March 21, 1995

Mr. John J. Barton
Vice President and Director
GPU Nuclear Corporation
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, NJ 08731

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M89741)

Dear Mr. Barton:

The Commission has issued the enclosed Amendment No. 178 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated June 22, 1994.

The amendment changes Technical Specification (TS) Sections 1.6, 3.2.A, 3.9.f.5 and 4.2.A which specify the Shutdown Margin requirements that ensure the reactor can be made subcritical and can be maintained sufficiently subcritical to preclude inadvertent criticality in any core condition. The amendment also includes a definition of Shutdown Margin, TS Section 1.45. Administrative changes to TS Sections 1.7 and 3.2.b.2(b) are also included to simplify definitions and eliminate unnecessary notes and references.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by:

Alexander W. Dromerick, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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Docket No. 50-219

Enclosures: 1. Amendment No. 178 to DPR-16
2. Safety Evaluation

cc w/encls: See next page

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SNorris	E. Kendrick

*SEE PREVIOUS CONCURRENCE

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NAME	SNorris	ADromerick:bf	PMcKee	CWoodhead	
DATE	03/20/95	03/20/95	03/21/95	03/09/95	03/ /95

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 21, 1995

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Vice President and Director
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Office of Nuclear Reactor Regulation

Docket No. 50-219

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cc w/encls: See next page

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

GPU NUCLEAR CORPORATION

AND

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 178
License No. DPR-16

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

   A. The application for amendment by GPU Nuclear Corporation, et al. (the licensee), dated June 22, 1994, 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-16 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 178, are hereby incorporated in the license. GPU Nuclear Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance, to be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Phillip F. McKee

Phillip F. McKee, Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 21, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 178

FACILITY OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
1.0-1	1.0-1
1.0-2	1.0-2
1.0-8	1.0-8
3.2-1	3.2-1
3.2-2	3.2-2
3.2-3	3.2-3
3.2-4	3.2-4
3.2-5	3.2-5
3.2-6	3.2-6
3.2-7	3.2-7
3.2-8	3.2-8
3.2-9	3.2-9
3.2-10	3.2-10
3.2-11	3.2-11
3.2-12	3.2-12
3.9-2	3.9-2
4.2-1	4.2-1
4.2-2	4.2-2
4.2-3	4.2-3
4.2-4	4.2-4

SECTION I
DEFINITIONS

The following frequently used terms are defined to aid in the uniform interpretation of the specifications.

1.1 OPERABLE-OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling of seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

A verification of operability is an administrative check, by examination of appropriate plant records (logs, surveillance test records) to determine that a system, subsystem, train, component or device is not inoperable. Such verification does not preclude the demonstration (testing) of a given system, subsystem, train, component or device to determine operability.

1.2 OPERATING

Operating means that a system or component is performing its required function.

1.3 POWER OPERATION

Power operation is any operation when the reactor is in the startup mode or run mode except when primary containment integrity is not required.

1.4 STARTUP MODE

The reactor is in the startup mode when the reactor mode switch is in the startup mode position. In this mode, the reactor protection system scram trips initiated by condenser low vacuum and main steam line isolation valve closure are bypassed when reactor pressure is less than 600 psig; the low pressure main steamline isolation valve closure is bypassed; the IRM trips for rod block and scram are operable; and the SRM trips for rod block are operable.

1.5 RUN MODE

The reactor is in the run mode when the reactor mode switch is in the run mode position. In this mode, the reactor protection system is energized with APRM protection and the control rod withdrawal interlocks are in service.

1.6 SHUTDOWN CONDITION

The reactor is in the SHUTDOWN CONDITION when there is fuel in the reactor vessel, the reactor is subcritical, all operable control rods are fully inserted, and the mode switch is in the shutdown mode position. In this position, a control rod block is initiated.

1.7 COLD SHUTDOWN CONDITION

The reactor is in the COLD SHUTDOWN CONDITION when the reactor is in the SHUTDOWN CONDITION, and (except during reactor vessel pressure testing), the reactor coolant system is maintained at less than 212°F and vented.

1.8 PLACE IN SHUTDOWN CONDITION

Proceed with and maintain an uninterrupted normal plant shutdown operation until the shutdown condition is met.

1.9 PLACE IN COLD SHUTDOWN CONDITION

Proceed with and maintain an uninterrupted normal plant shutdown operation until the cold shutdown condition is met.

1.10 PLACE IN ISOLATED CONDITION

Proceed with and maintain an uninterrupted normal isolation of the reactor from the turbine condenser system including closure of the main steam isolation valves.

1.11 REFUEL MODE

The reactor is in the refuel mode when the reactor mode switch is in the refuel mode position and there is fuel in the reactor vessel. In this mode the refueling platform interlocks are in operation.

1.12 REFUELING OUTAGE

For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled refueling outage. Following the first refueling outage, successive tests or surveillances shall be performed at least once per 24 months.

1.13 PRIMARY CONTAINMENT INTEGRITY

Primary containment integrity means that the drywell and adsorption chamber are closed and all of the following conditions are satisfied:

A. All non-automatic primary containment isolation valves which are not required to be open for plant operation are closed.

B. At least one door in the airlock is closed and sealed.

C. All automatic containment isolation valves specified in Table 3.5.2 are operable or are secured in the closed position.

D. All blind flanges and manways are closed.

1.43 CORE OPERATING LIMITS REPORT

The Oyster Creek CORE OPERATING LIMITS REPORT (COLR) is the document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.f. Plant operation within these operating limits is addressed in individual specifications.

1.44 LOCAL LINEAR HEAT GENERATION RATE

The LOCAL LINEAR HEAT GENERATION RATE (LLHGR) shall be applicable to a specific planar height and is equal to the AVERAGE PLANAR LINEAR GENERATION RATE (APLHGR) at the specified height multiplied by the local peaking factor at that height.

1.45 SHUTDOWN MARGIN (SDM)

SHUTDOWN MARGIN is the amount of reactivity by which the reactor would be subcritical when the control rod with the highest reactivity worth is fully withdrawn, all other operable control rods are fully inserted, all inoperable control rods are at their current position, reactor water temperature is 68°F, and the reactor fuel is xenon free. Determination of the control rod with the highest reactivity worth includes consideration of any inoperable control rods which are not fully inserted.

3.2 REACTIVITY CONTROL

Applicability: Applies to core reactivity and the operating status of the reactivity control systems for the reactor.

Objective: To assure reactivity control capability of the reactor.

Specification:

A. Core Reactivity

1. The SHUTDOWN MARGIN (SDM) under all operational conditions shall be equal to or greater than:

   (a) 0.38% delta k/k, with the highest worth control rod analytically determined; or

   (b) 0.28% delta k/k, with the highest worth control rod determined by test.

2. If one or more control rods are determined to be inoperable as defined in Specification 3.2.B.4 while in the STARTUP MODE or the RUN MODE, then a determination of whether Specification 3.2 A. is met must be made within 6 hours. If a determination cannot be made within the specified time period, then assume Specification 3.2 A.1 is not met.

3. If Specification 3.2.A.1 is not met while in the STARTUP Mode or the RUN MODE, meet Specification 3.2.A.1 within 6 hours or be in the SHUTDOWN CONDITION within the following 12 hours.

4. If Specification 3.2.A.1 is not met while in the SHUTDOWN CONDITION, or the COLD SHUTDOWN CONDITION, then:

   (a) Fully insert all insertable control rods within 1 hour, AND

   (b) Comply with the requirements of Specifications 3.2.C and 3.5.B.

5. If Specification 3.2.A.1 is not met while in the REFUEL MODE, then:

   (a) Immediately suspend CORE ALTERATIONS except for fuel assembly removal, AND

   (b) Immediately initiate action to fully insert all insertable control rods in control cells containing one or more fuel assemblies, AND

   (c) Comply with the requirements of Specifications 3.2.C and 3.5.B.

B. Control Rod System

1. The control rod drive housing support shall be in place during power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.2.A is met.

2. The Rod Worth Minimizer (RWM) shall be operable during each reactor startup until reactor power reaches 10% of rated power except as follows:

   (a) Should the RWM become inoperable after the first 12 rods have been withdrawn, the startup may continue provided that a second licensed operator verifies that the licensed operator at the reactor console is following the rod program.

   (b) Should the RWM be inoperable before a startup is commenced or before the first twelve rods are withdrawn, one startup during each calendar year may be performed without the RWM provided that the second licensed operator verifies that the licensed operator at the reactor console is following the rod program and provided that a reactor engineer from the Core Engineering Group also verifies that the rod program is being followed. A startup without the RWM as described in this subsection shall be reported in a special report to the Nuclear Regulatory Commission (NRC) within 30 days of the startup stating the reason for the failure of the RWM, the action taken to repair it and the schedule for completion of the repairs.

   Control rod withdrawal sequences shall be established with a banked position withdrawal sequence so that the rod drop accident design limit of 280 cal/gm is not exceeded. For control rod withdrawal sequences not in strict compliance to BPWS, the maximum in sequence rod worth shall be $\leq 1.0\% \Delta K$.

3. The average of the scram insertion times of all operable control rods shall be no greater than:

Rod Length Inserted (%)	Insertion Time (Seconds)
5	0.375
20	0.900
50	2.00
90	5.00

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 greater than:

Rod Length Inserted (%)	Insertion Time (Seconds)
5	0.398
20	0.954
50	2.120
90	5.300

Any four rod group may contain a control rod which is valved out of service provided the above requirements and Specification 3.2.A are met. Time zero shall be taken as the de-energization of the pilot scram valve solenoids.

4. Control rods which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure, the reactor shall be brought to a shutdown condition within 48 hours unless investigation demonstrates that the cause of the failure is not due to a failed control rod drive mechanism collet housing. Inoperable control rods shall be valved out of service, in such positions that Specification 3.2.A is met. In no case shall the number of rods valved out of service be greater than six during the power operation. If this specification is not met, the reactor shall be placed in the shutdown condition.

5. Control Rods shall not be withdrawn for approach to criticality unless at least two source range channels have an observed count rate equal to or greater than 3 counts per second.

C. Standby Liquid Control System

1. The standby liquid control system shall be operable at all times when the reactor is not shut down by the control rods such that Specification 3.2.A is met and except as provided in Specification 3.2.C.3.

2. The standby liquid control solution shall have a Boron-10 isotopic enrichment equal to or greater than 35 atom %, be maintained within the cross-hatched volume-concentration requirement area in Figure 3.2-1 and at a temperature not less than the temperature presented in Figure 3.2-2 at all times when the standby liquid control system is required to be operable.

3. (a) If one standby liquid control system pumping circuit becomes inoperable during the RUN mode and Specification 3.2.A is met, the reactor may remain in operation for a period not to exceed 7 days, provided the pump in the other circuit is verified daily to be operable, otherwise be in the Shutdown condition within 24 hours.

(b) If the solution is outside the cross-hatched volume-concentration area but within the shaded volume-concentration area of Figure 3.2-1, return the solution to the cross-hatched area within 7 days. If, after this time period, the requirement is still not met, submit a report to the NRC within 7 days advising them of plans to return the solution to the cross-hatched volume-concentration area.

(c) If the solution is outside the cross-hatched volume concentration area and outside the shaded volume-concentration area of Figure 3.2-1, return the solution to within the shaded volume-concentration area of Figure 3.2-1 or be in the Shutdown condition within 24 hours.

(d) If the solution temperature is less than the minimum shown in Figure 3.2-2, increase the temperature to greater than the minimum and verify the solution is within the shaded volume-concentration area of Figure 3.2-1 or be in the Shutdown condition within 24 hours.

(e) If the enrichment requirement of 3.2.C.2 is not met:

(1) Return the Boron-10 isotopic enrichment to greater than or equal to 35 atom % within 7 days of the receipt of the enrichment report. If, after this time period, the enrichment requirement is still not met, submit a report to the NRC within 7 days advising them of the plans to return the solution to greater than or equal to 35 atom % Boron-10 isotopic enrichment.

(2) A check shall be made to ensure that the sodium pentaborate solution meets the original design criteria by comparing the enrichment, concentration and volume to established criteria (Boron-10 equal to or greater than 82 pounds). If the sodium pentaborate solution does not meet the original criteria, be in the Shutdown condition within 24 hours.

D. <u>Reactivity Anomalies</u>

The difference between an observed and predicted control rod inventory shall not exceed the equivalent of one percent in reactivity. If this limit is exceeded and the discrepancy cannot be explained, the reactor shall be brought to the cold shutdown condition by normal orderly shutdown procedure. Operation shall not be permitted until the cause has been evaluated and appropriate corrective action has been completed. The NRC shall be notified within 24 hours of this situation in accordance with Specification 6.6.

Bases:

Limiting conditions of operation on core reactivity and the reactivity control systems are required to assure that the excess reactivity of the reactor core is controlled at all times. The conditions specified herein assure the capability to provide reactor shutdown from steady state and transient conditions and assure the capability of limiting reactivity insertion rates under accident conditions to values which do not jeopardize the reactor coolant system integrity or operability of required safety features.

The core reactivity limitation is required to assure the reactor can be shut down at any time when fuel is in the core. It is a restriction that must be incorporated into the design of the core fuel; it must be applied to the conditions resulting from core alterations; and it must be applied in determining the required operability of the core reactivity control devices. The basic criterion is that the core at any point in its operation be capable of being made subcritical in the cold, xenon-free condition with the operable control rod of highest worth fully withdrawn and all other operable rods fully inserted. At most times in core life, more than one control rod drive could fail mechanically and this criterion would still be met.

The SDM limit specified in Section 3.2.A.1 accounts for the uncertainty in the demonstration of SDM by testing. Separate SDM limits are provided for testing where the highest worth control rod is determined analytically or by measurement (Ref. 11). This is due to the reduced uncertainty in the SDM test when the highest worth control rod is determined by measurement. When SDM is demonstrated by calculations not associated with a test, additional margin must be added to the specified SDM limit to account for uncertainties in the calculation. To ensure adequate SDM during the design process, a design margin of 1.0% delta k is included to account for uncertainties in the design calculations (Ref. 11).

Inability to meet SDM limits in the STARTUP MODE or RUN MODE is most probably due to inoperable control rods. Reduced SDM is not considered an immediate threat to nuclear safety, therefore time is allowed for analysis to insure Specification 3.2 A. is met, and for repair before requiring the plant to undergo a transient to achieve the SHUTDOWN CONDITION. The allowed times of 6 hours for analysis and an additional 6 hours for repair, if 3.2.A.1 is not met, are considered reasonable while limiting the potential for further reduction in SDM or the occurrence of a transient.

If SDM cannot be restored, shutdown is required to minimize the potential for, and consequences of, an accident or malfunction of equipment important to safety. The allowed completion time of 12 hours is considered reasonable to achieve the SHUTDOWN CONDITION from full power in an orderly manner and without challenging plant systems.

Inability to meet SDM limits in the SHUTDOWN CONDITION or COLD SHUTDOWN CONDITION could be due to inoperable control rods, discovery of errors in the SDM analysis, or discovery of errors in previous CORE ALTERATIONS. Inserting control rods maximizes SDM and, since all operable control rods are required to be fully inserted in these conditions, should be able to be completed in 1 hour. The Standby Liquid Control System is allowed to be inoperable and Secondary Containment Integrity is allowed to be relaxed in these conditions when SDM limits are met. Therefore, they may not be available when the inability to meet SDM limits is recognized. The Standby Liquid Control System is needed to provide negative reactivity if control rods are not adequate to maintain the reactor subcritical. Secondary Containment Integrity is needed to provide means for control of potential radioactive releases.

Inability to meet SDM limits in the REFUEL MODE is most probably due to CORE ALTERATION errors. CORE ALTERATIONs are suspended to prevent further reduction in SDM. Fuel assembly removal and control rod insertion reduce total reactivity and are allowed in order to recover SDM. Control rods in control cells which do not contain fuel do not affect the reactivity of the core and therefore do not have to be inserted. The Standby Liquid Control System is allowed to be inoperable in this mode when SDM limits are met and, therefore, may not be available when the inability to meet SDM limits is recognized. The Standby Liquid Control System is needed to provide negative reactivity if control rods are not adequate to maintain the reactor subcritical. Secondary Containment Integrity is needed to provide means for control of potential radioactive releases.

Fuel bundles containing gadolinia as a burnable neutron absorber results in a core reactivity characteristic which increases with exposure, goes through a maximum and then decreases. Thus, it is possible that a core could be more reactive later in the cycle than at the beginning. Satisfaction of the above criterion can be demonstrated conveniently only at the time of refueling since it requires the core to be cold and xenon-free. The demonstration is designed to be done at these times and is such that if it is successful, the criterion is satisfied for the entire subsequent fuel cycle. The criterion will be satisfied by demonstrating Specification 4.2.A at the beginning of each fuel cycle with the core in the cold, xenon-free condition. This demonstration will include consideration for the calculated reactivity characteristic during the following operating cycle and the uncertainty in this calculation.

The control rod drive housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure (2). The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the reactor coolant system. The support is not required when no fuel is in the core since no nuclear consequences could occur in the absence of fuel.

The support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. The support is not required if all control rods are fully inserted since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod (3).

The Rod Worth Minimizer (4) provides automatic supervision of conformance to the specified control rod patterns. It serves as a back-up to procedural control of control rod worth. In the event that the RWM is out of service when required, a licensed operator can manually fulfill the control rod pattern conformance functions of the RWM in which case the normal procedural controls are backed up by independent procedural controls to assure conformance during control rod withdrawal. This allowance to perform a startup without the RWM is limited to once each calendar year to assure a high operability of the RWM which is preferred over procedural controls.

Control rod drop accident (RDA) results for plants using banked position withdrawal sequences (BPWS) show that in all cases the peak fuel enthalpy in an RDA would be much less than the 280 cal/gm design limit even with the maximum incremental rod worth. The BPWS is developed prior to initial operation of the unit following any refueling outage and the requirement that the operator follow the BPWS is supervised by the RWM or a second licensed operator. If it is necessary to deviate slightly from the BPWS sequence (i.e., due to an inoperable control rod) no further analysis is needed if the maximum incremental rod worth in the modified sequence is equal to or less than 1.0% delta K. An incremental control rod worth of less than or equal to 1.0% delta K will not result in a peak fuel enthalpy above the design limit of 280 cal/gm as documented in reference 10.

The BPWS limits the reactivity worths of control rods and together with the integral rod velocity limiters and the action of the control rod drive system limits potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. Method and basis for the rod drop accident analyses are documented in Reference 5.

The control rod system is designed to bring the reactor subcritical from a scram signal at a rate fast enough to prevent fuel damage. Scram reactivity curve for the transient analyses is calculated and evaluated with each reload core. In the analytical treatment of the transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typical time delay of about 210 milliseconds estimated from scram test results. Approximately the first 90 milliseconds of each of these time intervals result from the sensor and circuit delays when the pilot scram solenoid de-energizes. Approximately 120 milliseconds later, the control rod motion is estimated to actually begin.

However, 200 milliseconds is conservatively assumed for this time interval in the transient analyses and this is also included in the allowable scram insertion times of Specification 3.2.B.3. The specified limits provide sufficient scram capability to accommodate failure to scram of any one operable rod. This failure is in addition to any inoperable rods that exist in the core, provided that those inoperable rods met the core reactivity Specification 3.2.A.

Control rods (6) which cannot be moved with control rod drive pressure are clearly indicative of an abnormal operating condition on the affected rods and are, therefore, considered to be inoperable. Inoperable rods are valved out of service to fix their position in the core and assure predictable behavior. If the rod is fully inserted and then valved out of service, it is in a safe position of maximum contribution to shutdown reactivity. If it is valved out of service in a non-fully inserted position, that position is required to be consistent with the shutdown reactivity limitation stated in Specification 3.2.A, which assures the core can be shutdown at all times with control rods. Before a rod is valved out of service in a non-fully inserted position, an analysis is performed to insure Specification 3.2.A is met.

The number of rods permitted to be valved out of service could be many more than six allowed by the specification, particularly late in the operating cycle; however, the occurrence of more than six could be indicative of a generic problem and the reactor will be shut down. Also, if damage within the control rod drive mechanism and in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing and requiring increased surveillance after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings. Placing the reactor in the shutdown condition inserts the control rods and accomplishes the objective of the specifications on control rod operability. This operation is normally expected to be accomplished within eight hours.

The source range monitor (SRM) system (7) performs no automatic safety function. It does provide the operator with a visual indication of neutron level which is needed for knowledgeable and efficient reactor startup at low neutron levels. The results of the reactivity accidents are functions of the neutron flux. The requirement of at least 3 cps assures that any transient begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to critical using homogeneous patterns of scattered control rods.

The standby liquid control system is designed to bring the reactor to a cold shutdown condition from the full power steady state operating condition at any time in core life independent of the control rod system capabilities (8). If the reactor is shutdown by the control rod system and would be subcritical in its most reactive condition as required in Specification 3.2.A, there is no requirement for operability of this system. To bring the reactor from full power to cold shutdown, sufficient liquid control must be inserted to give a negative reactivity worth equal to the combined effects of rated coolant voids, fuel Doppler, xenon, samarium, and temperature change plus shutdown margin. This requires a Boron-10 concentration of 110 ppm in the reactor. An additional 25% Boron-10, which results in an average Boron-10 concentration in the reactor of 138 ppm, is inserted to provide margin for mixing uncertainties in the reactor. An amount of Boron-10 equal to or greater than 82 pounds will bring the reactor to cold shutdown.

The standby liquid control system is also required to meet 10 CFR 50.62 (Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants). The standby liquid control system must have the equivalent control capacity (injection rate) of 86 gpm at 13% by wt. natural sodium pentaborate for a 251" diameter reactor pressure vessel in order to satisfy 10 CFR 0.62 requirements. The equivalency requirement is fulfilled by a combination of concentration, Boron-10 enrichment and flow rate of sodium pentaborate solution. A minimum of 15.0 wt.% solution and 35 atom % Boron-10 enrichment at a 30 gpm pump flow rate satisfies the ATWS Rule (10 CFR 50.62) equivalency requirement and assures that the reactor is shutdown before unacceptable containment conditions develop.

The standby liquid control system is required to insert the solution within 120 minutes in order to override the rate of reactivity insertion due to cooldown of the reactor following the xenon peak, the 3737 gallons 5 wt. % point represents the allowable maximum volume-minimum concentration values which satisfy this requirement. Compliance with 10 CFR 50.62 (use of enriched boron) results in the cold shutdown B-10 concentration in the reactor, at the maximum concentration - minimum volume points chosen 19.6 wt. % 913 gallons, being injected in approximately 26 minutes. Thus, the system will insert the solution in the time interval of between 26-120 minutes.

The shaded area of Figure 3.2-1 represents the acceptable values of liquid control tank volume and solution concentration which assure that, with one 30 gpm liquid control pump, the reactor can be brought to the cold shutdown condition from a full power steady state operating condition at any time in core life independent of the control rod system capabilities. The cross-hatched area of Figure 3.2-1 represents the acceptable values of liquid control tank volume and solution concentration which assure that the equivalency requirements of 10 CFR 50.62 are satisfied. The maximum volume of 4213 gal is established by the tank capacity. The tank volume requirements include consideration for 137 gal of solution which is contained below the point where the pump takes suction from the tank and, therefore, cannot be inserted into the reactor.

The solution saturation temperature varies with the concentration of sodium pentaborate. The solution will be maintained at least 5°F above the saturation temperature to guard against precipitation. The 5°F margin is included in Figure 3.2-2. Temperature and liquid level alarms for the system are annunciated in the control room.

The acceptable time out of service for a standby liquid control system pumping circuit as well as other safety features is determined to be 10 days. However, the allotted time out of service for a standby liquid control system pumping circuit is conservatively set at 7 days in the specification. Systems are designed with redundancy to increase their availability and to provide backup if one of the components is temporarily out of service.

During each fuel cycle, excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity is indicated by the integrated worth of control rods inserted into the core, referred to as the control rod inventory in the core. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of actual rod inventory with expected inventory based on appropriately corrected past data. Experience at Oyster Creek and other operating BWR's indicates that the control rod inventory should be predictable to the equivalent of one percent in reactivity. Deviations beyond this magnitude would not be expected and would require thorough evaluation. One percent reactivity limit is considered safe since an insertion of this reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

<u>References</u>:

(1) FDSAR, Volume I, Section III-5.3.1
(2) FDSAR, Volume I, Section VI-3
(3) FDSAR, Volume I, Section III-5.2.1
(4) FDSAR, Volume I, Section VII-9
(5) NEDO-24195, General Electric Reload Fuel Application for Oyster Creek
(6) FDSAR, Volume I, Section III-5 and Volume II, Appendix B
(7) FDSAR, Volume I, Sections VII-4.2.2 and VII-4.3.1
(8) FDSAR, Volume I, Section VI-4
(9) FDSAR, Amendment No. 55, Section 2
(10) C. J. Paone, Banked Position Withdrawal Sequence, January 1988 (NEDO-21231)
(11) UFSAR, Volume 4, Section 4.3.2.4.1

Fig. 3.2-1 Sodium Pentaborate Solution Volume - Concentration Requirement

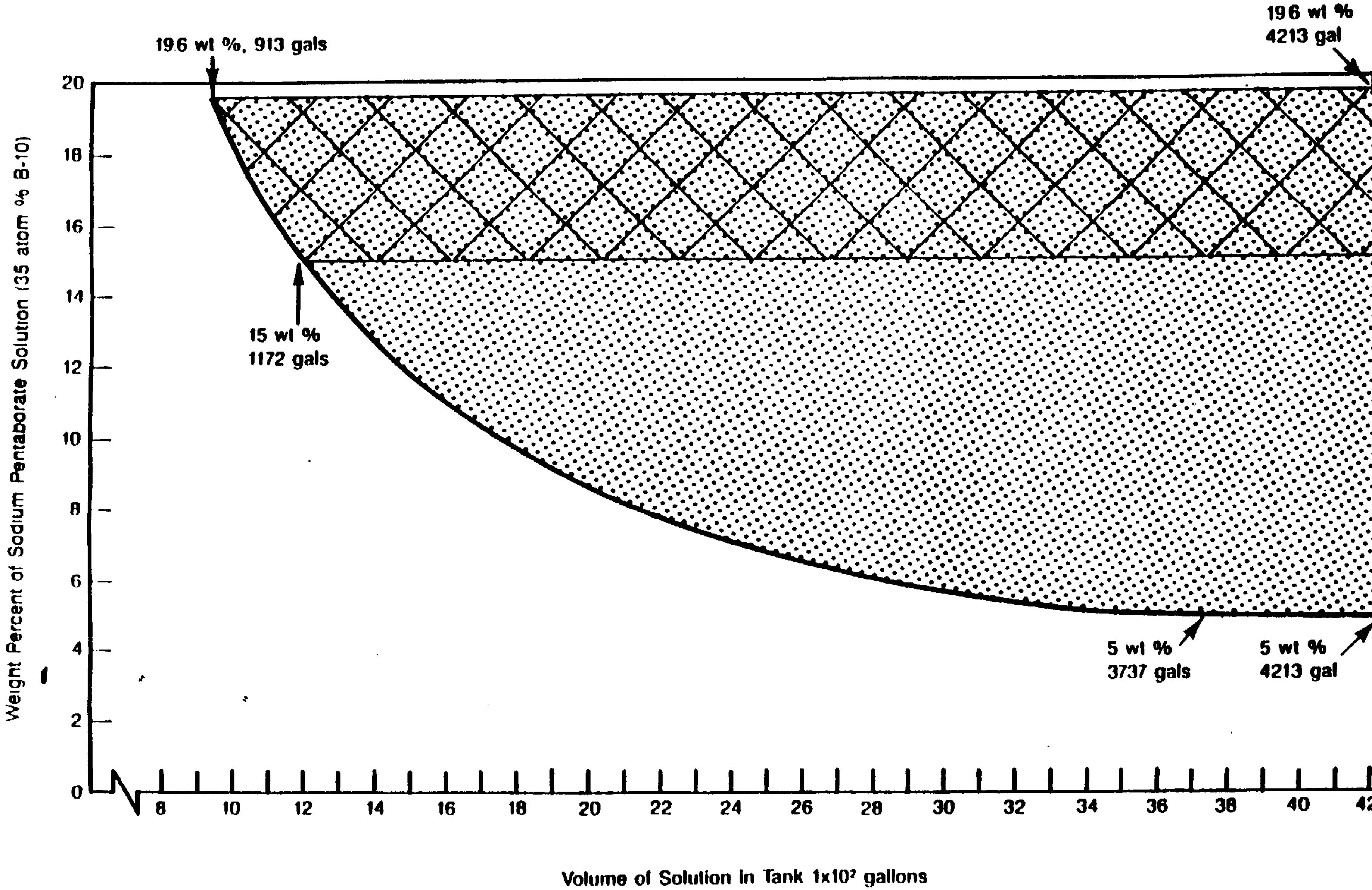
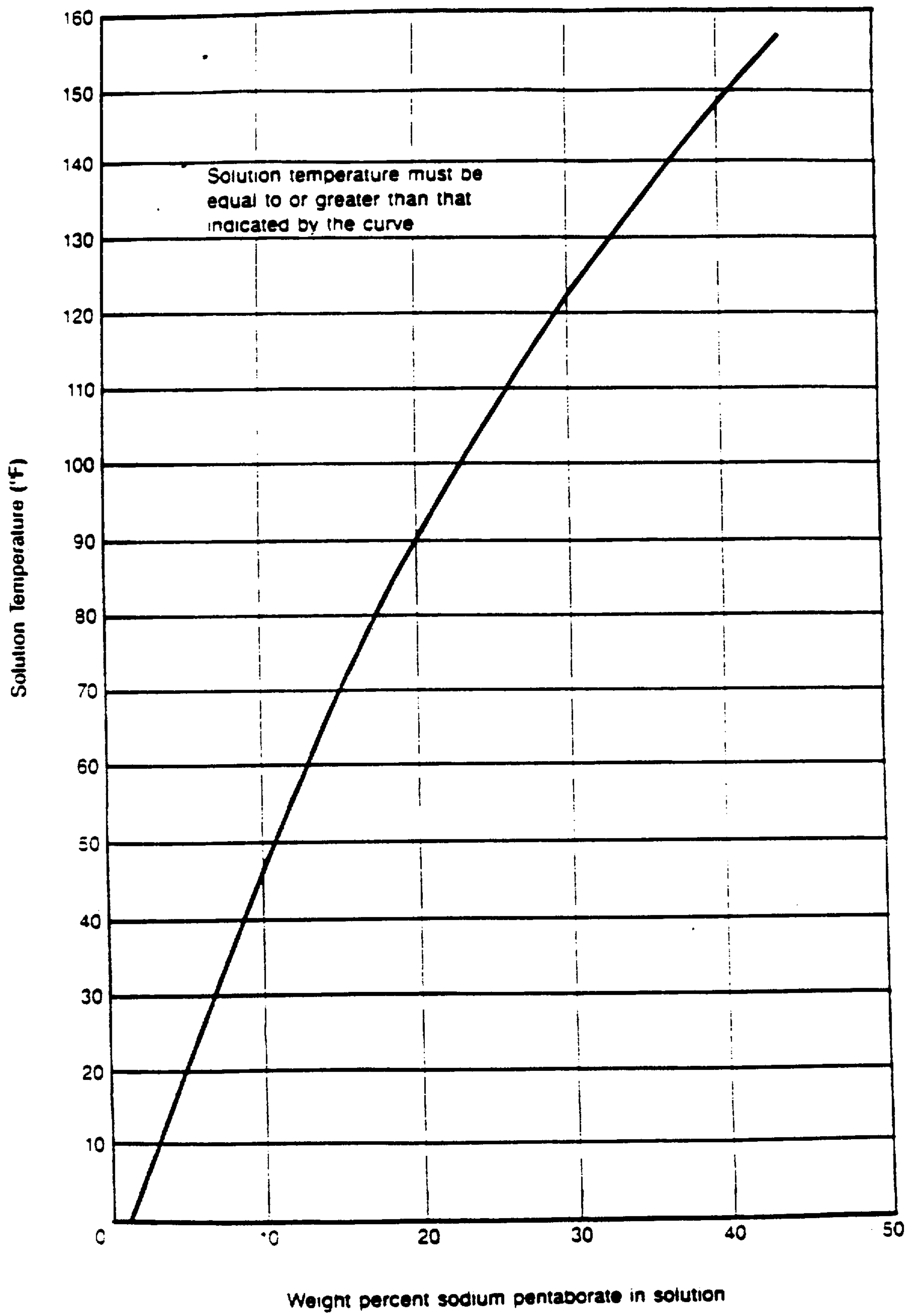


FIGURE 3.2.2 - Sodium Pentaborate Solution Temperature Requirements



being removed and one shall be located in an adjacent quadrant.

3. All other control rods are fully inserted with the exception of one rod which may be partially withdrawn not more than two notches to perform refueling interlock surveillance.

4. The four fuel assemblies are removed from the core cell surrounding each control rod or rod drive mechanism to be removed.

5. The SHUTDOWN MARGIN requirements of Specification 3.2.A are met.

6. An evaluation will be conducted for each refuel/reload to ensure that actual core criticality of the proposed order of defueling and refueling is bounded by previous analysis performed to support such defueling and refueling activities, otherwise a new analysis shall be performed.

   The new analysis must show that sufficient conservatism exists for the proposed order of defueling and refueling before such operation shall be allowed to proceed.

G. With any of the above requirements not met, cease core alterations or control rod removal as appropriate, and initiate action to satisfy the above requirements.

Basis: During refueling operations, the reactivity potential of the core is being altered. It is necessary to require certain interlocks and restrict certain refueling procedures such that there is assurance that inadvertent criticality does not occur.

Addition of large amounts of reactivity to the core is prevented by operating procedures, which are in turn backed up by refueling interlocks (1) on rod withdrawal and movement of the refueling platform. 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. With the mode switch in the refuel position only one control rod can be withdrawn (1,2).

The one rod withdrawal interlock may be bypassed in order to allow multiple control rod removal for repair, modifications, or core unloading. The requirements for simultaneous removal of more than one Control rod are more stringent than the requirements for removal of a single control rod, since in the latter

4.2 REACTIVITY CONTROL

Applicability: Applies to the surveillance requirements for reactivity control.

Objective: To verify the capability for controlling reactivity.

Specification:

A. SDM shall be verified:

1. Prior to each CORE ALTERATION and

2. Once within 4 hours following the first criticality following any CORE ALTERATION.

B. The control rod drive housing support system shall be inspected after reassembly.

C. 1. After each major refueling outage and prior to resuming power operation, all operable control rods shall be scram time tested from the fully withdrawn position with reactor pressure above 800 psig.

2. Following each reactor scram from rated pressure, the mean 90% insertion time shall be determined for eight selected rods. If the mean 90% insertion time of the selected control rod drives does not fall within the range of 2.4 to 3.1 seconds or the measured scram time of any one drive for 90% insertion does not fall within the range of 1.9 to 3.6 seconds, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is maintained.

3. Following any outage not initiated by a reactor scram, eight rods shall be scram tested with reactor pressure above 800 psig provided these have not been measured in six months. The same criteria of 4.2.C(2) shall apply.

D. Each partially or fully withdrawn control rod shall be exercised at least once each week. This test shall be performed within 24 hours in the event power operation is continuing with two or more inoperable control rods or in the event power operation is continuing with one fully or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance need not be completed within 24 hours if the number of inoperable rods has been reduced to less than two and if it has been demonstrated that control rod drive mechanism collet housing failure is not the cause of an immovable control rod.

E. Surveillance of the standby liquid control system shall be as follows:

1. Pump operability	Once/3 months
2. Boron concentration determination	Once/month
3. Functional test	Once every 24 months
4. Solution volume and temperature check	Once/day
5. Solution Boron-10 Enrichment	Once every 24 months. Enrichment analyses shall be received no later than 30 days after sampling. If not received within 30 days, notify NRC (within 7 days) of plans to obtain test results.

F. At specific power operation conditions, the actual control rod configuration will be compared with the expected configuration based upon appropriately corrected past data. This comparison shall be made every equivalent full power month. The initial rod inventory measurement performed with equilibrium conditions are established after a refueling or major core alteration will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle.

G. The scram discharge volume drain and vent valves shall be verified open at least once per 31 days, except in shutdown mode*, and shall be cycled at least one complete cycle of full travel at least quarterly.

H. All withdrawn control rods shall be determined OPERABLE by demonstrating the scram discharge volume drain and vent valves OPERABLE. This will be done at least once per refueling cycle by placing the mode switch in shutdown and by verifying that:

a. The drain and vent valves close within 30 seconds after receipt of a signal for control rods to scram, and

b. The scram signal can be reset and the drain and vent valves open when the scram discharge volume trip is bypassed.

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* These valves may be closed intermittently for testing under administrative control.

BASIS:

Adequate SDM must be demonstrated to ensure that the reactor can be made subcritical from any initial operating condition. Adequate SDM is demonstrated by testing before or during the first startup after fuel movement, control rod replacement, or shuffling within the reactor pressure vessel. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Since core reactivity will vary during the cycle as a function of fuel depletion and poison burnup, the beginning of cycle (BOC) test must also account for changes in core reactivity during the cycle. Therefore, to obtain the SDM, the initial measured value must be increased by an adder, "R", which is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated BOC core reactivity. If the value of R is negative (that is, BOC is the most reactive point in the cycle), no correction to the BOC measured value is required.

The SDM may be demonstrated during an in sequence control rod withdrawal, in which the highest worth control rod is analytically determined, or during local criticals. Local critical tests require the withdrawal of out of sequence control rods.

The frequency of 4 hours after reaching criticality is allowed to provide a reasonable amount of time to perform the required calculations and have appropriate verification.

During REFUEL MODE, adequate SDM is required to ensure that the reactor does not reach criticality during core alterations. An evaluation of each in vessel fuel movement during fuel loading (including shuffling fuel within the core) shall be performed to ensure adequate SDM is maintained during refueling. This evaluation can be a bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. For the SDM demonstrations that rely solely on calculation, additional margin must be added to the SDM limit of 0.38% delta k/k to account for uncertainties in the calculation.

The control rod drive housing support system[2] is not subject to deterioration during operation. However, reassembly must be assured following a partial or complete removal.

The scram insertion times for all control rods[3] will be determined at the time of each refueling outage. The scram times generated at each refueling outage when compared to scram times previously recorded gives a measurement of the functional effects of deterioration for each control rod drive. The more frequent scram insertion time measurements of eight selected rods are performed on a representative sample basis to monitor performance and give an early indication of possible deterioration and required maintenance. The times given for the eight-rod tests are based on the testing experience of control rod drives which were known to be in good condition.

The weekly control rod exercise test serves as a periodic check against deterioration of the control rod system. Experience with this control rod system has indicated that weekly tests are adequate, and that rods which move by drive pressure will scram when required as the pressure applied is much higher. The requirement to exercise the control rods within 24 hours of a condition with two or more control rods which are valved out of service or one fully or partially withdrawn control rod which can not be moved provides assurance of the reliability of the remaining control rods.

Pump operability, boron concentration, solution temperature and volume of standby liquid control system[4] are checked on a frequency consistent with instrumentation checks described in Specification 4.1. Experience with similar systems has indicated that the test frequencies are adequate. The only practical time to functionally test the liquid control system is during a refueling outage. The functional test includes the firing of explosive charges to open the shear plug valves and the pumping of demineralized water into the reactor to assure operability of the system downstream of the pumps. The test also includes recirculation of liquid control solution to and from the solution tanks.

Pump operability is demonstrated on a more frequent basis. This test consists of recirculation of demineralized water to a test tank. A continuity check of the firing circuit on the shear plug valves is provided by pilot lights in the control room. Tank level and temperature alarms are provided to alert the operator to off-normal conditions.

Figure 3.2.1 was revised to reflect the minimum and maximum weight percent of sodium pentaborate solution, and the minimum atom percent of B-10 to meet 10 CFR 50.62(c)(4). Since the weight percent of sodium pentaborate can change with water makeup or water evaporation, frequent surveillances are performed on the solution concentration, volume and temperature. The sodium pentaborate is enriched with B-10 at the chemical vendor's facility to meet the minimum atom percent. Preshipment samples of batches are analyzed for B-10 enrichment and verified by an independent laboratory prior to shipment to Oyster Creek. Since the B-10 enrichment will not change while in storage or in the SLCS tank, the surveillance for B-10 enrichment is performed on a 24 month interval. An additional requirement has been added to evaluate the solution's capability to meet the original design shutdown criteria whenever the Boron-10 enrichment requirement is not met.

The functional test and other surveillance on components, along with the monitoring instrumentation, gives a high reliability for standby liquid control system operability.

<u>References</u>

(1) FDSAR, Volume II, Figure III-5-11
(2) FDSAR, Volume I, Section VI-3
(3) FDSAR, Volume I, Section III-5 and Volume II, Appendix B
(4) FDSAR, Volume I, Section VI-4

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 178

TO FACILITY OPERATING LICENSE NO. DPR-16

GPU NUCLEAR CORPORATION AND

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated June 22, 1994, GPU Nuclear Corporation (GPUN), the licensee for Oyster Creek Nuclear Generating Station (OCNGS), requested a Technical Specification (TS) change. This request proposes changes to Section 1.6, 3.2.A, 3.9.f.5, and 4.2.A which specify the Shutdown Margin (SDM) requirements that ensure the reactor can be made subcritical to preclude inadvertent criticality in any core condition. This request also includes a definition of Shutdown Margin, Section 1.45. Administrative changes to TS Sections 1.7 and 3.2.b.2(b) are also included to simplify definitions and eliminate unnecessary notes and references.

2.0 EVALUATION

The requested changes clarify the requirements for demonstrating SDM, incorporate new, more restrictive SDM limits, and identify required Limiting Condition for Operation (LCO) actions if the SDM is not met in each operation mode. These actions were not identified in the current TSs. The surveillance requirements are also revised to identify conditions under which the SDM must be verified. The revised SDM limits account for the uncertainty in demonstration of adequate margin analytically (0.38% delta-k) or by measurement (0.28% delta k). Since the revised limits are also more restrictive than the current value of 0.25% delta-k, this is acceptable. The staff finds the above TS change request is acceptable with respect to SDM limits.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (59 FR 37072). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: E. D. Kendrick

Date: March 21, 1995