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

GPU NUCLEAR CORPORATION

AND

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 124
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 May 10, 1988 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 Provisional 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. 124, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, 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: July 14, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 124

PROVISIONAL 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. The corresponding overleaf pages are also provided to maintain document completeness.

Remove	Insert
Page 3.2.3	Page 3.2.3
Page 3.2.4	Page 3.2.4
Page 3.2.5	Page 3.2.5
Page 3.2.6	Page 3.2.6
Page 3.2.7	Page 3.2.7
Page 3.2.8	Page 3.2.8
Figure 3.2.1	Figure 3.2.1
Figure 3.2.2	Figure 3.2.2
Page 4.2-2	Page 4.2-2
Page 4.2-3	Page 4.2-3
Page 4.2-4	Page 4.2-4
Page 6-18	Page 6-18

2. The standby liquid control solution shall have a Boron-10 isotopic enrichment equal to or greater than 35 atom %, be maintained within the crosshatched 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 demonstrated daily to be operable, otherwise be in the Shutdown condition within 24 hours.

(b) If the solution is outside the crosshatched volume-concentration area but within the shaded volume-concentration area of Figure 3.2-1, return the solution to the crosshatched 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 crosshatched volume-concentration area.

(c) If the solution is outside the crosshatched 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-concentratioin 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. Reactivity Anomalies

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.

In order to assure that the basic criterion will be satisfied an additional design margin was adopted; that the k_{eff} be less than 0.99 in the cold xenon-free condition with the rod of highest worth fully withdrawn and all others fully inserted. Thus the design requirement is k_{eff} less than 0.99, whereas the minimum condition for operation is k_{eff} less than 1.0 with the operable rod of highest worth fully withdrawn (1). This limit allows control rod testing at any time in core life and assures that the plant can be shut down by control rods alone.

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 the 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 initial 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 (3). 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 10CFR50.62 (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 percent by wt. natural sodium pentaborate for a 251" diameter reactor pressure vessel in order to satisfy 10CFR50.62 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 (10CFR50.62) 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 crosshatched 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 10CFR50.62 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 allowed 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.

References:
(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)

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

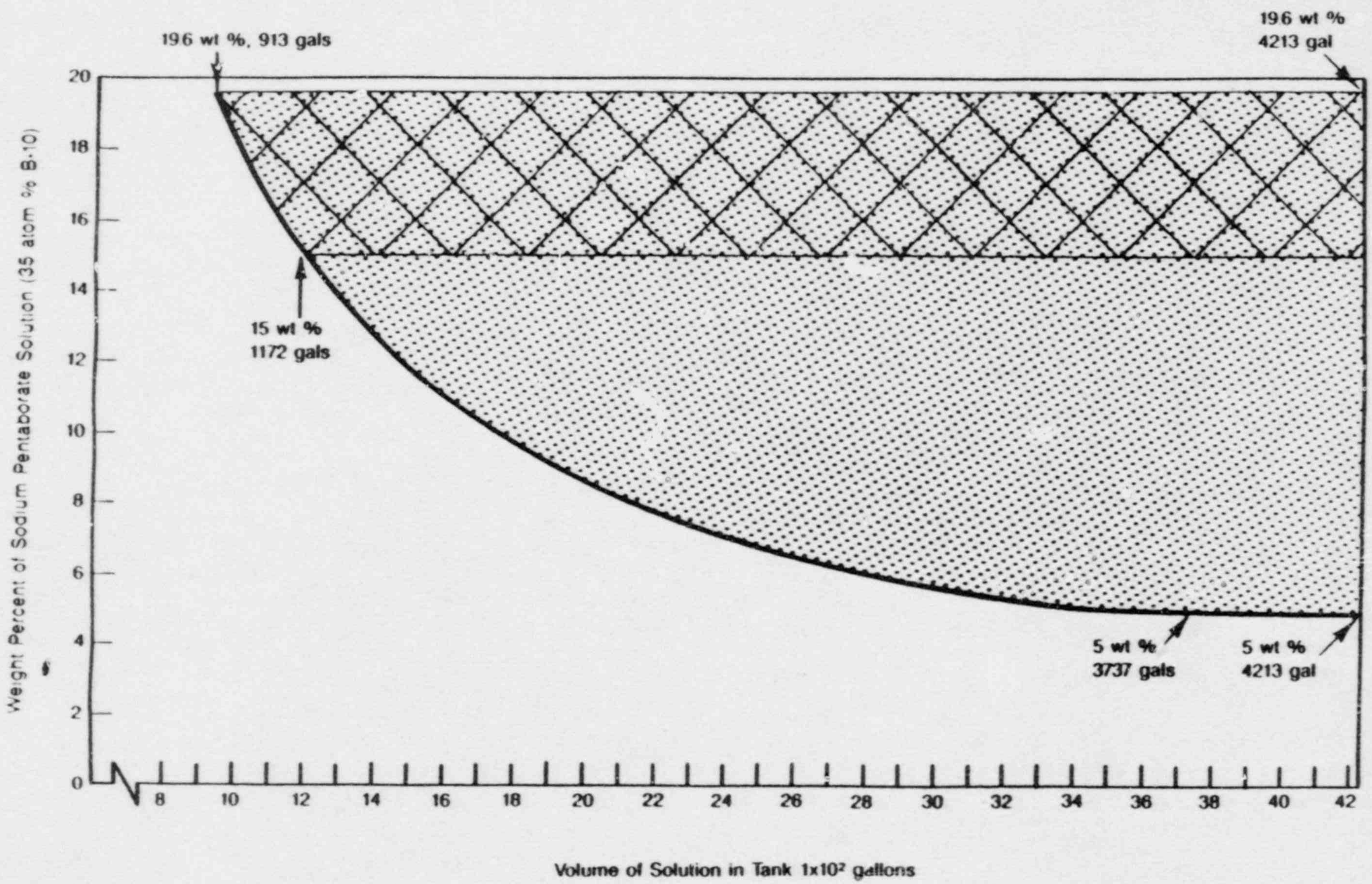
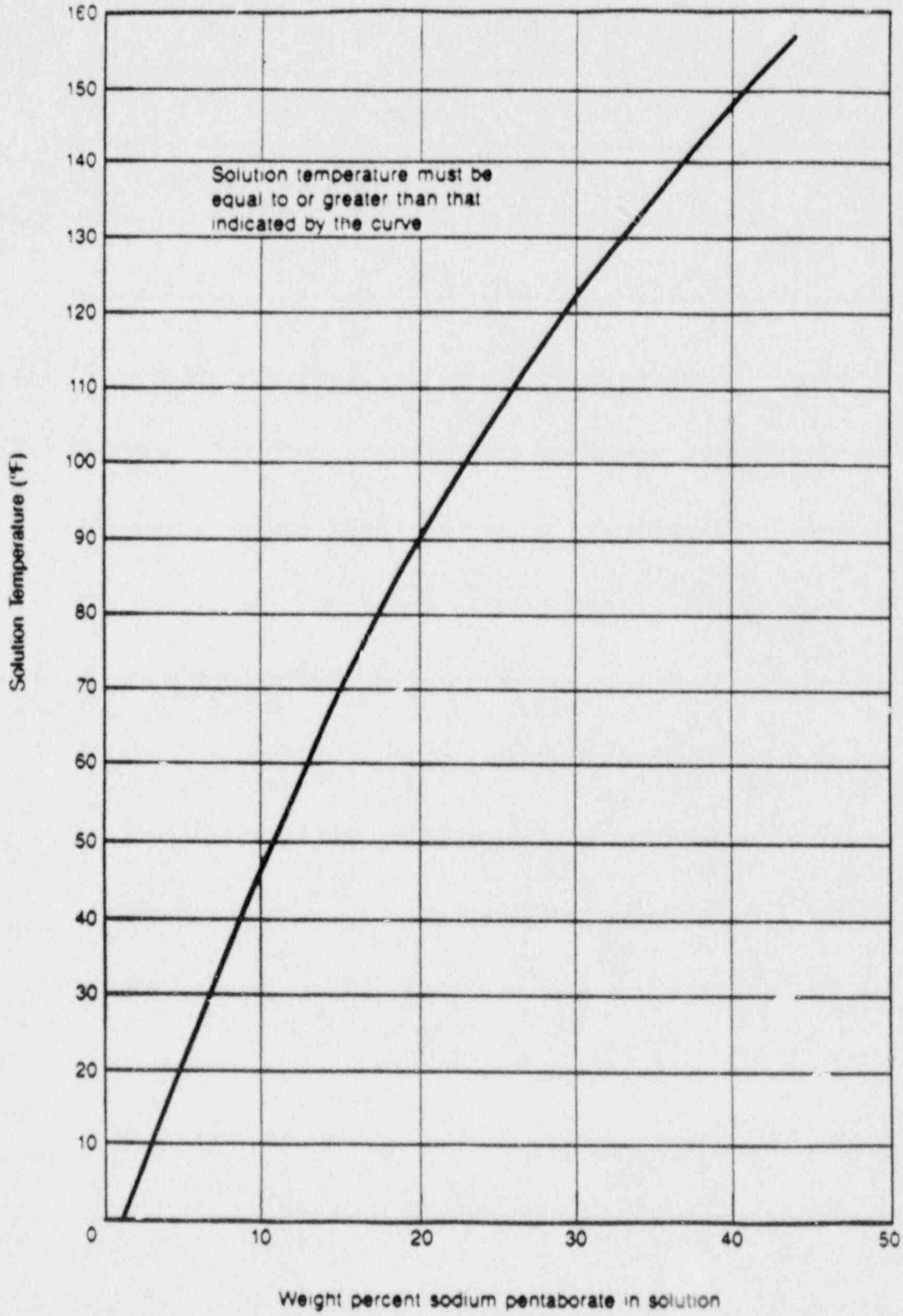


FIGURE 3.2.2 - Sodium Pentaborate Solution Temperature Requirements



3.	Functional test	Each refueling outage
4.	Solution volume and temperature check	Once/day
5.	Solution Boron-10 enrichment	Each refueling outage. Enrichment analyses shall be received no later than 30 days after startup from the refueling outage. 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.

Basis: The core reactivity limitation (Specification 3.2.A) requires that core reactivity be limited such that the core could be made subcritical at any time during the operating cycle, with the strongest operable control rod fully withdrawn and all other operable rods fully inserted. Compliance with this requirement can be demonstrated conveniently only at the time of refueling. Therefore, the demonstration must be such that it will apply to the entire subsequent fuel cycle. The demonstration is performed with the reactor core in the cold, xenon-free condition and will show

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

that the reactor is sub-critical at that time by at least R + 0.25% delta k with the highest worth operable control rod fully withdrawn.

The value of R is the difference between two calculated values of reactivity of the cold, xenon-free core with the strongest operable control rod fully withdrawn. The reactivity value at the beginning of life is subtracted from the maximum reactivity value anytime later in life to determine R, which must be a positive quantity or its value is conservatively taken as zero. The value of R shall include the potential shutdown margin loss assuming full B_4C settling in all possibly inverted tubes present in the core. The value 0.25% delta k in the expression R + 0.25% delta k serves at the beginning of life as a finite, demonstrable shutdown margin. This margin is demonstrated by full withdrawal of the strongest rod and partial withdrawal of a diagonally adjacent rod to a position calculated to insert an R + 0.25% delta k reactivity. Observation of subcriticality in this condition assures subcriticality with not only the strongest rod fully withdrawn but at least an R + 0.25% delta k margin beyond this.

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 [illegible]ck against deterioration of the control rod system. Experi[illegible] with this control rod system has indicated that weekly test[illegible] adequate, and that rods which move by drive pressure wi[illegible] scram when required as the pressure applied is much higher. The frequency of exercising the control rods has been increased under the conditions of two or more control rods which are valved out of service in order to provide even further 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.

Because Figure 3.2.1 has been revised to reflect the increased Boron-10 isotopic enrichment, 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.

References

(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

6.9.3 UNIQUE REPORTING REQUIREMENTS

Special reports shall be submitted to the Director of Regulatory Operations Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification.

a. Materials Radiation Surveillance Specimen Reports (4.3A)

b. Integrated Primary Containment Leakage Tests (4.5)

c. Results of required leak tests performed on sealed sources if the tests reveal the presence of 0.005 microcuries or more of removable contamination.

d. Inoperable Fire Protection Equipment (3.12)

e. Core Spray Sparger Inservice Inspection (Table 4.3.1-9)

Prior to startup of each cycle, a special report presenting the results of the inservice inspection of the Core Spray Spargers during each refueling outage shall be submitted to the Commission for review.

f. Liquid radwaste batch discharge exceeding Specification 3.6.B.1.

g. Main condenser offgas discharge without treatment per Specification 3.6.D.1.

h. Dose due to radioactive liquid effluent exceeding Specification 3.6.J.1.

i. Air dose due to radioactive noble gas in gaseous effluent exceeding Specification 3.6.L.1.

j. Air dose due to radiodine and particulates exceeding Specification 3.6.M.1.

k. Annual total dose due to radioactive effluents exceeding Specification 3.6.N.1.

l. Records of results of analyses required by the Radiological Environmental Monitoring Program.

m. Failures and challenges to Relief and Safety Valves

Failures and challenges to Relief and Safety Valves which do not constitute an LER will be the subject of a special report submitted to the Commission within 60 days of the occurrence. A challenge is defined as any automatic actuation (other than during surveillance or testing) of Safety or Relief Valves.

n. Plans for compliance with standby liquid control Specifications 3.2.C.3(b) and 3.2.C.3(e)(1) or plans to obtain enrichment test results per Specification 4.2.E.5.