3.B Control Rod Withdrawal (Cont'd)

4. The source range monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least three 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 from cold conditions. For the initial core, when the startup source strength is low, the minimum requirement will be 0.3 counts per second, which assures any transient would begin at or above 10^{-12} of rated power. 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 SRMs are provided as an added conservatism. Change #2

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

   During reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCHFRs less than 1.0. 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 a Plant Engineer to identify these limiting patterns and the

BASES:

3.3.B.5 and 4.3.B.5 Control Rod Withdrawal (Cont'd)

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.

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 MCHFR from becoming less than 1.0. The limiting power transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram (FSAR Figure 3.6.14) with the average response of all the drives as given in the above Specification, provide the required protection, and MCHFR remains greater than 1.0. The scram times for all control rods will be determined at the time of each refueling outage. The scram insertion times given in Specification 3.3.C for reactor pressures in excess of 950 psig, when met, insure that adequate insertion rates will result at all reactor pressures below 950 psig. The transient and accident analysis for the plant takes account of the slower scram insertion rates which are characteristic of the drives at certain reactor pressures below 950 psig.

D. Control Rod Accumulators

At reactor pressures in excess of 950 psig, even those control rods with inoperable accumulators will be able to meet required scram insertion times due to the action of reactor pressure. Thus, above this pressure, a control rod drive is not designated as inoperable when the associated accumulator is unavailable. It should also be noted that control rods can be driven in under all operating conditions without the use of the accumulator.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10 Core Alterations	4.10 Core Alterations
Applicability: Applies to the fuel handling and core reactivity limitations.	Applicability: Applies to the periodic testing of those interlocks and instrumentation used during refueling and core alterations.
Objective: To ensure that core reactivity is within the capability of the control rods and to prevent criticality during refueling.	Objective: To verify the operability of instrumentation and interlocks used in refueling and core alterations.
Specification:	Specification:
A. Refueling Interlocks The reactor mode switch shall be locked in the REFUEL position during core alterations and the refueling interlocks shall be operable.	A. Refueling Interlocks 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.
B. Core Monitoring During core alterations, two SRMs 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: 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).	B. Core Monitoring Prior to making any alterations to the core, the SRMs shall be functionally tested and checked for neutron response. Thereafter, while required to be operable, the SRMs will be checked daily for response.

Change #2

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.B Core Monitoring (Cont'd)	4.10.B Core Monitoring (Cont'd)
2. The SRM shall have a minimum of three cps with all rods fully inserted in the core. (0.3 cps for the initial core when the source is at low strength.)	
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 36'-9".	Whenever irradiated fuel is stored in the spent fuel pool, the water level shall be recorded daily.

BASES:

3.10 Core Alterations

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.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality.

The refueling interlocks reinforce operational procedures that prohibit taking the reactor critical under certain situations encountered during refueling operations by restricting the movement of control rods and the operation of refueling equipment.

The refueling 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.

BASES:

3.10.A Refueling Interlocks (Cont'd)

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.

B. Core Monitoring

The SRMs are provided to monitor the core during periods of plant shutdown and to guide the operator during refueling operations and plant startup. Requiring two operable SRMs 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 three counts per second (0.3 cps for the initial core when the source strength is low) 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.

Change #2

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 36'-9" is established because it would be significant change from the normal level (-1 foot) and is well above the level to assure adequate cooling.