BASES

APPLICABLE SAFETY ANALYSIS (continued)

Automatic enforcement of these reactor core SLs is provided by the following functions (Ref. 5):

a. High pressurizer pressure trip;

b. Low pressurizer pressure trip;

c. Overtemperature ΔT trip;

d. Overpower ΔT trip;

e. Power Range Neutron Flux trip; and

f. Steam generator safety valves.

Additional anticipatory trip functions are also provided for specific abnormal conditions.

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (Ref. 6) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

Figure B 2.1.1-1 shows an example of the reactor core safety limits of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is greater than or equal to the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the core exit quality is within the limits defined by the DNBR correlation. From this type of figure, the curves on Figure 2.1.1-1 of the accompanying specification can be generated. Each of the curves of Figure 2.1.1-1 has three distinct slopes. Working from left to right, the first slope ensures that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid such that overtemperature ΔT indication remains valid. The second slope ensures that the hot leg steam quality remains ≤ 15%. The final slope ensures that DNBR is always ≥ 1.4.

(continued)

BASES

BACKGROUND (continued)

The COLR provides peaking factor limits that ensure that the design basis value for departure from nucleate boiling ratio (DNBR) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements.

(continued)

BASES

BACKGROUND (continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;

b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F (Ref. 1);

c. During an ejected rod accident, the energy deposition to the fuel will be below 200 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

c. Steam Line Isolation - Containment Pressure - High High (continued)

Containment Pressure - High High must be OPERABLE in MODES 1, 2, and 3, because there is sufficient energy in the primary and secondary side to pressurize the containment following a pipe break. This would cause a significant increase in the containment pressure, thus allowing detection and closure of the MSIVs. The steam line isolation Function must be OPERABLE in MODES 2 and 3 unless both MSIVs are closed and de-activated. In MODES 4, 5, and 6 the steam line isolation Function is not required to be OPERABLE because there is not enough energy in the primary and secondary sides to pressurize the containment to the Containment Pressure - High High setpoint.

d. Steam Line Isolation - High Steam Flow Coincident With Safety Injection and Coincident With T_{avg} - Low

This Function provides closure of the MSIVs during an SLB or inadvertent opening of multiple SG atmospheric relief or safety valves to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

d. Steam Line Isolation - High Steam Flow Coincident With Safety Injection and Coincident With T_{avg} - Low

The specified Allowable Value is based on steam line breaks occurring from no-load conditions(1005 psig). Specifically, steam line breakswhich result in a > 10% RTP step change (0.66E6 lbm/hr) are considered. The steam flow signal to this function's bistables are not pressure compensated (i.e., only the main control board indicators are compensated). However, the high steam flow bistable setpoint is determined from the expected flow transmitter differential pressure under steam conditions of 0.66E6 lbm/hr at 1005 psig. Steam breaks which result in higher flowrates or lower pressure generate larger differential pressures such that the high steam flow bistables would be tripped. Steam line breaks which result in a < 10% RTP step change can be manually isolated by operators. The high steam flow bistables are OPERABLE if they are placed in the tripped condition since the specified Trip Setpoint and Allowable Value are met. However, all applicable surveillances related to the tripped channel must continue to be performed and met.

Two steam line flow channels per steam line are required to be OPERABLE for this Function. These are combined in a one-out-of-two logic to indicate high steam flow in one steam line. PT-464 and PT-465 are the two channels required for steam line A. FT-474 and FT-475 are the two channels required for steam line B. Each steam line is considered a separate function for the purpose of this LCO. The steam flow transmitters provide control inputs, but the control function cannot initiate events that the function acts to mitigate. Therefore, additional channels are not required to address control protection interaction issues. The one-out-of-two configuration allows online testing because trip of one high steam flow channel is not sufficient to cause initiation.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

d. Steam Line Isolation - High Steam Flow Coincident With Safety Injection and Coincident With T_{avg} - Low (continued)

This Function must be OPERABLE in MODES 1, 2, and 3 when a secondary side break or stuck open valve could result in rapid depressurization of the steam lines. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless both MSIVs are closed and de-activated. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the plant to have an accident.

e. Steam Line Isolation - High High Steam Flow Coincident With Safety Injection

This Function provides closure of the MSIVs during a large steam line break to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment.

The specified Allowable Value is based on steamline breaks occurring from full power steam conditions which result in ≥ 109% RTP steam flow. The steam flow signal to this function's bistables are not pressure compensated (i.e., only the main control board indicators are compensated). However, the high-high steam flow bistable setpoint is determined from the expected flow transmitter differential pressure under steam conditions of 3.7E6 lbm/hr at 755 psig. Steam breaks which result in higher flowrates or lower pressure generate larger differential pressures such that the high-high steam flow bistables would be tripped.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

e. Steam Line Isolation - High High Steam Flow Coincident With Safety Injection (continued)

Two steam line flow channels per steam line are required to be OPERABLE for this Function. These are combined in a one-out-of-two logic to indicate high-high steam flow in one steam line. FT-464 and FT-465 are the two channels required for steam line A. FT-474 and FT-475 are the two channels required for steam line B. Each steam line is considered a separate function for the purpose of this LCO. The steam flow transmitters provide control inputs, but the control function cannot initiate events that the Function acts to mitigate. Therefore, additional channels are not required to address control protection interaction issues.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the departure from nucleate boiling (DNB) design criterion will be met for each of the transients analyzed.

The design method employed to meet the DNB design criterion for fuel assemblies is the Revised Thermal Design Procedure (RTDP). With the RTDP methodology, uncertainties in plant operating parameters, computer codes and DNB correlation predictions are considered statistically to obtain DNB uncertainty factors. Based on the DNB uncertainty factors, RTDP design limit departure from nucleate boiling ratio (DNBR) values are determined in order to meet the DNB design criterion.

The RTDP design limit DNBR values are 1.24 and 1.23 for the typical and thimble cells, respectively, for fuel analyses with the WRB-1 correlation.

Additional DNBR margin is maintained by performing the safety analyses to DNBR limits higher than the design limit DNBR values. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility. The safety analysis DNBR value is 1.40 for the typical and thimble cells.

(continued)

BASES

BACKGROUND (continued)

For the WRB-1 correlation, the 95/95 DNBR correlation limit is 1.17. The W-3 DNB correlation is used where the primary DNBR correlation is not applicable. The WRB-1 correlation was developed based on mixing vane data and therefore is only applicable in the heated rod spans above the first mixing vane grid. The W-3 correlation, which does not take credit for mixing vane grids, is used to calculate DNBR values in the heated region below the first mixing vane grid. In addition, the W-3 correlation is applied in the analysis of accident conditions where the system pressure is below the range of the primary correlations. For system pressures in the range of 500 to 1000 psia, the W-3 correlation limit is 1.45. For system pressures greater than 1000 psia, the W-3 correlation limit is 1.30.

The RCS pressure limit as specified in the COLR, is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNB limits.

The RCS coolant average temperature limit as specified in the COLR, is consistent with full power operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average temperature will cause the core to approach DNB limits.

The RCS flow rate as specified in the COLR, normally remains constant during an operational fuel cycle with both pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. Flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SW system satisfies Criterion 3 of the NRC Policy Statement.

LCO

In the event of a DBA, one SW train and the loop header is required to be OPERABLE to provide the minimum heat removal capability to ensure that the system functions to remove post accident heat loads as assumed in the safety analyses. To ensure this requirement is met, two trains of SW and the loop header must be OPERABLE (see Figure B 3.7.8-1). At least one SW train will operate assuming that the worst case single active failure occurs coincident with the loss of offsite power.

A SW train is defined based on electrical power source such that SW Pumps A and C form one train and SW Pumps B and D form the second train. A SW train is considered OPERABLE when one pump in the train is OPERABLE and capable of taking suction from the screenhouse and providing cooling water to the loop header as assumed in the accident analyses. This includes consideration of available net positive suction head (NPSH) to the SW pumps and the temperature of the suction source. The following are the minimum requirements of the screenhouse bay with respect to OPERABILITY of the SW system:

a. Level ≥ 14 feet; and

b. Temperature ≥ 30°F and ≤ 80°F.

The screenhouse bay level verification should normally be performed using LI-3006. Monitoring screenhouse bay temperature (normally performed by using T3001) is an acceptable means of ensuring inlet temperature to safety related loads are within limits since significant portions of the service water piping runs underground. This tends to warm the water at the lower limit and cool the water at the upper limit. In addition, if a SW pump fails on Inservice Testing Program surveillance (e.g., pump developed head), the pump is only declared inoperable when the flowrate to required components is below that required to provide the heat removal capability assumed in the accident analyses (Ref. 1).

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the SW train cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

With both SW trains or the loop header inoperable, the plant is in a condition outside of the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

Required Action C.1 is modified by a Note requiring that the applicable Conditions and Required Actions of LCO 3.7.7, "CCW System," be entered for the component cooling water heat exchanger(s) made inoperable by SW. This note is provided since the inoperable SW system may prevent the plant from reaching MODE 5 as required by LCO 3.0.3 if both CCW heat exchangers are rendered inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR verifies that adequate NPSH is available to operate the SW pumps and that the SW suction source temperature is within the limits assumed by the accident analyses and the system design. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.12 Spent Fuel Pool (SFP) Boron Concentration

BASES

BACKGROUND

The water in the spent fuel pool (SFP) normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that a limiting k_{eff} of 0.95 be maintained in the absence of soluble boron. Hence, the design of both SFP regions is based on the use of unborated water such that the SFP design and configuration control (i.e., controlling the movement of the fuel assembly and checking the location of each assembly after movement) maintains each region in a subcritical condition during normal operation with the regions fully loaded. The Region 2 SFP design uses boraflex material that is secured to the rack structure. Testing has demonstrated that boraflex degradation is occurring such that boron must be credited to maintain $k_{eff} \leq 0.95$ at all times until a long-term solution is reached (Reference 5).

The double contingency principle discussed in ANSI N-16.1-1975 (Ref. 1) and Reference 2 allows credit for soluble boron under abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenarios are associated with the movement of fuel from Region 1 to Region 2, and accidental misloading of a fuel assembly in Region 2. Either scenario could potentially increase the reactivity of Region 2. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the storage racks with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.13, "Spent Fuel Pool (SFP) Storage" and by maintaining the minimum boron concentration required to address boraflex degradation. Within 7 days prior to movement of an assembly into a SFP region, it is necessary to perform SR 3.7.12.1. Prior to moving an assembly into a SFP region, it is also necessary to perform SR 3.7.13.1 or 3.7.13.2 as applicable.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The postulated accidents in the SFP can be divided into two basic categories (Ref. 3 and 4). The first category are events which cause a loss of cooling in the SFP. Changes in the SFP temperature could result in an increase in positive reactivity. However, the positive reactivity is ultimately limited by a combination of voiding (which would result in the addition of negative reactivity), the SFP geometry, and the high boron concentration in the SFP. The second category is related to the movement of fuel assemblies in the SFP (i.e., a fuel handling accident) and is the most limiting accident scenario with respect to reactivity. The types of accidents within this category include an incorrectly transferred fuel assembly (e.g., transfer from Region 1 to Region 2 of an unirradiated or an insufficiently depleted fuel assembly) and a dropped fuel assembly. However, for both of these accidents, the negative reactivity effect of the soluble boron compensates for the increased reactivity. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time.

The concentration of dissolved boron in the SFP satisfies Criterion 2 of the NRC Policy Statement.

LCO

The SFP boron concentration is required to be ≥ 2300 ppm. With the assumed boraflex available per the original design of the SFP storage racks, this value is 450 ppm. The specified concentration of dissolved boron in the SFP preserves the assumptions used in the analyses of the potential critical accident scenarios as described in References 3 and 4 (i.e., a fuel handling accident) and the boraflex degradation issue described in Reference 5. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage.

(continued)

BASES (continued)

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the SFP to compensate for boraflex degradation to ensure the SFP k_{eff} remains $\leq$ 0.95 at all times. This is expected to be corrected by December 31, 1999 per Specification 4.3.1.1.b.

ACTIONS

A.1 and A.2

When the concentration of boron in the SFP is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The initiation of actions to restore concentration of boron is simultaneous with suspending movement of fuel assemblies. This is necessary to compensate for any boraflex degradation within the SFP racks.

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply since if the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

This SR verifies that the concentration of boron in the SFP is within the limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate since the boron is credited with maintaining the SFP subcritical due to boraflex degradation. Also, the volume and boron concentration in the pool is normally stable and all water level changes and boron concentration changes are controlled by plant procedures.

This SR is required to be performed prior to fuel assembly movement into Region 1 or Region 2 and must continue to be performed until the necessary SFP verification is accomplished (i.e., SR 3.7.13.1 and 3.7.13.2).

REFERENCES

1. ANSI N16.1-1975, "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."

2. Letter from B.K. Grimes, NRC, to All Power Reactor Licensees, Subject: "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978.

3. Framatome Technologies, Inc., "R.E. Ginna Nuclear Power Plant, Spent Fuel Pool Re-racking Licensing Report," Section 4, February 1997.

4. UFSAR, Section 15.7.3.

5. Letter from R.C. Mecredy, RG&E, to G.S. Vissing, NRC, Subject: "Boraflex Degradation," dated March 30, 1998.

B 3.7 PLANT SYSTEMS

B 3.7.13 Spent Fuel Pool (SFP) Storage

BASES

BACKGROUND

The spent fuel pool (SFP) is divided into two separate and distinct regions (see Figure B 3.7.13-1) which, for the purpose of criticality considerations, are considered as separate pools (Ref. 1). Region 1, with 294 storage positions, is designed to accommodate new or spent fuel utilizing a checkerboard arrangement. All fuel assemblies stored in Region 1 must have a k-infinity that is ≤ 1.458. The existing design uses Integral Fuel Burnable Absorbers (IFBAs) as the poison for fuel assemblies with enrichments > 4.05 wt% to help achieve this k-infinity limit. IFBAs consist of neutron absorbing material which provides equivalencing reactivity holddown (i.e., neutron poison) that allows storage of higher enrichment fuel. The neutron absorbing material is a non-removable or integral part of the fuel assembly once it is applied. The infinite multiplication factor, K-infinity, is a reference criticality point of each fuel assembly that if maintained ≤ 1.458, will result in a k_{eff} ≤ 0.95 for Region 1. The K-infinity limit is derived for constant conditions of normal reactor core configuration (i.e., typical geometry of fuel assemblies in vertical position arranged in an infinite array) at cold conditions (i.e., 68°F and 14.7 psia). Fuel assemblies with minimum burnups above the curve in Figure 3.7.13-1 (area A) may be stored at any location within Region 1. Fuel assemblies with minimum burnups below the curve in Figure 3.7.13-1 (area B) may be stored in cells with lead-in funnels only.

(continued)

BASES (continued)

BACKGROUND (continued)

Region 2, with 1075 storage positions, is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domains according to Figure 3.7.13-2, in the accompanying LCO. Fuel assemblies with initial enrichments and burnups within domain A1 of Figure 3.7.13-2 may be stored in any location in Region 2. Fuel assemblies with initial enrichments and burnups within domain A2 of Figure 3.7.13-2 shall be stored face-adjacent to a Type A1 of A2 assembly, or a water cell (empty cell). Fuel assemblies with initial enrichments and burnups within domain B of Figure 3.7.13-2 shall be stored face-adjacent to a Type A1 assembly or a water cell (empty cell). Fuel assemblies with initial enrichments and burnups within domain C of Figure 3.7.13-2 shall be stored face-adjacent to a water cell (empty cell) only. The word "face-adjacent" on Figure 3.7.13-2 is defined to mean that the flat surface of a fuel assembly in one cell faces the flat surface of the fuel assembly in the next cell. The storage of fuel assemblies which are within the acceptable ranges of Figure 3.7.13-2 in Region 2 ensures a $K_{eff} \leq 0.95$ in this region.

Consolidated rod storage canisters can also be stored in either region in the SFP provided that the minimum burnup of Figure 3.7.13-1 and Figure 3.7.13-2 are met (Ref. 2). The canisters are stainless steel containers which contain the fuel rods of a maximum of two fuel assemblies (i.e., 358 rods). All bowed, broken, or otherwise failed fuel rods are first stored in a stainless steel tube of 0.75 inch outer diameter before being placed in a canister. Each canister will accommodate 110 failed fuel rod tubes.

(continued)

BASES (continued)

BACKGROUND (continued)

The water in the SFP normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that a limiting k_{eff} of 0.95 be maintained in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water such that the SFP design and configuration control (i.e., controlling the movement of the fuel assembly and checking the location of each assembly after movement) maintains each region in a subcritical condition during normal operation with the regions fully loaded. The SFP design uses boraflex material that is secured to the rack structure. Testing has demonstrated that boraflex degradation is ocurring such that boron must be temporarily credited to maintain $k_{eff} \leq 0.95$ at all times until a long-term solution is reached (Reference 6).

The double contingency principle discussed in ANSI N16.1-1975 (Ref. 3) and Reference 4 allows credit for soluble boron under abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenarios are associated with the movement of fuel from Region 1 to Region 2, and accidental misloading of a fuel assembly in Region 2. Either scenario could potentially increase the reactivity of Region 2. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the storage racks with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with this LCO and by maintaining the minimum boron concentration required to address boraflex degradation per LCO 3.7.12. Within 7 days prior to movement of an assembly into a SFP region, it is necessary to perform SR 3.7.12.1. Prior to moving an assembly into a SFP region, it is also necessary to perform SR 3.7.13.1 or 3.7.13.2 as applicable.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The postulated accidents in the SFP can be divided into two basic categories (Refs. 2 and 5). The first category are events which cause a loss of cooling in the SFP. Changes in the SFP temperature could result in an increase in positive reactivity. However, the positive reactivity is ultimately limited by a combination of voiding (which would result in the addition of negative reactivity), the SFP geometry, and the high boron concentration in the SFP. The second category is related to the movement of fuel assemblies in the SFP (i.e., a fuel handling accident) and is the most limiting accident scenario with respect to reactivity. The types of accidents within this category include an incorrectly transferred fuel assembly (e.g., transfer from Region 1 to Region 2 of an unirradiated or an insufficiently depleted fuel assembly) and a dropped fuel assembly. However, for both of these accidents, the negative reactivity effect of the soluble boron compensates for the increased reactivity. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time.

The configuration of fuel assemblies in the spent fuel pool satisfies Criterion 2 of the NRC Policy Statement.

LCO

The restrictions on the placement of fuel assemblies within the SFP ensure the k_{eff} of the SFP will always remain < 0.95. For fuel assemblies stored in Region 1, each assembly must have a K-infinity of $\leq$ 1.458 with initial enrichment and burnup within the acceptable area of Figure 3.7.13-1. For fuel assemblies stored in Region 2, initial enrichment and burnup shall be within the acceptable area of Figure 3.7.13-2. The word "face-adjacent" on Figure 3.7.13-2 is defined to mean that the flat surface of a fuel assembly in one cell faces the flat surface of the assembly in the next cell.

The x-axis of both figures is the nominal U-235 enrichment wt% which does not include the ± 0.05 wt% tolerance that is allowed for fuel manufacturing and listed in Specification 4.3.1.1.

(continued)

BASES (continued)

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the SFP.

ACTIONS

A.1

When the configuration of fuel assemblies stored in either Region 1 or Region 2 of the SFP is not within the LCO limits, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Specification 4.3.1.1. This compliance can be made by relocating the fuel assembly to a different region or to an acceptable new location within the same region.

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply since if the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR verifies by administrative means that the K-infinity of each fuel assembly is $\leq$ 1.458 prior to storage in Region 1 and that the initial enrichment and burnup is in accordance with Figure 3.7.13-1. If the initial enrichment of a fuel assembly is $\leq$ 4.05 wt%, a K-infinity of $\leq$ 1.458 is always maintained. For fuel assemblies with enrichment > 4.05 wt%, a minimum number of IFBAs must be present in each fuel assembly such that k-infinity $\leq$ 1.458 prior to storage in Region 1. This verification is only required once for each fuel assembly since the burnable poisons, if required, are an integral part of the fuel assembly and will not be removed. The initial enrichment of each assembly will also not change (i.e., increase) while partially burned assemblies are less reactive than when they were new (i.e., fresh). Performance of this SR ensures compliance with Specification 4.3.1.1.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1 (continued)

Though not required for this LCO, this SR must also be performed after completion of fuel movement within Region 1 to exit the Applicability of LCO 3.7.12, "SFP Boron Concentration."

SR 3.7.13.2

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.13-2 in the accompanying LCO prior to storage in Region 2. Once a fuel assembly has been verified to be within the acceptable range of Figure 3.7.13-1, further verifications are no longer required since the initial enrichment or burnup will not adversely change. For fuel assemblies in the unacceptable range of Figure 3.7.13-1, performance of this SR will ensure compliance with Specification 4.3.1.1.

Though not required for this LCO, this SR must also be performed after completion of fuel movement within Region 2 to exit the Applicability of LCO 3.7.12.

REFERENCES

1. UFSAR, Section 9.1.2.

2. Framatome Technologies, Inc., "R.E. Ginna Nuclear Power Plant, Spent Fuel Pool Re-racking Licensing Report," Section 4, February 1997.

3. ANSI N16.1-1975, "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."

(continued)

BASES (continued)

REFERENCES (continued)

4. Letter from B.K. Grimes, NRC, to All Power Reactor Licensees, Subject: "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978.

5. UFSAR, Section 15.7.3.

6. Letter from R.C. Mecredy, RG&E, to G.S. Vissing, NRC, Subject: "Boraflex Degradation," dated March 30, 1998.

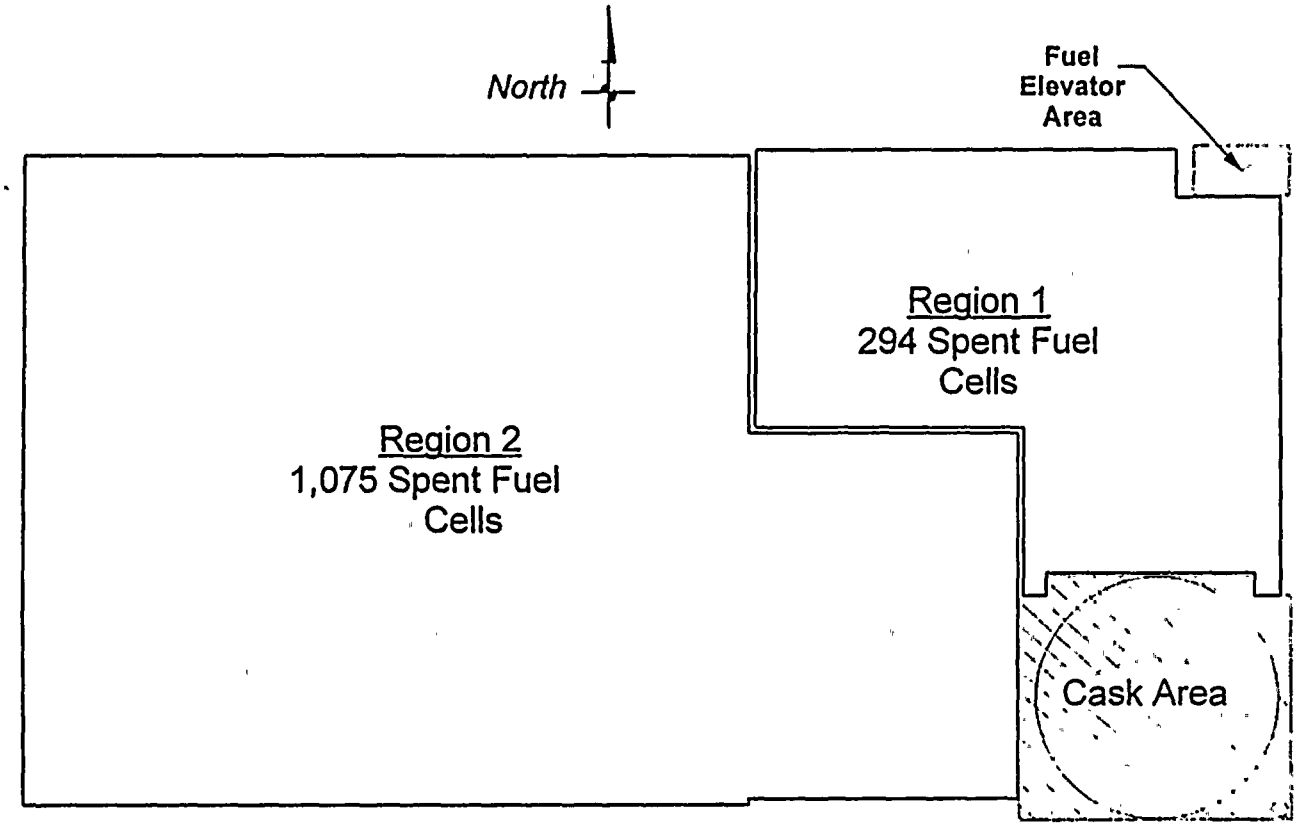

Figure B 3.7.13-1
Spent Fuel Pool

BASES

BACKGROUND (continued)

The onsite standby power sources consist of two 1950 kW continuous rating emergency diesel generators (DGs) connected to the safeguards buses to supply emergency power in the event of loss of all other AC power. The DGs are located in separate rooms in a Seismic Category I structure located adjacent to the northeast wall of the Turbine Building. Each DG room has its own ventilation system. The ventilation system is designed to maintain the DG room between 60°F and 104°F during normal operation and to remove any hydrocarbon gases in the room (Ref. 3). Each ventilation system consists of two fans and associated ductwork and dampers that fail open on loss of instrument air and control power. One fan is given a start permissive on DG actuation and will start when bus voltage is restored. The second fan designed to start when the room temperature reaches 90°F. The second fan's discharge air flow is directed to the DG instrument panel and starts after the room temperature reaches a preset temperature to prevent potentially freezing the cooling water jacket piping during cold weather conditions. During accident conditions the ventilation system has been analyzed to maintain the room ≤125°F on a maximum degree day and ≤140°F if only one fan is running (Ref. 9).

The DGs utilize an air motor for starting. The air motor is supplied by two receivers which provide sufficient air for five DG starts before requiring a recharge of the receivers. The DGs are supplied by separate fuel oil day tanks which can be cross-tied if required. Additional fuel oil can be transferred from redundant underground fuel oil storage tanks. A dedicated fuel oil transfer pump is used for this transfer. Redundancy of pumps and piping precludes the failure of one pump, or the rupture of any pipe, valve, or tank, to result in the loss of more than one DG.

DG A is dedicated to safeguards Buses 14 and 18 and DG B is dedicated to safeguards Buses 16 and 17. A DG starts automatically on a safety injection (SI) signal or on an undervoltage signal on its corresponding 480 V buses (refer to LCO 3.3.4, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation"). In the event of only an SI signal, the DGs automatically start and operate in the standby mode without tying to the safeguards buses.

(continued)

BASES

LCO (continued)

A DG is considered OPERABLE when:

a. The DG is capable of starting, accelerating to rated speed and voltage, and connecting to its respective 480 V safeguards buses on actuation of Loss of Power (LOP) DG Instrumentation within 10 seconds;

(continued)

BASES

LCO (continued)

b. All loads on each 480 V safeguards bus except for the safety related motor control centers, CCW pump, and CS pump are capable of being tripped on an undervoltage signal (CCW pump must be capable of being tripped on coincident SI and undervoltage signal);

c. The DG is capable of accepting required loads both manually and within the assumed loading sequence intervals following a coincident SI and undervoltage signal, and continue to operate until offsite power can be restored to the safeguards bus (i.e., 40 hours);

d. The DG day tank is available to provide fuel oil for ≥ 1 hour at 110% design loads;

e. The fuel oil transfer pump from the fuel oil storage tank to the associated day tank is OPERABLE including all required piping, valves, and instrumentation (long-term fuel oil supplies are addressed by LCO 3.8.3, "Diesel Fuel Oil");

f. A ventilation train consisting of at least one of two fans and the associated ductwork and dampers is OPERABLE; and

g. The service water (SW) Δp through the diesel generator heat exchangers is within the limits specified in plant operating procedures for SW system configuration.

Any 480 V bus fault which opens and/or prevents closure of the breakers from offsite power or the DGs requires declaring the offsite power source or DG inoperable, as applicable.

The AC sources in one train must be separate and independent of the AC sources in the other train. For the DGs, separation and independence must be complete assuming a single active failure. For the independent offsite power source, separation and independence are to the extent practical (i.e., operation is preferred in the 50/50 mode, but may also exist in the 100/0 or 0/100 mode).

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Chapter 8.

2. Atomic Industrial Forum (AIF) GDC 39, Issued for comment July 10, 1967.

3. UFSAR, Section 9.4.9.5.

4. UFSAR, Chapter 6.

5. UFSAR, Chapter 15.

6. 10 CFR 50, Appendix A, GDC 17.

7. "American National Standard, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.

8. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984.

9. UFSAR Section 3.11

BASES

LCO (continued)

A DG is considered OPERABLE when:

a. The DG is capable of starting, accelerating to rated speed and voltage, and connecting to its respective 480 V safeguards buses on actuation of Loss of Power (LOP) DG Instrumentation within 10 seconds;

b. All loads on each 480 V safeguards bus except for the safety related motor control centers, component cooling water (CCW) pump, and containment spray (CS) pump are capable of being tripped on an undervoltage signal (CCW pump must be capable of being tripped on coincident safety injection (SI) and undervoltage signal);

c. The DG is capable of accepting required loads manually. Since most equipment which receives a SI signal are isolated in these MODES due to maintenance or low temperature overpressure protection concerns, and the DBA of concern (i.e., a fuel handling accident) would not generate a SI signal, manual loading of the DGs will most likely be required. These loads must be capable of being added to the OPERABLE DG within 10 minutes;

d. The DG day tank is available to provide fuel oil for ≥ 1 hour at 110% design loads;

e. The fuel oil transfer pump from the fuel oil storage tank to the associated day tank is OPERABLE including all required piping, valves, and instrumentation (long-term fuel oil supplies are addressed by LCO 3.8.3, "Diesel Fuel Oil");

f. A ventilation train consisting of at least one of two fans and the associated ductwork and dampers is OPERABLE; and

g. The service water (SW) Δp through the diesel generator heat exchangers is within the limits specified in plant operating procedures for SW system configuration.

(continued)

BASES

BACKGROUND (continued)

Each battery provides a separate source of DC power independent of AC power. Each of the two batteries is capable of carrying its expected shutdown loads following a plant trip and a loss of all AC power for a period of 4 hours without battery terminal voltage falling below 108.6 V. Major loads and approximate operating times on each battery are discussed in the UFSAR (Ref. 2).

There are four battery chargers available to the batteries. Chargers 1A and 1B are rated at 150 amps and chargers 1A1 and 1B1 are rated at 200 amps. Battery chargers 1A and 1A1 are normally aligned to battery A, and battery chargers 1B and 1B1 are normally aligned to battery B. A charging capacity of at least 150 amps is normally required to supply the necessary DC loads on each train and to provide a full battery charge to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated Design Basis Accident (DBA). The DC power distribution system is described in more detail in Bases for LCO 3.8.9, "Distribution Systems—MODES 1, 2, 3, and 4," and LCO 3.8.10, "Distribution Systems—MODES 5 and 6."

The DC electrical power distribution subsystem also provide DC electrical power to the inverters, which in turn power the AC instrument buses. The inverters are described in more detail in Bases for LCO 3.8.7, "AC Instrument Bus Sources—MODES 1, 2, 3, and 4," and LCO 3.8.8, "AC Instrument Bus Sources—MODES 5 and 6."

Train A Engineered Safety Feature (ESF) equipment is supplied from battery A, while Train B ESF equipment is supplied from battery B. Additionally, the 480 V ESF switchgear and diesel generator (DG) control panels are supplied from either battery by means of an automatic transfer circuit in the switchgear and control panels. The normal supply from Train A (Buses 14 and 18 and DG A) is from DC distribution panels A. These panels also provide the emergency DC supply for Train B. Similarly, the normal supply from Train B (Buses 16 and 17 and DG B) is from DC distribution panels B. These panels also provide the emergency dc supply for Train A.

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Battery Cell Parameters

BASES

BACKGROUND

Each DC electrical power train contains a 125 VDC battery which is capable of carrying the expected shutdown loads following a plant trip and a loss of all AC power for a period of 4 hours without battery terminal voltage falling below 108.6 V. Major loads and approximate operating times on each battery are discussed in the UFSAR (Ref. 1). The batteries are normally in standby since the associated battery chargers provide for the required DC system loads.

The batteries for Train A and Train B DC electrical power are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and 100% design demand. Battery size is based on 125% of required capacity for aging considerations.

This LCO delineates the limits on electrolyte temperature, level, float voltage, and specific gravity for the DC power source batteries to ensure that the batteries are capable of performing their safety function as required by LCO 3.8.4, "DC Sources—MODES 1, 2, 3, and 4," and LCO 3.8.5, "DC Sources—MODES 5 and 6."

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

This SR verifies that the electrolyte level of each connected battery cell is above the top of the plates and not overflowing. This is consistent with IEEE-450 (Ref. 4) and ensures that the plates suffer no physical damage and maintain adequate electron transfer capability. The Frequency of 31 days is consistent with IEEE-450.

SR 3.8.6.2

This SR verifies that the float voltage of each connected battery cell is > 2.07 V. This limit is based on IEEE-450 (Ref. 4) which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement. The frequency of 31 days is also consistent with IEEE-450.

SR 3.8.6.3

This SR verifies the specific gravity of the designated pilot cell in each battery is ≥ 1.195. This value is based on manufacturer recommendations. According to IEEE-450 (Ref. 4), the specific gravity readings are based on a temperature of 77°F (25°C). The specific gravity readings are corrected for actual electrolyte temperature. For each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the reading; 1 point is subtracted for each 3°F below 77°F. The specific gravity of the electrolyte in a cell increases with a loss of water due to electrolysis or evaporation.

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is further discussed in IEEE-450. The Frequency of 31 days is consistent with IEEE-450.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.6.4

This SR verifies the average electrolyte temperature of the designated pilot cell in each battery is ≥ 55°F. This temperature limit is an initial assumption of the battery capacity calculations. The Frequency of 31 days is consistent with IEEE-450 (Ref. 4).

SR 3.8.6.5

This SR verifies that the average temperature of every fifth cell of each battery is ≥ 55°F. This is consistent with the recommendations of IEEE-450 (Ref. 4). Lower than normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. The Frequency of 92 days is consistent with IEEE-450.

SR 3.8.6.6

This SR verifies the specific gravity of each connected cell is not more than 0.020 below average of all connected cells and that the average of all connected cells is ≥ 1.195. This value is based on manufacturer recommendations and IEEE-450 (Ref. 4) which ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery. The temperature correction for specific gravity readings is the same as that discussed for SR 3.8.6.3. The Frequency of 92 days is consistent with IEEE-450.

REFERENCES

1. UFSAR, Section 3.8.2.
2. UFSAR, Chapter 6.
3. UFSAR, Chapter 15.
4. IEEE-450-1980.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.9.1

This SR verifies that the electrical power trains are functioning properly, with all required power source circuit breakers closed, tie-breakers open, and the buses energized from their allowable power sources. Required voltage for the AC electrical power distribution subsystem is ≥ 420 VAC; for the DC electrical power distribution subsystem, ≥ 108.6 VDC; and for AC instrument bus electrical power distribution subsystem, between 113 VAC and 123 VAC. Required voltage for the twinco panels (except MQ-400E) supplied by the 120 VAC instrument buses is between 115.6 VAC and 120.4 VAC. Required voltage for twinco panel MQ-400E is between 115.6 VAC and 124.4 VAC. Required voltage for inverter MQ-483 is between 107 volts and 129.8 volts. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The Frequency of 7 days takes into account the redundant capability of the AC, DC, and AC instrument bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 15.
3. UFSAR, Section 8.3.1.
4. 10 CFR 50, Appendix A, GDC 17.
5. UFSAR, Figure 8.3-1.
6. UFSAR, Figure 8.3-6.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1

This Surveillance verifies that the electrical power distribution trains are functioning properly, with all the required power source circuit breakers closed, required tie-breakers open, and the required buses energized from their allowable power sources. Required voltage for the AC power distribution electrical subsystem is ≥ 420 VAC, for the DC power distribution electrical subsystem ≥ 108.6 VDC, and for AC instrument bus power distribution electrical subsystem is between 113 VAC and 123 VAC. Required voltage for the twinco panels (except MQ-400E) supplied by the 120 VAC instrument buses is between 115.6 VAC and 120.4 VAC. Required voltage for twinco panel MQ-400E is between 115.6 VAC and 124.4 VAC. Required voltage for inverter MQ-483 is between 107 volts and 129.8 volts. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The Frequency of 7 days takes into account the capability of the AC, DC, and AC instrument bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES None.

50-244 Distribution Sheet 3/8/00

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ROBERT C. MECREDY
Vice President
Nuclear Operations

March 8, 2000

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Guy S. Vissing
Project Directorate I-1
Washington, D.C. 20555

Subject: Application for Amendment to Facility Operating License
Credit for Soluble Boron in Spent Fuel Pool
Rochester Gas and Electric Corporation
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Reference: (a) Letter from Robert C. Mecredy (RG&E) to Guy S. Vissing (NRC), "Boraflex Degradation", dated March 30, 1998.

(b) Letter from Guy S. Vissing (NRC) to Robert C. Mecredy (RG&E), "Issuance of Amendment No. 72 to Facility Operating License No. DPR-18, R. E. Ginna Nuclear Power Plant", dated July 30, 1998.

(c) Letter from Robert C. Mecredy (RG&E) to Guy S. Vissing (NRC), "Application for Amendment to Facility Operating License Date Change for Boraflex Degradation Temporary Measures", dated October 20, 1999.

(d) Letter from Guy S. Vissing (NRC) to Robert C. Mecredy (RG&E), "Issuance of Amendment Regarding a Change from December 31, 1999, to June 30, 2001, Specified in the Technical Specifications (TS) 4.3.1.1.b Note Associated with Maintaining Spent Fuel Pool Boron Concentration >2300 Parts per Million (PPM) at all Times Until a Permanent Resolution to Current Criticality Concerns are Implemented", dated December 21, 1999.

Dear Mr. Vissing:

The enclosed License Amendment Request (LAR) proposes to revise the Ginna Station Improved Technical Specifications (ITS) associated with the Spent Fuel Pool (SFP) Storage (LCO 3.7.13) and Design Features Fuel Storage (4.3).

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In Reference (a), RG&E notified the NRC that testing of boraflex panels contained within Region 2 of the spent fuel pool (SFP) indicated degradation such that certain portions of the criticality analysis provided to the NRC may no longer be conservative. This included the ability of Region 2 to maintain a $k_{eff} \leq 0.95$ if flooded with unborated water. The letter described interim compensatory actions taken by RG&E until a permanent solution with respect to boraflex degradation could be engineered. By Reference (b) the NRC provided approval of the interim measures and the addition of a footnote to the ITS associated with the Design Features Fuel Storage Specification 4.3.1.1.b which required that 2300 ppm boron be maintained in the SFP until December 31, 1999. By Reference (c), RG&E requested a revision to the footnote date for the interim measures, and by Reference (d) the NRC provided approval until June 30, 2001.

This request provides a permanent solution with respect to the boraflex degradation concern through a revision to the storage configuration requirements within the existing storage racks and taking credit for a limited amount of soluble boron, based on a revision of the criticality safety analyses. This would allow resolution of this issue without requiring a physical modification to the storage racks.

RG&E requests that this amendment be approved by May 31, 2001. The requested approval date is based on the date when the authorization for interim measures expires. RG&E requests that upon NRC approval, this LAR should be effective immediately and implemented within 30 days, to allow time for required documentation changes.

Very truly yours,

Robert C. Mecredy
Vice President
Nuclear Operations Group

Attachments:

I. License Amendment Request
II. No Significant Hazards Consideration Determination
III. Environmental Impact Consideration Determination
IV. Marked up Copy of R.E. Ginna Nuclear Power Plant Improved Technical Specifications
V. Proposed Revised R.E. Ginna Nuclear Power Plant Improved Technical Specifications

Enclosures:

1. R.E. Ginna Spent Fuel Pool Boron Dilution Analysis, January 2000
2. R.E. Ginna Nuclear Power Plant Criticality Safety Analysis for the Spent Fuel Storage Rack Using Soluble Boron Credit, February 2000

xc: Mr. Guy S. Vissing (Mail Stop 8C2)
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
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U.S. NRC Ginna Senior Resident Inspector

Mr. F. William Valentino, President
New York State Energy, Research, and Development Authority
Corporate Plaza West
286 Washington Avenue Extension
Albany, NY 12203-6399

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Rochester Gas and Electric Corporation	)	Docket No. 50-244
(R.E. Ginna Nuclear Power Plant)	)	

APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Pursuant to Section 50.90 of the regulations of the U.S. Nuclear Regulatory Commission (the "Commission"), Rochester Gas and Electric Corporation ("RG&E"), holder of Facility Operating License No. DPR-18, hereby requests that the Improved Technical Specifications set forth in Appendix A to that license be amended. This request for change in Improved Technical Specifications is to revise the storage requirements contained within the Spent Fuel Pool (SFP) Storage (LCO 3.7.13) and Design Features Fuel Storage (4.3) specifications.

A description of the amendment request, necessary background information, and justification of the requested change are provided in Attachment I. The no significant hazards consideration determination is provided as Attachment II. The environmental impact consideration determination is provided as Attachment III. A marked up copy of the current Ginna Station Improved Technical Specifications which shows the requested change is set forth in Attachment IV. The proposed revised Improved Technical Specifications are provided in Attachment V.

The evaluation set forth in Attachment I and III demonstrates that the proposed change does not involve a significant change in the types or a significant increase in the amounts of effluents or any change in the authorized power level of the facility. The proposed change also does not involve a significant hazards consideration, as documented in Attachment II.

WHEREFORE, Applicant respectfully requests that Appendix A to Facility Operating License No. DPR-18 be amended in the form attached hereto as Attachment V.

Rochester Gas and Electric Corporation

By _____
Robert C. Mecredy
Vice President
Nuclear Operations Group

Subscribed and sworn to before me
on this 8th day of March, 2000.

_____
Notary Public

SHARON P. SORTINO
Notary Public, State of New York
Registration No. 01S06017755
Monroe County
Commission Expires December 21, 20__

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ROBERT C. MECREDY
Vice President
Nuclear Operations

November 30, 1999

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Guy S. Vissing
Project Directorate I-1
Washington, D.C. 20555

Subject: Application for Amendment to Facility Operating License
Ventilation Filter Testing Program Change (5.5.10)
Rochester Gas and Electric Corporation
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Reference: NRC Generic Letter 99-02: Laboratory Testing of Nuclear-Grade Activated Charcoal, dated June 3, 1999.

Dear Mr. Vissing:

The enclosed License Amendment Request (LAR) proposes to revise the Ginna Station Improved Technical Specifications associated with the Ventilation Filter Testing Program (VFTP) (5.5.10).

This LAR is being proposed as the result of NRC Generic Letter 99-02, which requested that all addressees determine whether their technical specifications (TS) reference American Society for Testing and Materials (ASTM) D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," for charcoal adsorber laboratory testing. Addressees whose TS do not reference ASTM D3803-1989 were requested to either amend their TS to reference ASTM D3803-1989 or propose an alternative test protocol. RG&E is proposing to revise the Ginna Station Improved Technical Specifications to reference ASTM D3803-1989 for charcoal adsorber laboratory testing. Also, as the result of the more conservative nature of the proposed testing, RG&E is also proposing to revise the acceptance criteria for the charcoal adsorber.

RG&E requests that upon NRC approval, this LAR should be effective immediately and implemented within 30 days.

Very truly yours,

Robert C. Mecredy

Attachments

xc: Mr. Guy S. Vissing (Mail Stop 8C2)
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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New York State Energy, Research, and Development Authority
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286 Washington Avenue Extension
Albany, NY 12203-6399

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Rochester Gas and Electric Corporation	)	Docket No. 50-244
(R.E. Ginna Nuclear Power Plant)	)	

APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Pursuant to Section 50.90 of the regulations of the U.S. Nuclear Regulatory Commission (the "Commission"), Rochester Gas and Electric Corporation ("RG&E"), holder of Facility Operating License No. DPR-18, hereby requests that the Improved Technical Specifications set forth in Appendix A to that license be amended. This request for change in Improved Technical Specifications is to revise the Ventilation Filter Testing Program (5.5.10) requirements to reference ASTM D3803-1989 for charcoal adsorber laboratory testing and to revise the acceptance criteria.

A description of the amendment request, necessary background information, justification of the requested changes, and environmental impact considerations determination are provided in Attachment I. The no significant hazards consideration evaluation is provided as Attachment II. A marked up copy of the current Ginna Station Improved Technical Specifications which shows the requested changes is set forth in Attachment III. The proposed revised Improved Technical Specifications are provided in Attachment IV.

The evaluation set forth in Attachment I demonstrates that the proposed changes do not involve a significant change in the types or a significant increase in the amounts of effluent or any change in the authorized power level of the facility. The proposed changes also do not involve a significant hazards consideration, as documented in Attachment II.

WHEREFORE, Applicant respectfully requests that Appendix A to Facility Operating License No. DPR-18 be amended in the form attached hereto as Attachment IV.

Rochester Gas and Electric Corporation

By Robert C. Mecredy

Robert C. Mecredy
Vice President
Nuclear Operations Group

Subscribed and sworn to before me
on this 30th day of November, 1999.

Sharon P. Sortino
Notary Public

SHARON P. SORTINO
Notary Public, State of New York
Registration No. 01S06017755
Monroe County
Commission Expires December 21, 2000

Attachment I
R.E. Ginna Nuclear Power Plant

LICENSE AMENDMENT REQUEST
VENTILATION FILTER TESTING PROGRAM CHANGE

This attachment provides a description of the amendment request and necessary justification for the proposed changes. The attachment is divided into five sections as follows. Section A identifies all changes to the current Ginna Station Improved Technical Specifications (ITS) while Section B provides the background and history associated with the changes being requested. Section C provides detailed justification for the proposed changes. An environmental impact consideration of the requested changes is provided in Section D. Section E lists all references used in Attachments I and II.

A. DESCRIPTION OF AMENDMENT REQUEST

This License Amendment Request (LAR) proposes to revise Ginna Station ITS to reflect the change in testing standards for the charcoal adsorber laboratory testing. The change is summarized below and shown in Attachments III and IV.

1. ADMINISTRATIVE CONTROLS 5.5

   a. Specification 5.5.10 is changed to provide a reference to the test methodology modifications within the subsequent specifications.

   b. Specification 5.5.10 (a.3) is changed to provide a reference to ASTM D3803-1989, and to provide a specific test temperature and relative humidity. The allowable limit for methyl iodide penetration is also changed.

   c. Specification 5.5.10 (c.5) is changed to provide a reference to ASTM D3803-1989, and to provide a specific test temperature and relative humidity. The allowable limit for methyl iodide penetration is also changed.

   d. Specification 5.5.10 (d.3) is changed to provide a reference to ASTM D3803-1989, and to provide a specific test temperature and relative humidity. The allowable limit for methyl iodide penetration is also changed.

B. BACKGROUND

The installation of charcoal filters in an effluent stream provides an effective means of removing iodide from the stream and thereby reduces doses resulting from an effluent release. The installed charcoal filter systems and their effectiveness has been assumed to reduce the consequences of design basis accidents in three areas of Ginna Station. The Containment Post-Accident Charcoal System is installed in containment to mitigate the consequences of a loss of coolant accident and other less severe accidents which may occur inside the containment. Charcoal filtration is provided in the Control Room Emergency Air Treatment System to maintain doses to control room operators at acceptable levels following a design basis accident. The Spent Fuel Pool Charcoal Adsorber System is installed in the immediate vicinity of the spent fuel pool in the auxiliary building to assist in the mitigation of the consequences of a fuel handling accident in the auxiliary building. Testing the capacity of the charcoal to adsorb iodide assures that an acceptable removal efficiency under accident conditions would be obtained. The difference between the test requirement for the removal efficiency for methyl iodide and the percentage assumed in the associated accident analysis provides adequate safety margin for degradation of the filter after the tests.

In July 1980 the NRC issued an SER (Reference 1) approving the incorporation of changes to the Technical Specification test requirements for the installed charcoal filter systems. The changes specified the applicable test conditions for temperature and relative humidity, and the associated methyl iodide removal efficiency acceptance criteria. The test conditions were specific to the individual filters and were established to be representative of the values of temperature and relative humidity which enveloped the conditions that may be present at the systems following a design basis accident.

In February 1996 the NRC issued an SER (Reference 2) approving the full conversion of the Ginna Technical Specifications (TS) to a version based on NUREG-1431, "Standard Technical Specifications, Westinghouse Plants" (Reference 3). As part of the conversion the details (temperature and relative humidity) associated with the methyl iodide testing of charcoal adsorbers in the plant ventilation systems were allowed to be relocated to plant procedures. The current requirements of the Ginna Ventilation Filter Testing Program (VFTP), as stated in the Improved Technical Specifications (ITS) are to test the charcoal adsorbers in accordance with Regulatory Guide 1.52, Revision 2. The plant procedures which implement the VFTP provide reference to standards ANSI/ASME N-509-1976 (Nuclear Power Plant Air Cleaning Unit and Components) and ANSI/ASME N-510-1975 (Testing of Nuclear Air Cleaning Systems).

In June 1999 the NRC issued Generic Letter 99-02 (Reference 4), which was issued to alert addressees that the NRC had determined that testing nuclear-grade activated charcoal to standards other than American Society for Testing and Materials (ASTM) D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," does not provide assurance for complying with the current licensing basis as it relates to the dose limits of General Design Criterion (GDC) 19 of

Appendix A to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR) and Subpart A of 10 CFR Part 100. Utilities were requested to evaluate their charcoal adsorber test requirements and if it is determined to be necessary to adopt the ASTM D3803-1989 protocol, then to submit a TS amendment request to require testing to this protocol within 180 days of the date of the generic letter. The NRC also requested that the amendment request contain the test temperature, relative humidity, and penetration at which the proposed TS will require the test to be performed and the basis for these values. Because ASTM D3803-1989 is a more accurate and demanding test than older tests, the NRC is allowing utilities that upgrade their TS to this new protocol to be able to use a safety factor as low as 2 for determining the acceptance criteria for charcoal adsorber efficiency.

C. JUSTIFICATION OF CHANGES

This section provides the justification for all changes described in Section A above and shown on Attachment IV. The justifications are organized based on whether the change is: more restrictive (M), less restrictive (L), administrative (A), or the requirement is relocated (R). The justifications listed below are also referenced in the technical specification(s) which are affected (see Attachment III).

C.1 More Restrictive

M.1 The requirements for testing the Containment Post-Accident Charcoal System (5.5.10.a.3) will be revised to specifically provide reference to standard ASTM D3803-1989 as the test methodology and to specify a test temperature of 30°C (86°F) and a relative humidity of 95%. The 30°C (86°F) test temperature is considered to be a more conservative value, based on industry testing as described in Generic Letter 99-02 (Reference 4), than the current value of 141°C (286°F) which is specified in the RG&E filter inspection and testing program procedure. The 95% relative humidity is due to there being no humidity control. ASTM D3803-1989 is considered to be the most accurate and most realistic protocol for testing charcoal in ventilation systems because it offers the greatest assurance of accurately and consistently determining the capability of the charcoal. It requires the test to be performed at a constant low temperature of 30°C (86°F) and it provides for smaller tolerances in temperature, humidity, and air flow; and it has a humidity pre-equilibration. The proposed new test protocol and conditions will provide a higher assurance of the ability of the charcoal adsorbers to perform as assumed in the accident analysis and follows the guidance of NUREG-1431.

M.2 The requirements for testing the Control Room Emergency Air Treatment System (5.5.10.c.5) will be revised to specifically provide reference to standard ASTM D3803-1989 as the test methodology and to specify a test temperature of 30°C (86°F) and a relative humidity of 95%. The 30°C (86°F) test temperature is considered to be a more conservative value, based on industry testing as described in Generic Letter 99-02 (Reference 4), than the current value of 52°C (125°F) which is specified in the RG&E filter inspection and testing program procedure. The 95% relative humidity is due to there being no humidity control. ASTM D3803-1989 is considered to be the most accurate and most realistic protocol for testing charcoal in ventilation systems because it offers the greatest assurance of accurately and consistently determining the capability of the charcoal. It requires the test to be performed at a constant low temperature of 30°C (86°F) and it provides for smaller tolerances in temperature, humidity, and air flow; and it has a humidity pre-equilibration. The proposed new test protocol and conditions will provide a higher assurance of the ability of the charcoal adsorbers to perform as assumed in the accident analysis and follows the guidance of NUREG-1431.

M.3 The requirements for testing the Spent Fuel Pool Charcoal Adsorber System (5.5.10.d.3) will be revised to specifically provide reference to standard ASTM D3803-1989 as the test methodology and to specify a test temperature of 30°C (86°F) and a relative humidity of 95%. The 30°C (86°F) test temperature is considered to be a more conservative value, based on industry testing as described in Generic Letter 99-02 (Reference 4), than the current value of 66°C (150°F) which is specified in the RG&E filter inspection and testing program procedure. The 95% relative humidity is due to there being no humidity control. ASTM D3803-1989 is considered to be the most accurate and most realistic protocol for testing charcoal in ventilation systems because it offers the greatest assurance of accurately and consistently determining the capability of the charcoal. It requires the test to be performed at a constant low temperature of 30°C (86°F) and it provides for smaller tolerances in temperature, humidity, and air flow; and it has a humidity pre-equilibration. The proposed new test protocol and conditions will provide a higher assurance of the ability of the charcoal adsorbers to perform as assumed in the fuel handling accident analysis and follows the guidance of NUREG-1431.

C.2 <u>Less Restrictive</u>

L.1 The methyl iodide penetration requirements of the Containment Post-Accident Charcoal System (5.5.10.a.3) will be revised to increase the allowed penetration from 10% to 14.5%. This change is the result of reducing the safety factor used in calculating the allowed penetration to a value of 2. The laboratory test acceptance criteria contain a safety factor to ensure that the efficiency assumed in the accident analysis is still valid at the end of the operating cycle. Because ASTM D3803-1989 is a more accurate and

demanding test than older tests, Generic Letter 99-02 allows utilities that upgrade their TS to this new protocol to be able to use a safety factor as low as 2 for determining the acceptance criteria for charcoal filter efficiency.

$$\text{Allowable Penetration} = \frac{100\% - \text{Accident Analysis Methyl Iodide Assumption}}{\text{Safety Factor}}$$

$$\text{Allowable Penetration} = \frac{100\% - 71\%}{2}$$

$$\text{Allowable Penetration} = 14.5\%$$

L.2 The methyl iodide penetration requirements of the Control Room Emergency Air Treatment System (5.5.10.c.5) will be revised to increase the allowed penetration from 10% to 14.5%. This change is the result of reducing the safety factor used in calculating the allowed penetration to a value of 2. The laboratory test acceptance criteria contain a safety factor to ensure that the efficiency assumed in the accident analysis is still valid at the end of the operating cycle. Because ASTM D3803-1989 is a more accurate and demanding test than older tests, Generic Letter 99-02 allows utilities that upgrade their TS to this new protocol to be able to use a safety factor as low as 2 for determining the acceptance criteria for charcoal filter efficiency.

$$\text{Allowable Penetration} = \frac{100\% - \text{Accident Analysis Methyl Iodide Assumption}}{\text{Safety Factor}}$$

$$\text{Allowable Penetration} = \frac{100\% - 71\%}{2}$$

$$\text{Allowable Penetration} = 14.5\%$$

L.3 The methyl iodide penetration requirements of the Spent Fuel Pool Charcoal Adsorber System (5.5.10.d.3) will be revised to increase the allowed penetration from 10% to 14.5%. This change is the result of reducing the safety factor used in calculating the allowed penetration to a value of 2. The laboratory test acceptance criteria contain a safety factor to ensure that the efficiency assumed in the accident analysis is still valid at the end of the operating cycle. Because ASTM D3803-1989 is a more accurate and demanding test than older tests, Generic Letter 99-02 allows utilities that upgrade their TS to this new protocol to be able to use a safety factor as low as 2 for determining the acceptance criteria for charcoal filter efficiency.

$$\text{Allowable Penetration} = \frac{100\% - \text{Accident Analysis Methyl Iodide Assumption}}{\text{Safety Factor}}$$

$$\text{Allowable Penetration} = \frac{100\% - 71\%}{2}$$

$$\text{Allowable Penetration} = 14.5\%$$

There are no administrative (A) or relocated (R) changes associated with this LAR.

D. ENVIRONMENTAL IMPACT CONSIDERATION

RG&E has evaluated the proposed changes and determined that:

1. The changes do not involve a significant hazards consideration as documented in Attachment II; and

2. The changes do not involve a significant change in the types or significant increase in the amounts of any effluent that may be released offsite since more stringent test protocol are being imposed and there is no change in accident assumptions; and

3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure since no new or different type of equipment are required to be installed as a result of this LAR.

Accordingly, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed changes is not required.

E. REFERENCES

a. Letter from Dennis M. Crutchfield (NRC) to Leon D.White (RG&E), "Issuance of Amendment No. 34 to Provisional Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant", dated July 14, 1980.

2. Letter from Allen R. Johnson (NRC) to Robert C. Mecredy (RG&E), "Issuance of Amendment No. 61 to Facility Operating License No. DPR-18, R. E. Ginna Nuclear Power Plant", dated February 13, 1996.

3. NUREG-1431 Rev.1, "Standard Technical Specifications, Westinghouse Plants".

4. NRC Generic Letter 99-02, "Laboratory testing of Nuclear-Grade Activated Charcoal", dated June 3, 1999.

Attachment II
R.E. Ginna Nuclear Power Plant

SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

The proposed changes to the Ginna Station Improved Technical Specifications as identified in Attachment I Section A and justified by Section C have been evaluated with respect to 10 CFR 50.92(c) and shown not to involve a significant hazards consideration as described below. This attachment is organized based on Attachment I Section C.

Evaluation of More Restrictive Changes

The more restrictive changes associated with the providing a reference to American Society for Testing and Materials (ASTM) D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon,", and to provide a specific test temperature and relative humidity for testing the charcoal adsorbers do not involve a significant hazards consideration as discussed below:

1) Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The changes add a reference to the latest approved test protocol and provide for specific test conditions. This does not increase the probability of an accident previously evaluated since the tests are of themselves not an accident initiator. The proposed changes are in accordance with NUREG-1431 guidance and provide a higher assurance of the ability of the charcoal adsorbers to perform as assumed in the accident analysis. Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

2) Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes add specific details of charcoal adsorber testing and do not of themselves involve a physical alteration of the plant (ie. no new or different type of equipment will be added to perform the required testing) or changes in the methods governing normal plant operation. The changes only involve implementing currently approved test methodology. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

3) Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes only add conservatism in the test requirements for the charcoal adsorbers credited in the accident analysis. ASTM D3803-1989 is considered to be the most accurate and most realistic protocol for testing charcoal in ventilation systems because it offers the greatest assurance of accurately and consistently determining the capability of the charcoal. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the preceding information, it has been determined that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

Evaluation of Less Restrictive Changes

The less restrictive changes associated with revising the allowable test limit for methyl iodide penetration of charcoal adsorbers do not involve a significant hazards consideration as discussed below:

1) Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The changes revise the acceptance criteria for the allowed penetration of methyl iodide during the testing of charcoal adsorbers in the plant ventilation systems. This does not increase the probability of an accident previously evaluated since the tests are of themselves not an accident initiator. Because ASTM D3803-1989 is a more accurate and demanding test than older tests this new protocol will allow the use a safety factor of 2 for determining the acceptance criteria for charcoal filter efficiency. The new acceptance criteria continue to ensure that the efficiency assumed in the accident analysis is still valid. Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

2) Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes of revising charcoal adsorber testing acceptance criteria do not of themselves involve a physical alteration of the plant (ie. no new or different type of equipment will be added to perform the required testing) or changes in the methods governing normal plant operation. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

3) Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes only revise the test acceptance criteria of charcoal adsorbers as the result of implementing testing in accordance with ASTM D3803-1989. ASTM D3803-1989 is considered to be the most accurate and most realistic protocol for testing charcoal in ventilation systems because it offers the greatest assurance of accurately and consistently determining the capability of the charcoal. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the preceding information, it has been determined that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

Attachment III
R.E. Ginna Nuclear Power Plant

Mark-up of Existing Ginna Station Technical Specifications

Included pages:

5.0-12
5.0-13

5.5 Programs and Manuals (continued)

5.5.10 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature filter ventilation systems and the Spent Fuel Pool (SFP) Charcoal Adsorber System. The test frequencies and methods will be in accordance with Regulatory Guide 1.52, Revision 2, except that in lieu of 18 month test intervals, a 24 month interval will be implemented.

Insert 1

a. Containment Post-Accident Charcoal System

1. Demonstrate the pressure drop across the charcoal adsorber bank is < 3 inches of water at a design flow rate (± 10%).

2. Demonstrate that an in-place Freon test of the charcoal adsorber bank shows a penetration and system bypass < 1.0%, when tested under ambient conditions.

3. Demonstrate for a carbon sample that a laboratory analysis shows the iodine removal efficiency of ≥ 90% of radioactive methyl iodide.

Insert 2

b. Containment Recirculation Fan Cooler System

1. Demonstrate the pressure drop across the high efficiency particulate air (HEPA) filter bank is < 3 inches of water at a design flow rate (± 10%).

2. Demonstrate that an in-place dioctylphthalate (DOP) test of the HEPA filter bank shows a penetration and system bypass < 1.0%.

c. Control Room Emergency Air Treatment System (CREATS)

1. Demonstrate the pressure drop across the HEPA filter bank is < 3 inches of water at a design flow rate (± 10%).

2. Demonstrate that an in-place DOP test of the HEPA filter bank shows a penetration and system bypass < 1.0%.

(continued)

5.5 Programs and Manuals (continued)

5.5.10 VFTP (continued)

3. Demonstrate the pressure drop across the charcoal adsorber bank is < 3 inches of water at a design flow rate (± 10%).

4. Demonstrate that an in-place Freon test of the charcoal adsorber bank shows a penetration and system bypass < 1.0%, when tested under ambient conditions.

5. Demonstrate for a carbon sample that a laboratory analysis shows the iodine removal efficiency of ≥ 90% of radioactive methyl iodide.

Insert 3

d. SFP Charcoal Adsorber System

1. Demonstrate that the total air flow rate from the charcoal adsorbers shows at least 75% of that measured with a complete set of new adsorbers.

2. Demonstrate that an in-place Freon test of the charcoal adsorbers bank shows a penetration and system bypass < 1.0%, when tested under ambient conditions.

3. Demonstrate for a carbon sample that a laboratory analysis shows the iodine removal efficiency of ≥ 90% of radioactive methyl iodide.

Insert 4

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP frequencies.

(continued)

Insert 1

The test methods will be in accordance with Regulatory Guide 1.52, Revision 2, except as modified below.

Insert 2

Demonstrate that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows a methyl iodide penetration of less than 14.5% when tested in accordance with ASTM D3803-1989 at a test temperature of 30°C (86°F) and a relative humidity of 95%. L.1 M.1

Insert 3

Demonstrate that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows a methyl iodide penetration of less than 14.5% when tested in accordance with ASTM D3803-1989 at a test temperature of 30°C (86°F) and a relative humidity of 95%. L.2 M.2

Insert 4

Demonstrate that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows a methyl iodide penetration of less than 14.5% when tested in accordance with ASTM D3803-1989 at a test temperature of 30°C (86°F) and a relative humidity of 95%. L.3 M.3

Attachment IV
R.E. Ginna Nuclear Power Plant

Proposed Ginna Station Technical Specifications

Included pages:

5.0-12
5.0-13

5.5 Programs and Manuals (continued)

5.5.10 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature filter ventilation systems and the Spent Fuel Pool (SFP) Charcoal Adsorber System. The test frequencies will be in accordance with Regulatory Guide 1.52, Revision 2, except that in lieu of 18 month test intervals, a 24 month interval will be implemented. The test methods will be in accordance with Regulatory Guide 1.52, Revision 2, except as modified below.

a. Containment Post-Accident Charcoal System

1. Demonstrate the pressure drop across the charcoal adsorber bank is < 3 inches of water at a design flow rate (± 10%).

2. Demonstrate that an in-place Freon test of the charcoal adsorber bank shows a penetration and system bypass < 1.0%, when tested under ambient conditions.

3. Demonstrate that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows a methyl iodide penetration of less than 14.5% when tested in accordance with ASTM D3803-1989 at a test temperature of 30°C (86°F) and a relative humidity of 95%.

b. Containment Recirculation Fan Cooler System

1. Demonstrate the pressure drop across the high efficiency particulate air (HEPA) filter bank is < 3 inches of water at a design flow rate (± 10%).

2. Demonstrate that an in-place dioctylphthalate (DOP) test of the HEPA filter bank shows a penetration and system bypass < 1.0%.

c. Control Room Emergency Air Treatment System (CREATS)

1. Demonstrate the pressure drop across the HEPA filter bank is < 3 inches of water at a design flow rate (± 10%).

(continued)

5.5 Programs and Manuals (continued)

5.5.10 VFTP (continued)

2. Demonstrate that an in-place DOP test of the HEPA filter bank shows a penetration and system bypass < 1.0%.

3. Demonstrate the pressure drop across the charcoal adsorber bank is < 3 inches of water at a design flow rate (± 10%).

4. Demonstrate that an in-place Freon test of the charcoal adsorber bank shows a penetration and system bypass < 1.0%, when tested under ambient conditions.

5. Demonstrate that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows a methyl iodide penetration of less than 14.5% when tested in accordance with ASTM D3803-1989 at a test temperature of 30°C (86°F) and a relative humidity of 95%.

d. SFP Charcoal Adsorber System

1. Demonstrate that the total air flow rate from the charcoal adsorbers shows at least 75% of that measured with a complete set of new adsorbers.

2. Demonstrate that an in-place Freon test of the charcoal adsorbers bank shows a penetration and system bypass < 1.0%, when tested under ambient conditions.

3. Demonstrate that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows a methyl iodide penetration of less than 14.5% when tested in accordance with ASTM D3803-1989 at a test temperature of 30°C (86°F) and a relative humidity of 95%.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP frequencies.

(continued)

Unit Serial No: 616277 Version: 02.31

Network Address: 00:40:af:4b:3a:a8

Network Topology: Ethernet Connector: RJ45

Network Speed: 10 Megabits

Novell Network Information enabled
    Print Server Name: RDP_616277
    Password Defined: No
    Preferred Server Name not defined
    Directory Services Context not defined
    Frame Type: 802.2 On 802.3

Peer-to-Peer Information enabled
    Frame Type: 802.2 On 802.3
    Network ID: 0

TCP/IP Network Information enabled
    Frame Type: Ethernet II Protocol Address: Not Configured
    Subnet Mask: 255.0.0.0 Default Gateway: 0.0.0.0

AppleTalk Network Information enabled
    Frame Type: 802.2 SNAP On 802.3
    Protocol Address: Net Number 65384 Node Number 224 Socket Number 129
    Preferred AppleTalk Zone: Default Zone

Novell inactive

Peer-to-Peer Connection Information
    Printer Name: RDP_616277

AppleTalk Connection Information
    AppleTalk Printer Name: RDP_616277

TCP/IP Connection Information
    Port Number : 10001

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Subject:
Letter forwarding application for amend to license DPR-18, proposing r ev to ventilation filter testing program (5.5.10) requirements to refe rence ASTM D3803-1989 for charcoal adsorber lab testing & to revise th e acceptance criteria. Tech specs, enclosed.

Body:
PDR ADOCK 05000244 P

Docket: 05000244, Notes: License Exp date in accordance with 10CFR2,2. 109(9/19/72).

A Subsidiary of RGS Energy Group, Inc.

ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE, ROCHESTER, N.Y. 14649-0001 • 716 546-2700 www.rge.com

ROBERT C. MECREDY
Vice President
Nuclear Operations

October 20, 1999

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Guy S. Vissing
Project Directorate I-1
Washington, D.C. 20555

Subject: Application for Amendment to Facility Operating License
Date Change for Boraflex Degradation Temporary Measures
Rochester Gas and Electric Corporation
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Reference: (a) Letter from Robert C. Mecredy (RG&E) to Guy S. Vissing (NRC), "Boraflex Degradation", dated March 30, 1998.

(b) Letter from Guy S. Vissing (NRC) to Robert C. Mecredy (RG&E), "Issuance of Amendment No. 72 to Facility Operating License No. DPR-18, R. E. Ginna Nuclear Power Plant", dated July 30, 1998.

Dear Mr. Vissing:

The enclosed License Amendment Request (LAR) proposes to revise the Ginna Station Improved Technical Specifications (ITS) associated with the Design Features Fuel Storage (4.3).

In Reference (a), RG&E notified the NRC that testing of boraflex panels contained within Region 2 of the spent fuel pool (SFP) indicated degradation such that certain portions of the criticality analysis provided to the NRC may no longer be conservative. This included the ability of Region 2 to maintain a $k_{eff} \leq 0.95$ if flooded with unborated water. The letter described interim compensatory actions taken by RG&E until a permanent solution with respect to boraflex degradation could be engineered.

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By Reference (b) the NRC provided approval of the interim measures and the addition of a footnote to the ITS associated with the Design Features Fuel Storage Specification 4.3.1.1.b which required that 2300 ppm boron be maintained in the SFP until December 31, 1999. RG&E has been evaluating an analytical approach to the boraflex degradation which would allow resolution of the issue without requiring physical modifications to the SFP. RG&E currently plans to have a permanent solution to the boraflex degradation concern approved by the NRC and implemented by June 30, 2001. This reflects the time needed to finalize the criticality analysis, perform required fuel assembly movements, and obtain NRC review and approval of the required ITS changes. RG&E expects to submit a LAR to remove the conservative boron concentration requirement no later than March 10, 2000. Therefore, the purpose of this letter is to revise the date by which Specification 4.3.1.1.b will be met or amended.

RG&E requests that upon NRC approval, this LAR should be effective immediately and implemented within 30 days.

Very truly yours,

Robert C. Mecredy

Attachments

xc: Mr. Guy S. Vissing (Mail Stop 8C2)
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

U.S. NRC Ginna Senior Resident Inspector

Mr. F. William Valentino, President
New York State Energy, Research, and Development Authority
Corporate Plaza West
286 Washington Avenue Extension
Albany, NY 12203-6399

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Rochester Gas and Electric Corporation	)	Docket No. 50-244
(R.E. Ginna Nuclear Power Plant)	)	

APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Pursuant to Section 50.90 of the regulations of the U.S. Nuclear Regulatory Commission (the "Commission"), Rochester Gas and Electric Corporation ("RG&E"), holder of Facility Operating License No. DPR-18, hereby requests that the Improved Technical Specifications set forth in Appendix A to that license be amended. This request for change in Improved Technical Specifications is to revise the date within the footnote to Design Features Fuel Storage Specification 4.3.1.1.b which applies to the temporary measures associated with boraflex degradation.

A description of the amendment request, necessary background information, justification of the requested change, and environmental impact considerations determination are provided in Attachment I. The no significant hazards consideration evaluation is provided as Attachment II. A marked up copy of the current Ginna Station Improved Technical Specification which shows the requested change is set forth in Attachment III. The proposed revised Improved Technical Specification is provided in Attachment IV.

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The evaluation set forth in Attachment I demonstrates that the proposed change does not involve a significant change in the types or a significant increase in the amounts of effluent or any change in the authorized power level of the facility. The proposed change also does not involve a significant hazards consideration, as documented in Attachment II.

WHEREFORE, Applicant respectfully requests that Appendix A to Facility Operating License No. DPR-18 be amended in the form attached hereto as Attachment IV.

Rochester Gas and Electric Corporation

By Robert C. Mecredy

Robert C. Mecredy
Vice President
Nuclear Operations Group

Subscribed and sworn to before me
on this 20th day of October, 1999.

Sharon P. Sortino
Notary Public

SHARON P. SORTINO
Notary Public, State of New York
Registration No. 01S06017755
Monroe County
Commission Expires December 21, 2000

Attachment I
R.E. Ginna Nuclear Power Plant

LICENSE AMENDMENT REQUEST
DATE CHANGE FOR BORAFLEX DEGRADATION TEMPORARY MEASURES

This attachment provides a description of the amendment request and necessary justification for the proposed changes. The attachment is divided into five sections as follows. Section A identifies all changes to the current Ginna Station Improved Technical Specifications (ITS) while Section B provides the background and history associated with the changes being requested. Section C provides detailed justification for the proposed changes. An environmental impact consideration of the requested changes is provided in Section D. Section E lists all references used in Attachments I and II.

A. DESCRIPTION OF AMENDMENT REQUEST

This License Amendment Request (LAR) proposes to revise Ginna Station ITS to reflect the new date by which the Spent Fuel Pool (SFP) boraflex degradation issue will be resolved. The change is summarized below and shown in Attachments III and IV.

1. DESIGN FEATURES 4.3

   a. The note associated with Specification 4.3.1.1.b is revised to add a new date of June 30, 2001 as the date until which the SFP shall be maintained with a boron concentration ≥ 2300 ppm.

B. BACKGROUND

In 1998, RG&E identified boraflex degradation in Region 2 as a result of testing (Reference 1). To compensate for this degradation, RG&E proposed that the spent fuel pool boron concentration be maintained ≥ 2300 ppm at all times until a permanent resolution could be implemented. The NRC provided approval of this temporary measure within an SER (Reference 2) associated with changes to the SFP storage requirements. At that time RG&E had planned to have a permanent solution to the boraflex degradation concern implemented by December 31, 1999. The proposed date reflected the time needed to both evaluate, design, and implement necessary modifications to the SFP.

During the summer and fall of 1998, RG&E met with a number of vendors to evaluate the available options for modifying the spent fuel pool storage configuration. These options included SFP storage rack inserts and spent fuel assembly inserts. Early in 1999, RG&E was made aware of a potential analytical approach for resolving the issue which would eliminate the need for a permanent modification. A scoping study was contracted which evaluated a number of changes to the methodology and inputs to the current criticality analysis. This scoping study was completed in July of 1999 with a preliminary conclusion that new proposed storage requirements could be met by moving spent fuel assemblies to new locations within the existing storage racks and taking credit for a limited amount of soluble boron. This would allow resolution of this issue without requiring a modification to the storage racks.

As the result of this preliminary conclusion, RG&E is requesting a revision to the date specified in the Specification 4.3.1.1.b note associated with maintaining spent fuel pool boron concentration ≥2300 ppm at all times until a permanent resolution can be implemented. The status of this issue was discussed with NRC staff during a telephone conference call held August 11, 1999, at which time it was stated that the NRC review and approval of new proposed storage requirements could take approximately one year to complete. RG&E is currently expecting a final criticality analysis to be completed and the new proposed SFP storage requirements amendment request to be submitted to the NRC by March 10, 2000. Therefore, RG&E is requesting that the specified date in ITS be revised to June 30, 2001, which also accounts for any potential delays in the submittal or approval process.

C. JUSTIFICATION OF CHANGES

This section provides the justification for all changes described in Section A above and shown on Attachment IV. The justifications are organized based on whether the change is: more restrictive (M), less restrictive (L), administrative (A), or the requirement is relocated (R). The justifications listed below are also referenced in the Technical Specification(s) which are affected (see Attachment III).

C.1 Administrative

A.1 The date specified in the Specification 4.3.1.1.b note associated with maintaining spent fuel pool boron concentration ≥2300 ppm at all times until a permanent resolution can be implemented will be revised to June 30, 2001. The basis for the temporary compensatory measure was detailed in Reference 3 and received NRC approval by Reference 2. Extending the date is required to allow for an analytical resolution of the boraflex degradation issue without requiring a plant modification. The conclusion in Reference 3 that a boron dilution event is not credible remains valid and therefore the extension of the completion date is of an administrative nature.

There are no less restrictive (L), more restrictive (M), or relocated (R) changes associated with this LAR.

D. ENVIRONMENTAL IMPACT CONSIDERATION

RG&E has evaluated the proposed changes and determined that:

1. The changes do not involve a significant hazards consideration as documented in Attachment II; and

2. The changes do not involve a significant change in the types or significant increase in the amounts of any effluent that may be released offsite since the change is administrative in nature; and

3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure since no new or different type of equipment are required to be installed as a result of this LAR.

Accordingly, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed changes is not required.

E. REFERENCES

1. Letter from Robert C. Mecredy (RG&E) to Guy S. Vissing (NRC), "Boraflex Degradation", dated March 30, 1998.

2. Letter from Guy S. Vissing (NRC) to Robert C. Mecredy (RG&E), "Issuance of Amendment No. 72 to Facility Operating License No. DPR-18, R. E. Ginna Nuclear Power Plant", dated July 30, 1998.

3. Letter from Robert C. Mecredy (RG&E) to Guy S. Vissing (NRC), "Application for Amendment to Facility Operating License, Revised Spent Fuel Pool Storage Requirements, Revision 1", dated April 27, 1998.

Attachment II
R.E. Ginna Nuclear Power Plant

SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

The proposed change to the Ginna Station Improved Technical Specifications as identified in Attachment I Section A and justified by Section C has been evaluated with respect to 10 CFR 50.92(c) and shown not to involve a significant hazards consideration as described below.

Evaluation of Administrative Change

The administrative change associated with the revision of the date specified in the Specification 4.3.1.1.b note associated with maintaining spent fuel pool boron concentration ≥2300 ppm at all times until a permanent resolution can be implemented does not involve a significant hazards consideration as discussed below:

1) Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the required completion date for resolution of a boraflex degradation issue. As described in the bases for LCO 3.7.12, increases in spent fuel pool temperature, with the corresponding decrease in water density and void formation from boiling, will generally result in an decrease in reactivity due to the decrease in moderation effects. The only exception are temperature bands where positive reactivity is added as a result of the high boron concentration. This effect is bounded by the reactivity added as a result of a misloaded fuel assembly. With respect to the more limiting dropped fuel assembly accidents, boraflex neutron absorber panels were originally assumed in the criticality analysis. Requiring a high concentration of soluble boron in place of boraflex panels ensures that the spent fuel pool remains subcritical with $k_{eff} \leq 0.95$ for these accidents. Fuel assembly movement will continue to be controlled in accordance with plant procedures and LCO 3.7.13 which specifies limits on fuel assembly storage locations. Periodic surveillances of boron concentration are required every 7 days with level verified every 7 days during fuel movement per LCO 3.7.11. Due to the large inventory within the spent fuel pool, dilution of the soluble boron within the pool is very unlikely without being detected by operations personnel during auxiliary operator rounds or available level detection systems. There is also a large margin between the analyzed boron concentration to maintain the pool subcritical $k_{eff} \leq 0.95$ and the current required value. The extension of the date does not invalidate this conclusion. Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

2) Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. Revising the date for requiring that 2300 ppm boron be maintained in the spent fuel pool, to address any potential dissolution of boraflex in neutron absorber panels, does not create the possibility of a new or

different kind of accident since the spent fuel pool is required to be maintained with a high boron concentration. Assuming a boron dilution event to the level required to reach k_{eff}>0.95 conditions within the spent fuel pool would require either overfill of the pool or a controlled feed and bleed process with unborated water. In both cases, more than 105,000 gallons of unborated water would be required to reach k_{eff}>0.95. There is no source of unborated water of this size available to reach the spent fuel pool under procedural control or via a pipe break other than a fire water system pipe break or SW leak through the spent fuel pool heat exchangers. However, there are numerous alarms available within the control room to indicate this condition including high spent fuel pool water level and sump pump actuations within the residual heat removal pump pit (lowest location in the Auxiliary Building). Auxiliary operators also perform regularly scheduled tours within the Auxiliary Building. This provides sufficient time to terminate the event such that there is no credible spent fuel pool dilution accident. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

3) Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. High levels of soluble boron in the spent fuel pool provides a significant negative reactivity such that k_{eff} is maintained ≤0.95. The proposed surveillance frequency will ensure that the necessary boron concentration is maintained. A boron dilution event which would remove the soluble boron from the pool has been shown to not be credible. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the preceding information, it has been determined that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

Attachment III
R.E. Ginna Nuclear Power Plant

Mark-up of Existing Ginna Station Technical Specifications

Included pages:

4.0-2

4.0 DESIGN FEATURES

4.2 Reactor Core (continued)

4.2.2 Control Rod Assemblies

The reactor core shall contain 29 control rod assemblies. The control material shall be silver indium cadmium.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.05 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water*, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

c. Consolidated rod storage canisters may be stored in the spent fuel storage racks provided that the fuel assemblies from which the rods were removed meet all the requirements of LCO 3.7.13 for the region in which the canister is to be stored. The average decay heat of the fuel assembly from which the rods were removed for all consolidated fuel assemblies must also be ≤ 2150 BTU/hr.

4.3.1.2 The new fuel storage dry racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.05 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and

A.1

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR.

June 30, 2001

* Until December 31, 1999, the spent fuel storage racks shall be maintained with a $k_{eff} \leq 0.95$ when flooded with water containing ≥ 2300 ppm soluble boron

(continued)

Attachment IV
R.E. Ginna Nuclear Power Plant

Proposed Ginna Station Technical Specifications

Included pages:

4.0-2

4.0 DESIGN FEATURES

4.2 Reactor Core (continued)

4.2.2 Control Rod Assemblies

The reactor core shall contain 29 control rod assemblies. The control material shall be silver indium cadmium.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.05 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water*, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

c. Consolidated rod storage canisters may be stored in the spent fuel storage racks provided that the fuel assemblies from which the rods were removed meet all the requirements of LCO 3.7.13 for the region in which the canister is to be stored. The average decay heat of the fuel assembly from which the rods were removed for all consolidated fuel assemblies must also be ≤ 2150 BTU/hr.

4.3.1.2 The new fuel storage dry racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.05 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR.

* Until June 30, 2001, the spent fuel storage racks shall be maintained with a $k_{eff} \leq 0.95$ when flooded with water containing ≥ 2300 ppm soluble boron

(continued)