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VISSING,G.S.

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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: Report of Facility Changes, Tests, and Experiments
Conducted Without Prior Commission Approval
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Vissing:

The subject report is hereby submitted as required by 10 CFR 50.59(b). The enclosed report contains descriptions and summaries of the safety evaluations conducted in support of proposed changes to the facility and procedures described in the UFSAR and special tests, from January 1998 through June 1999, performed under the provisions of 10 CFR 50.59.

Very truly yours,

Robert C. Mecredy

Robert C. Mecredy

1/1

Attachment

260005

IE47

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1999 REPORT

OF

FACILITY CHANGES, TESTS, AND EXPERIMENTS

CONDUCTED WITHOUT PRIOR NRC APPROVAL

FOR JANUARY 1998 THROUGH JUNE 1999

UNDER THE PROVISIONS OF 10 CFR 50.59

R.E. GINNA NUCLEAR POWER PLANT
DOCKET NO. 50-244
ROCHESTER GAS AND ELECTRIC CORPORATION

DATED OCTOBER 20, 1999

SEV-1090

TECHNICAL SPECIFICATION BASES CHANGE FOR SCREENHOUSE BAY LOWER TEMPERATURE LIMIT

The purpose of this safety evaluation is to address changing the Technical Specification Bases for LCO 3.7.8. This change is being made to better correlate the lake (i.e., ultimate heat sink) environmental conditions with plant operations. Specifically, the minimum screenhouse bay operability requirements will be changed. Revision 1 of this evaluation changed the screenhouse bay temperature from "Temperature ≥ 35°F..." to "Temperature ≥ 32°F..." in accordance with a sensitivity analysis. Revision 2 of this evaluation supports a change in the minimum operating temperature of the service water system from 32°F to 30°F.

The probability of occurrence of an accident previously evaluated in the SAR is not increased, because the change does not impact the capability to meet the accident analysis nor does it introduce any effects that could increase the probability of an accident. In addition, the reduction in the temperature does not adversely impact the ability of any equipment to perform their intended safety function.

The consequences of an accident previously evaluated in the SAR are not increased, because the radiological consequences meet the required acceptance criteria, thus the consequences are acceptable. This change does not introduce the possibility of an accident or equipment malfunction of a different type than previously evaluated in that the change affects only the parametric value used by current analyses.

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR is not increased, because the change affects a parameter associated with the SW system fluid and is minor in nature. A design analysis evaluated the impact of the fluid temperature change from 32°F to 30°F. The results did not show a reduction of the structural integrity of any components relied upon and hence the design basis would not be affected by the reduction in temperature limit to 30°F.

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR are not increased, because the change proposed does not affect the ability of any equipment relied upon to mitigate consequences from performing their functions. The structural integrity of critical components and their capability is not impacted by the SW temperature. Revision 1 of this evaluation examined the impact of a 30°F SW temperature on containment fan cooler performance and the affect on PCT, with the limiting case of PCT remaining less than 2200°F which is the approved criteria for PCT. This is documented in the UFSAR, Section 6.2.2.1.

The possibility of an accident of a different type than evaluated previously in the SAR is not created, because the change does not introduce any new initial conditions, or make any change to

the actuation of accident mitigating equipment.

The possibility of a malfunction of equipment important to safety of a different type than evaluated previously in the SAR is not created, because the temperature reduction from 32°F to 30°F will not reduce any of the performance characteristics of components, such as valves, pumps, or heat exchangers, and will not affect the structural integrity or stress levels of piping or pipe supports. Components receiving service water flow for the purpose of removing heat from the other fluid medium, are not adversely impacted and not subject to freezing, since the fluid medium is air, oil or treated water.

This change does not reduce the margin of safety as defined in the basis for the Technical Specifications, because the slight impact upon PCT does not result in a PCT above the criteria basis. Since all acceptance criteria are met there is no reduction in the margin of safety.

SEV-1094
REPLACEMENT OF RTD INPUT MODULES
IN THE REACTOR PROTECTION RACKS

The electronic components used to generate the T_{ave} and ΔT signals in the Reactor Protection System (RPS) are going to be changed to replace the aging loop modules which have no available replacements. This will require the removal of 20 Foxboro H-line modules which will be replaced with 24 modules manufactured by NUS. The new module arrangement will consist of 16 Resistance-to-Current (R/I) converters, and eight Time Domain Modules (all safety grade analog devices). Each protection channel will have four R/I convertors that will be used for the conversion of Hot leg and Cold leg temperatures in the Reactor Coolant System (RCS), and two Time Domain Modules that will be used to condition the RCS temperature inputs into T_{ave} and ΔT signals. One additional function of the Time Domain Modules will be to provide the required lag time associated with the temperature signal. The insertion of instrument loop lag time provides a compensating factor for the extremely fast responding loop RTDs with respect to the rest of the instrument loop. The lag time factor was part of the original instrument loop response calculation for both the T_{ave} and ΔT signals. The signal outputs of the Time Domain Modules will be identical to the outputs of the existing modules being removed, including lag time, and therefore will have no impact on the function of the loop downstream of the new modules. The T_{ave} and ΔT temperature loop for Channel 2 of the Reactor Protection System have been modified under the first phase of PCR 97-026.

The probability of occurrence of an accident previously evaluated in the SAR is not increased by this proposed modification. The change does not introduce any new failure modes or effects into the affected instrument loop nor does it functionally modify the loop (including delay times, setpoints and uncertainties) or associated RPS and control systems in any way.

The consequences of an accident previously evaluated in the SAR is not increased by this proposed modification. The proposed change does not create any new equipment interactions. Because there are no changes in loop failure modes and effects (note that the replacement equipment is also analog) and no new equipment interactions are added, the change cannot lead to a new type of malfunction.

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR is not increased. The change does not introduce any new failure modes or effects into the affected T_{ave} or ΔT instrument loops nor does it functionally modify the loop or associated RPS and control systems in any way not originally designed for.

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR is not increased. The change does not introduce any new failure modes or effects into the affected T_{ave} and ΔT instrument loops nor does it functionally modify the loop or associated RPS

and control systems in any way not originally designed for.

The possibility for an accident of a different type than any evaluated previously in the SAR is not created. The proposed change does not create any new equipment interactions.

The possibility of a malfunction of equipment important to safety of a different type then evaluated previously in the SAR is not created. The proposed change does not create any new equipment interactions.

The margin of safety as defined in the basis for any technical specification is not reduced by this proposed modification. The Overpower and Overtemperature setpoints, the process by which they are generated, and the total RPS delay time are all unaffected by the change.

SEV-1100
RWST ACCIDENT ANALYSIS UPPER TEMPERATURE LIMIT

The accident analysis assumes a RWST temperature range of 60° to 80°F. Recent temperature measurements in the Auxiliary Building indicate the upper temperature limit should be increased. This evaluation documents the efforts done to increase the upper limit from 80° to 104°F.

Increasing the assumed water temperature from 80° to 104°F does not change the function of the RWST, SI system or spray system. The effect of the temperature increase on the SI system and spray system in terms of available NPSH has been evaluated and determined not to be a concern.

The probability of occurrence of an accident is not increased by the assumption of RWST temperature. The RWST is not an accident precursor and therefore the change in maximum allowable temperature will not affect the probability of occurrence for any accident analysis described in the UFSAR.

The consequences of an accident have not increased because the acceptance criteria for the accident are still met. The peak containment pressure as a result of this change remains below the limit of 60 psig and therefore the control room and off-site dose radiological consequences due to the increase in RWST temperature still satisfy the limits established by GDC 19 and 10CFR100.

The probability of occurrence of a malfunction is not increased by the assumption of RWST temperature. The temperature increase from 80° to 104°F is within the design of the affected systems and therefore there is no change in the likelihood of failure.

The consequences of a malfunction have not increased because the acceptance criteria for the accident are still met. The peak containment pressure as a result of this change remains below the limit of 60 psig and therefore the control room and off-site dose radiological consequences due to the increase in RWST temperature still satisfy the limits established by GDC 19 and 10CFR100.

Increasing the assumed RWST temperature by 24°F does not cause a different type of accident than previously evaluated. The temperature change slightly affects the thermal hydraulic properties of the water which would not cause a new type of accident.

Increasing the assumed RWST temperature by 24°F does not cause a different type of malfunction than previously evaluated. The temperature change slightly affects the thermal hydraulic properties of the water which would not cause a new type equipment malfunction.

The margin of safety is between the acceptance criteria and the ultimate failure point. 60 psig is the acceptance criteria for containment. This value has not been exceeded by increasing the upper limit on RWST temperature. Therefore, there is no change in the margin of safety.

SEV-1102
PCN # 97-4346 SAFETY EVALUATION

This Safety Evaluation describes proposed changes to test procedure PT-60.4. This procedure is used to test the performance of the A Diesel Generator Lube Oil and Jacket Water coolers coincident with the monthly A Diesel Generator run done under PT-12.1. The fouling in the Diesel Generator A heat exchangers is determined analytically from PT-60.4 test measurements using a well developed methodology. The uncertainty in the determination of fouling is strongly dependent on the service water temperature difference across the coolers. In order to reduce the uncertainty in the fouling, the service water will be throttled to approximately 250 gpm. The following changes are evaluated:

> PCN # 97-4346 adds steps to PT-60.4 to unlock and throttle globe valve 4671 during testing of the Diesel Generator A coolers. Diesel Generator A will be declared INOPERABLE for the duration of time that valve 4671 is unlocked and throttled.
>
> PCN # 97-4346 adds a precaution to PT-60.4 to have an observer continually monitor the lubricating oil and jacket water outlet temperatures from Diesel Generator A, and record the values on a ten-minute frequency, whenever the engine is running and the service water is throttled. In the event that the jacket water temperature rises above the alarm setpoint of 182°F or the lubricating oil temperature rises above the alarm setpoint of 195°F, the HCO is informed and test personnel immediately open valve 4671. Test personnel also immediately open valve 4671 if the HCO receives a high-temperature alarm on the MCB.
>
> All other proposed changes to PT-60.4 are inconsequential. They involve installation of additional non-intrusive instrumentation (surface-mounted RTDs) and changes to the frequency and duration at which data is taken. These changes are intended to further improve the accuracy of the tests.

The proposed changes do not increase the probability of occurrence of an accident previously evaluated in the SAR. The emergency diesel generator is not an accident initiator, and temporarily throttling service water to the diesel generator coolers will not change the configuration of any other system in such a way as to impact the probability of another system initiating an accident.

The proposed changes do not increase the consequences of an accident previously evaluated in the SAR. Diesel Generator A, although INOPERABLE, is expected to function normally, and can be returned to OPERABLE status by opening and locking valve 4671. In addition to the normal MCB alarm, Diesel Generator A will be continually monitored locally to verify that the lube oil and jacket water temperatures do not exceed the alarm setpoint values. In the event that temperatures reach alarm setpoints, test personnel will take immediate action to open valve 4671. Therefore,

the probability of failure of Diesel Generator A is no higher than it is during the regular monthly PT-12.1 Surveillance Test.

The proposed changes do not increase the probability of occurrence of a malfunction of equipment important to safety. The tested emergency diesel generator can be restored to operable status immediately by opening and locking valve 4671. Since this corresponds to the analyzed configuration of the plant, there is no increased probability of malfunction of the diesel generator or any other equipment.

The proposed changes do not increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR. Accident analyses already assume the loss of a diesel generator.

The proposed changes do not increase the probability of an accident of a different type than any evaluated previously in the SAR. The proposed changes involve minor modifications to a test that is routinely carried out. The most severe occurrence would be the tripping of Diesel Generator A to prevent it from overheating. Contingent actions stemming from a diesel generator trip are already available.

The proposed changes do not increase the possibility of a malfunction of equipment important to safety of a different type then evaluated previously in the SAR. The Diesel Generator A lube oil and jacket water temperatures will not be allowed to rise above the currently established alarm setpoints. It has been established by the vendor that these are acceptable operating temperatures for the diesel engines.

The margin of safety as defined in the basis for any technical specification is not reduced, since no Technical Specifications are violated. Since the normal configuration of the system can be immediately re-established as necessary to provide adequate cooling to the diesel generator, there is no reduction in any safety margins.

SEV-1103
VACUUM FILL OF THE REACTOR COOLANT SYSTEM

Industry wide use of the vacuum fill method of increasing the reactor coolant system (RCS) level from mid loop to the narrow range on the pressurizer is to be evaluated.

This procedure is to be used during mode 5 prior to and during the final RCS loop fill process. It will be installed only during this process and will be removed when RCS refill is complete. The vacuum fill process will be incorporated into procedures O-2.3.1 and O-1B. The present method of RCS system fill requires a long and complicated vent procedure. This modification will allow a vacuum to drawn on the RCS when at midloop in order to allow the RCS to be filled without the need for venting.

The initial conditions for RCS vacuum fill are established during RCS low loop conditions. The RCS level is to be maintained between 10-12 inches indicated loop level and RCS temperature will be maintained <85°F throughout the vacuum venting process. Low loop procedure O-2.3.1 will be in effect, the level band restricts RHR flow to 800 gpm.

The vacuum operation will consist of a vacuum pump connected via 2 inch diameter vacuum rated hose to the pressurizer relief outlet piping to the pressurizer relief tank (PRT). There will be an option vacuum hose for the reactor vessel head vent. The pressurizer PORV and Block valves will be open to allow the PRT gas space and pressurizer relief tailpipes to be connected to the RCS. The pressurizer and PRT vent manifolds will supply the vacuum taps for reactor vessel level sightglass and RCS loop level instrumentation.

The RCS vacuum vent and fill procedure will maintain positive control over the RCS vents and the low temperature overpressure system (LTOP) alignment. The procedure maintains control over all equipment that can inject into the RCS and increase its pressure. This assures RCS boundary protection at low temperatures, therefore the initial conditions and probability of occurrence for any accident analysis previously evaluated in the UFSAR have not changed and are valid.

The RCS vacuum vent and fill procedure maintains control of reactor coolant boron, density, or operating temperature. The procedure monitors the dilution and boration paths to the RCS. The vacuum process will not influence coolant boron concentration, therefore the initial conditions and consequences of an accident previously described in the UFSAR for reactivity insertion have not changed and are valid.

The wall thickness of the pressurizer, steam generators and U-tubes, reactor coolant pumps and associated components exposed to the vacuum is sufficient to maintain the integrity of the systems during vacuum venting, and after the fill process is complete.

The integrity of the reactor coolant pump seals is assured by maintaining a positive pressure at the number 1 seal inlet area. The pressurizer relief tank is designed to withstand a full vacuum. The tank is equipped with an internal support for the rupture disk to prevent the damage to the disk. Therefore the integrity of the RCS remains unchanged and the probability of occurrence of a malfunction of equipment important to safety is not increased.

The containment isolation system will remain unaffected by this change. The system will still be able to achieve containment closure within the allowed 2 hour time period of generic letter 88-17, and be capable of preventing a radiation release within 10 CFR 100 limits. Therefore the ability to isolate containment during reduced RCS inventory operations remains unchanged and the probability of occurrence of a malfunction of equipment important to safety is not increased.

The ability of the Residual Heat Removal system to provide for core cooling when the RCS is in a reduced inventory condition will not change. The NPSH available for the RHR pumps is greater than required, per Design Analysis DA-ME-97-080,Rev 3, therefore the capability of RHR system to provide core cooling will not be adversely affected. The WCAP-11916 (section 2.5) was reviewed to verify that operating in midloop with the RCS at a vacuum did not invalidate its analysis. The analyses for vortex formation were most sensitive to fluid velocity with the density and viscosity of the fluid as secondary affects. None of these parameters are affected by the RCS being under a vacuum. The analysis therefore remains valid. There is no increase in the consequences of a malfunction previously evaluated in the UFSAR.

The RHR, charging, and safety injection systems will all be lined up and controlled per Operations procedure O-2.3.1 "Draining and Operating at Reduced Inventory in the Reactor Coolant System". This procedure implements RG&E's response to generic letter 88-17 concerns. The RCS is maintained in an analyzed condition per WCAP 11916. The RCS and mitigating systems are lined up and operating per established procedures. Therefore this system configuration and procedure does not create the possibility for an accident of a different type than any evaluated previously in the UFSAR.

With the steam generator intact and the pressurizer manway installed, the criteria is met for the RCS intact configuration. This configuration was analyzed and is one of the configurations that WCAP-11916 and Generic Letter 88-17 reviews. Therefore, the possibility of a malfunction of the RCS boundary of a different type than evaluated previously in the UFSAR is not created.

A KYPIPE analysis (noted on "Expeditious Actions" response to the NRC, dated January 4, 1997) of the RHR system verified that the gravity feed method would place approximately 7000 gallons of water in the RCS if initiated within 16 minutes of the event and assuming an intact, unvented RCS, that would pressurize according to the WCAP 11916 fig. 3.3.1-1. This was based on the decay heat load at 48 hours after shutdown. The vacuum fill evolution is taking place at greater than 300 hours after shutdown, the estimated time to saturation is approximately 27 minutes

and there is additional time needed to build up pressure in the RCS. The open PORV's and having one steam generator filled will further delay the increase in RCS pressure. Therefore additional time is available for the operators to increase RCS level using gravity feed. A pressure of approximately 42 psia was found to stop gravity feed flow from the RWST. The final recovery action of restarting RHR would occur after level is increased. Performing the RCS vacuum vent and fill under these conditions does not reduce the margin of safety as defined in the basis for any Technical Specification.

PCN # 97-4347 SAFETY EVALUATION

This Safety Evaluation describes proposed changes to test procedure PT-60.5. This procedure is used to test the performance of the Diesel Generator B Lube Oil and Jacket Water coolers coincident with the monthly Diesel Generator B run done under PT-12.2. The fouling in the Diesel Generator B heat exchangers is determined analytically from PT-60.5 test measurements using a well developed methodology. The uncertainty in the determination of fouling is strongly dependent on the service water temperature difference across the coolers. In order to reduce the uncertainty in the fouling, the service water will be throttled to approximately 250 gpm. The following changes are evaluated:

> PCN # 97-4347 adds steps to PT-60.5 to unlock and throttle globe valve 4672 during testing of the Diesel Generator B coolers. Diesel Generator B will be declared INOPERABLE for the duration of time that valve 4672 is unlocked and throttled.
>
> PCN # 97-4347 adds a precaution to PT-60.5 to have an observer continually monitor the lubricating oil and jacket water outlet temperatures from Diesel Generator B, and record the values on a ten-minute frequency, whenever the engine is running and the service water is throttled. In the event that the jacket water temperature rises above the alarm setpoint of 182°F or the lubricating oil temperature rises above the alarm setpoint of 195°F, the HCO is informed and test personnel immediately open valve 4672. Test personnel also immediately open valve 4672 if the HCO receives a high-temperature alarm on the MCB.
>
> All other proposed changes to PT-60.5 are inconsequential. They involve installation of additional non-intrusive instrumentation (surface-mounted RTDs) and changes to the frequency and duration at which data is taken. These changes are intended to further improve the accuracy of the tests.

The proposed changes do not increase the probability of occurrence of an accident previously evaluated in the SAR. The emergency diesel generator is not an accident initiator, and temporarily throttling service water to the diesel generator coolers will not change the configuration of any other system in such a way as to impact the probability of another system initiating an accident.

The proposed changes do not increase the consequences of an accident previously evaluated in the SAR. Diesel Generator B, although INOPERABLE, is expected to function normally, and can be returned to OPERABLE status by opening and locking valve 4672. In addition to the normal MCB alarm, Diesel Generator B will be continually monitored locally to verify that the lube oil and jacket water temperatures do not exceed the alarm setpoint values. In the event that temperatures reach alarm setpoints, test personnel will take immediate action to open valve 4672. Therefore,

the probability of failure of Diesel Generator B is no higher than it is during the regular monthly PT-12.2 Surveillance Test.

The proposed changes do not increase the probability of occurrence of a malfunction of equipment important to safety. The tested emergency diesel generator can be restored to operable status immediately by opening and locking valve 4672. Since this corresponds to the analyzed configuration of the plant, there is no increased probability of malfunction of the diesel generator or any other equipment.

The proposed changes do not increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR. Accident analyses already assume the loss of a diesel generator.

The proposed changes do not increase the probability of an accident of a different type than any evaluated previously in the SAR. The proposed changes involve minor modifications to a test that is routinely carried out. The most severe occurrence would be the tripping of Diesel Generator B to prevent it from overheating. Contingent actions stemming from a diesel generator trip are already available.

The proposed changes do not increase the possibility of a malfunction of equipment important to safety of a different type then evaluated previously in the SAR. The Diesel Generator B lube oil and jacket water temperatures will not be allowed to rise above the currently established alarm setpoints. It has been established by the vendor that these are acceptable operating temperatures for the diesel engines.

The margin of safety as defined in the basis for any technical specification is not reduced, since no Technical Specifications are violated. Since the normal configuration of the system can be immediately re-established as necessary to provide adequate cooling to the diesel generator, there is no reduction in any safety margins.

SEV-1105
VACUUM EFFECTS ON RCS INSTRUMENTATION DURING RCS VACUUM VENT AND FILL

The effects of having a vacuum on the Reactor Coolant System (RCS) instrumentation during the RCS vacuum vent and fill evolution are to be evaluated. The instrumentation will be exposed to RCS temperatures of 70 - 85 °F. The pressure will range from atmospheric to 28 inches of Hg vacuum or 0.948 psia. The RCS loop will be initially at the mid loop level. This level is 10 inches using local level indication and is at the 246' 10" elevation. The time duration of the exposure to vacuum is less than 6 hours. Once the RCS level is in the 80% (180 inches) wide range in the pressurizer, the vacuum will be removed and the system will be returned to normal operational pressures.

The RCS vacuum vent and fill procedure will maintain positive control over the RCS vents and the low temperature overpressure system (LTOP) alignment. The procedure maintains control over all equipment that can inject into the RCS and increase its pressure. This assures RCS boundary protection and RCS instrument operability at low temperatures per UFSAR chapter 5.2.2. The RCS instrument system will continue to accurately monitor and display the process variables needed to verify RCS parameters. Therefore the initial conditions and probability of occurrence for any accident analysis previously evaluated in the UFSAR have not changed.

The RCS vacuum vent and fill procedure maintains control of reactor coolant boron, density, and operating temperature. The procedure monitors the dilution and boration paths to the RCS. The RCS instrument system will continue to accurately monitor and display the process variables needed to verify RCS parameters. The vacuum process will not influence coolant boron concentration, therefore the initial conditions and consequences of an accident previously described in the UFSAR for reactivity insertion in chapters 15.4.4.2.2 or 15.4.4.2.6 have not changed and are valid.

The wall thickness of the RCS process instrumentation and sensing lines and associated components exposed to the vacuum is sufficient to maintain the integrity of the systems during vacuum venting, and after the fill process is complete. The integrity of the reactor coolant pump seal instrumentation is assured by maintaining a positive pressure at the number one seal inlet area. The pressurizer relief tank instrumentation is designed to withstand a full vacuum. The tank is equipped with an internal support for the rupture disk to prevent the damage to the disk. Therefore the integrity of the RCS instrumentation remains unchanged and the probability of occurrence of a malfunction of equipment important to safety is not increased.

The containment isolation system and its associated instrumentation will remain unaffected by this change. The system will still be able to achieve containment closure within the allowed 2 hour time period of generic letter 88-17, and be capable of preventing a radiation release within 10 CFR 100

limits. Therefore, the ability to isolate containment during reduced RCS inventory operations remains unchanged and the probability of occurrence of a malfunction of equipment important to safety is not increased.

The ability of the Residual Heat Removal system to provide for core cooling when the RCS is in a reduced inventory condition will not change. The NPSH available for the RHR pumps is greater than required, therefore the capability of RHR system to provide core cooling will not be adversely affected. The WCAP-11916 (section 2.5) was reviewed to verify that operating in midloop with the RCS at a vacuum did not invalidate its analysis. The analyses for vortex formation were most sensitive to fluid velocity with the density and viscosity of the fluid as secondary affects.

The RCS and RHR instrument systems will continue to accurately monitor and display the process variables needed to verify their parameters. None of these parameters are affected by the RCS being under a vacuum. The analysis therefore remains valid. There is no increase in the consequences of a malfunction previously evaluated in the UFSAR.

The RHR, charging, and safety injection systems will all be lined up and controlled per Operations procedure O-2.3.1 "Draining and Operating at Reduced Inventory in the Reactor Coolant System". This procedure implements RG&E's response to generic letter 88-17 concerns. The RCS is maintained in an analyzed condition per WCAP 11916. The RCS and mitigating systems are lined up and operating per established procedures. Therefore this system configuration and procedure does not create the possibility for an accident of a different type than any evaluated previously in the UFSAR.

With the steam generator intact and the pressurizer manway installed, the criteria is met for the RCS intact configuration. This configuration was analyzed and found acceptable in WCAP-11916. Therefore, the possibility of a malfunction of the RCS boundary of a different type than evaluated previously in the UFSAR is not created.

The RCS vacuum vent and fill process does not require a change to Ginna Technical Specifications. RCS pressure and temperature limits as stated in the Pressure Temperature Limits Report (PTLR) are not exceeded. The RCS and RHR instrument systems will continue to accurately monitor and display the process variables needed to verify their parameters. The shutdown requirements and PORV operability limits for the RCS are maintained. The margin of safety for the reactor coolant pressure boundary as defined by the ASME code for wall thickness, stress limits, integrity of systems and components is maintained.

SEV-1108
CYCLE 27 RELOAD

Cycle 27 consists of 41 new fuel assemblies from feed regions 29A, 29B, 29C, and 29D. This safety evaluation is valid for an end-of-cycle 26 burnup of 15,200 to 16,200 MWD/MTU and Cycle 27 burnup not to exceed 16,517 MWD/MTU without additional analysis. Cycle 27 characteristics are described in more detail in the "Reload Safety Evaluation - Cycle 27, Redesign". The fuel assemblies for Cycle 27 are mechanically the same as the Cycle 26 fuel assemblies except for the following.

1. The use of annular pellets in the axial blankets,
2. A reduction in backfill pressure in Integral Fuel Burnable Absorber (IFBA) rods to 100 psig,
3. Grooved top and bottom fuel rod end plugs,
4. 3-tab inconel grids,
5. New top nozzle spring pack design.

The Cycle 27 reload will not increase the probability of occurrence of an accident because the reload core does not affect accident initiators or equipment operation. The reload core does not cause a pipe to break or equipment to malfunction. Therefore, the reload core can not increase the probability of an accident. The fuel design change satisfies existing design criteria; therefore, the probability of failure does not increase. Gap reopening does not affect accident initiators.

The Cycle 27 reload does not increase the probability of a malfunction of equipment because the reload core does not affect equipment operation. The reload core does not cause equipment to malfunction. The fuel design change satisfy existing design criteria; therefore, the probability of failure does not increase. Gap reopening is not expected to lead to fuel failure. Violating the gap reopening SAFDL criteria does not result in exceeding the 17% oxidation limit.

The Cycle 27 reload does not increase the consequences of an accident because the core characteristics are bounded by parameters assumed in the accident analysis. When deviations occurred, reanalysis was performed to show the acceptance criteria was still satisfied. The fuel assembly changes do not degrade fuel performances. The resulting changes are still within acceptable ranges. Gap reopening could affect the 17% oxidation limit; however, this is not possible until the screening limit has been reached. Analysis has been performed which demonstrates compliance with the limit for all of Cycle 27.

The Cycle 27 reload does not increase the consequences of malfunction of equipment because the core characteristics are bounded by parameters assumed in the accident analysis. When deviations occurred, reanalysis was performed to show the acceptance criteria was still satisfied. The fuel assembly changes do not degrade fuel performances. The resulting changes are still within acceptable ranges. Gap reopening does not affect the consequences of equipment malfunction. For example, the consequences of a pump failure is not affected by gap reopening.

The Cycle 27 reload and fuel assembly changes do not cause a new type accident because the core parameters are bounded by those assumed in accident analysis and design parameters are still within the assumed ranges. Gap reopening is not an accident initiator.

The Cycle 27 reload and fuel assembly changes do not cause a new type of malfunction because the core parameters are bounded by those assumed in accident analysis and design parameters are still within the assumed ranges. Previous analysis assumed no gap reopening for simplicity. Analyses with gap reopening show acceptable consequences. Therefore, this condition is acceptable provided continued compliance with the 17% oxidation limit is maintained.

Since the assumptions in the safety and accident analysis including those related to the core design are bounding for the Cycle 27 reload, the conclusions in the Ginna UFSAR remain appropriate and the regulated acceptance criteria for the accident analysis has not been violated. There is no reduction in the margin of safety as defined in the basis for any Technical Specification.

No gap reopening is a Westinghouse design criteria used to simplify the design process. Analyses with gap reopening show all aspects of plant safety analyses remain bounding. The screening criteria provides the point at which compliance with the 17% oxidation needs to be re-evaluated. The plant specific analysis demonstrates continued compliance with the 17% oxidation criterion throughout Cycle 27.

SEV-1109

NEW PROCEDURE PT-60.3A, "CONTAINMENT RECIRCULATION FAN COOLER PERFORMANCE TEST"

This Safety Evaluation describes new procedure PT-60.3A. This procedure was developed to provide an simplified alternative to procedure PT-60.3. Simplification was desired to reduce the number of people and amount of equipment that would be required in containment to facilitate on-power testing. The new procedure only provides information necessary to determine the fouling of the Containment Recirculation Fan Coolers (CRFC). It DOES NOT test the CRFC motor coolers.

The actions in the procedure that have potential safety-significance include:

1. Throttling the service water flow to each CRFC down to ~300 gpm from the usual value of ~1200 gpm. This is only done to one CRFC at a time, and the CRFC is declared inoperable.

2. Isolation of service water flow to the fan motor cooler of the CRFC being tested. Again, the CRFC is declared inoperable when the motor cooler flow is isolated.

3, Installation and removal of intrusive test instrumentation (differential pressure cells). This will periodically cause the control room operators to get low flow alarms on FIA-2033, FIA-2034, FIA-2035, and FIA-2036. The operators are informed before these manipulations are done.

4. Positioning and repositioning of A-3.3 Containment Isolation Boundaries.

PT-60.3A does not increase the probability of occurrence of an accident previously evaluated in the SAR. The CRFCs and associated containment HVAC equipment are not accident initiators.

PT-60.3A does not increase the consequences of an accident previously evaluated in the SAR since accident analyses already assume the loss of a train of containment HVAC, and the inoperable duration of any CRFC will be much less than the LCO 3.6.6 allowed time of 7 days.

PT-60.3A does not increase the probability of occurrence of a malfunction of equipment important to safety. PT-60.3A does make a train of containment HVAC inoperable, which is already assumed in accident analyses. Manipulations on other systems, other than the service water supply to the inoperable train of CV HVAC, are not performed as part of the PT-60.3A procedure.

PT-60.3A does not increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR. Accident analyses already assume the loss of a train of

containment HVAC, and the environmental qualification profile is met with a containment HVAC train out of service. Moreover, manipulations of other systems and equipment important to safety are not performed as part of the PT-60.3A procedure, so there is no associated increase in probability or consequences.

PT-60.3A does not increase the probability of an accident of a different type than any evaluated previously in the SAR. The procedure involves manipulation of service water system valves in the supply to an inoperable CRFC, entry into the enclosure of the inoperable CRFC, and installation of test equipment only.

PT-60.3A does not increase the possibility of a malfunction of equipment important to safety of a different type than evaluated previously in the SAR. The procedure involves manipulation of service water system valves in the supply to an inoperable CRFC, entry into the enclosure of the inoperable CRFC, and installation of test equipment only. No other equipment is manipulated or expected to malfunction as a result of this procedure.

The margin of safety as defined in the basis for any technical specification is not reduced. PT-60.3A only affects a single train of containment HVAC (including the associated post-accident charcoal system), which is declared inoperable under LCO 3.6.6 for testing. With the exception of service water supply to the inoperable train, no other systems are affected by the testing.

The inoperable CRFC operates during the testing, and the reduced service water flow rate does not have a significant effect on the heat removal capability of the CRFC. The operable CRFCs are also available to maintain containment temperature below the normal operating Technical Specification limit of 120°F as defined in LCO 3.6.5.

SEV-1111

FUEL ASSEMBLY REPAIR PROCEDURE RF-73.1

In order to repair (reconstitute) selected fuel assemblies the preferred technique is to remove the top nozzle which allows access to the fuel pins. This differs from past methods of reconstitution which involved turning the fuel assembly upside down and removing the bottom nozzle. The removable top nozzle has been incorporated into Ginna fuel designs and it is desirable to utilize this method of reconstitution.

Fuel reconstitution is accomplished by removing defective rods and replacing them with "dummy" stainless steel rods. The acceptability of using a reconstituted fuel assembly in the reactor is not covered by this safety evaluation as that will be covered by a revision to the reload safety evaluation. This evaluation covers the process of reconstitution only.

The general process for reconstitution is as follows: Once a fuel assembly has been identified as a leaker and the defective pin(s) identified by a UT inspection the fuel assembly is transported to the new fuel elevator. The new fuel elevator will be outfitted with a special reconstitution basket that is compatible with the reconstitution tooling. Once the fuel assembly has been placed in the elevator the elevator will be raised to a height where the top nozzle lock tubes can be removed. This elevation is approximately 9 feet below the water surface. The lock tubes and top nozzle are then removed and the fuel assembly lowered to the rack elevation. Next the defective fuel pins are removed and placed in the existing failed fuel storage container. Dummy rod(s) are inserted in the location(s) previously occupied by the defective pins and the fuel assembly raised again to the 9 foot elevation and the top nozzle and lock tubes are reinstalled. The assembly is then lowered and transferred to its desired location.

The Ginna UFSAR states that the new fuel elevator is used for new fuel only. Since this procedure will deviate from that description this safety evaluation is being prepared to describe the additional use of the elevator for fuel repair activities.

Since the assembly to be reconstituted is contained in systems designed to handle its associated geometry and weight the probability of a fuel handling accident or any other accident in SAR is not increased.

Since the fuel assembly will be the only assembly in transit or being worked on during reconstitution activities and the activities performed at less than 23 feet of water coverage are limited in scope so as to not damage any fuel pins the consequences of a fuel handling accident remain bounded by the evaluated accident.

The probability of a malfunction of equipment important to safety is not increased because multiple layers of administrative and physical controls are in place to maintain sufficient water level above

the fuel assembly at all times.

The consequences of a malfunction of equipment important to safety are not increased because sufficient controls have been put in place to preclude overexposure of plant personnel as well as the public from reconstitution activities.

The possibility of an accident of a different type than any previously evaluated in the SAR has not been created because the new fuel elevator has sufficient controls in place to prevent the inadvertent withdrawal of a spent fuel assembly from the water. Any possible breakage of a single fuel rod during the reconstitution process is bounded by the fuel handling accident analysis which assumes all rods in a single assembly are failed.

The use of the new fuel elevator will not create the possibility of a malfunction of equipment important to safety because the adjusted elevator stop will be tested prior to placing a spent fuel assembly into it. Since the elevator is designed for the weight and geometry of the component that is being inserted into it this change does not create the possibility of its malfunction.

Since fuel handling, water level, boron concentration specifications are all maintained within their Technical Specification limits this procedure does not decrease the margin of safety as defined in the basis for spent fuel pool technical specifications.

SEV-1112

ACTION REPORT 97-1846 DISPOSITION FOR MAIN STEAM LINE A AND B CRACK REPAIR AT PENETRATION 401 AND 402

As a result of new ISI inspection methods for integral attachments to piping/components cracks were discovered in the gusset welds of MS penetrations 401 and 402 inside containment. The purpose of this safety evaluation is to review the root cause and corrective action taken as a result of the cracks and determine if the affected systems are operable. This revision of the safety evaluation was performed to update the references to the supporting analysis.

The root cause of the cracks was found to be due to poor weld joint design, referred to as a tee joint, which caused high residual stresses in the heat affected zone of the weld. Heavy presence of oxides is evidence that the cracks have existed for a long time, possibly from original construction initiation. Cracking in tee joints is a well know phenomena (Lamellar tearing) which was identified in the late 1960s for large section structural members. The literature reviewed shows cracks starting from the weld toe and propagating down into the base metal along the heat affected zone. Based on the report, further cracks should not develop since the initiating cause was the welding stresses, not service induced stresses (fatigue). All cracks were found at the outer toe of the weld.

The repair process removed gussets which were located adjacent to the cracked weld to allow access to the pipe wall for defect removal. Cracks were not found in any of the area between the outer toes of the two fillet welds on either side of the three gusset which were removed. The cracks were excavated down to "defect free" base metal and then rewelded to restore the required pipe wall. All repairs were done in accordance with the original plant construction code. The maximum crack depth was found to be less than 5/8" in all cases and started at the weld toe on the pipe. The removed gussets was not re-installed over the repaired pipe area per PCR 97-089, since they were not required to meet the design basis loads.

The FW system was found to have the same penetration design as the MS except with thinner members and smaller fillet welds.The inspections did not reveal any cracks. A review was also done of the remainder of the MS and FW system for other potential tee joint configurations which have the potential for cracks. No other attachments were found which were highly restrained and had weld sizes large enough to generate high residual stresses. A third review was done of the remainder of the plant piping systems and the results showed that the systems did not have a large enough pipe wall thickness or attachment welds to create the high residual stresses.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR are not increased by the proposed repair since the capability of the MS line penetrations to resist design loads has not been reduced beyond what was originally assumed.

The possibility for an accident or malfunction of a different type than evaluated previously in the UFSAR will not be created by the proposed repair. Since the repair meets the original code requirements and design basis, and will not change the function of the penetrations, no new types of accidents or malfunctions would be introduced.

The margin of safety, as defined in the basis of any Technical Specification, is not reduced by the proposed repair since it meets the original design basis and codes.

SEV-1114
CONTAINMENT RECIRCULATING FAN COOLER
COIL REPLACEMENT

The original Westinghouse Sturtevant Containment Recirculating Fan Cooler (CRFC) coils were replaced under EWR 5275 with enhanced design Marlo coils during the 1993 refueling outage. Piping modifications were also made in the vicinity of the coils to ease inspection and maintenance of the coils.

A large number of UFSAR changes were made as a result of the CRFC coil replacement since the heat removal from these units affects the relevant analyses for high-energy line breaks inside containment (e.g. LOCAs and MSLBs).

The original safety evaluation for this EWR was taken to Revision 1, but this later revision was never approved by PORC. This deficiency was discovered during the Service Water (SW) System Safety System Functional Inspection (SSFI) performed by Sargent & Lundy Engineers LLC in April and May 1997. This deficiency was documented by an Action Report. The purpose of Revision 1 was to close out open items identified by the original safety evaluation. Although a number of the Revision 0 open items were addressed by Revision 1, a number of open items were still identified by Revision 1. Close out of these additional open items was documented by inter-office correspondence prior to start-up from the 1993 Refueling Outage.

This document serve as the final 10CFR50.59 Safety Evaluation of record for this plant configuration change. As such, it documents the actions taken in 1993 prior to plant start-up to close-out all of the open items identified in the original EWR 5275 safety evaluation. This safety evaluation will be applicable to the modification as it was completed in 1993; it will NOT attempt to reconcile issues discovered between the time the modification was completed and the present day. All additional changes to the plant subsequent to the CRFC replacement in 1993 would have had their own 10CFR50.59 review/evaluation.

The replacement of the CRFCs by EWR 5275 does not increase the probability of occurrence of an accident previously evaluated in the UFSAR. The CRFCs are used to mitigate the consequences of design basis pipe ruptures inside containment. Additionally, during normal plant operations the replacement CRFCs are capable of performing the same heat removal function and ventilation function as is performed by the original CRFCs. As such normal operation of the CRFCs does not initiate any design basis accidents presently described in the UFSAR.

The replacement of the CRFCs by EWR 5275 does not increase the consequences of an accident previously evaluated in the UFSAR. The replacement CRFCs have enhanced heat removal capability when compared to the original CRFCs. Consequently, containment pressurization transient response to design basis accidents is improved. The peak clad temperature analysis is

not affected by the CRFCs since the minimum containment back-pressure curve used for the cladding analysis included margin which allows it to still be bounding when compared to the analysis with the replacement CRFCs. The control room and off-site dose radiological consequences due to the reduction in CRFC air flow rate under design basis accident conditions still satisfy the limits established by GDC 19 and 10CFR100.

The replacement of the CRFCs by EWR 5275 does not increase the probability of occurrence of a malfunction of equipment important to safety as previously evaluated in the UFSAR. The operation of the CRFCs under normal operating and design basis accident conditions has not been altered and does not directly impact the probability of equipment malfunction for other components. Since the normal operating and design basis containment pressure and temperature profiles are not adversely affected by the CRFC replacement, the EQ pressure and temperature profiles for safety related equipment in containment is still bounding. The SW flows to safety related loads supplied in parallel with the CRFCs is not adversely affected by the CRFC replacement. The electrical loading of the SW pumps, the CRFC fans and consequently the EDGs are not increased by the CRFC replacement.

The replacement of the CRFCs by EWR 5275 does not increase the consequences of occurrence of a malfunction of equipment important to safety as previously evaluated in the UFSAR. The functional heat removal and iodine removal capability of the CRFCs following design basis accidents has not been adversely affected by the CRFC equipment. Therefore, the operation of the CRFCs does not impact equipment mal-functions discussed in the UFSAR.

The replacement of the CRFCs by EWR 5275 does not create the possibility of an accident of a different type than those previously evaluated in the UFSAR. The operation of the CRFCs does not initiate any design basis accidents. The replacement CRFCs are similar in function, design and operation to the original CRFCs. The change in CRFC coil design and tube material have resulted in an enhancement in CRFC functional capability when compared to the original CRFCs.

The replacement of the CRFCs by EWR 5275 does not create the possibility of a malfunction of equipment important to safety of a different type than those previously evaluated in the UFSAR. The operational characteristics of the replacement CRFCs is similar to the original CRFCs. Both CRFC designs utilized finned tubing coils to cool containment air. The basic SW piping configuration to and from the CRFC coils is unchanged as is the air side flow train inside containment. No automatic control features are being added to the replacement CRFC coil design. The only changes to the coils is enhanced tubing materials for corrosion and erosion concerns and increased heat removal characteristics due to a different tube bundle design. None of the enhancements incorporated into the new CRFCs can cause a new type of CRFC malfunction when compared to the original CRFCs.

The replacement of the CRFCs by EWR 5275 does not decrease the margin of safety as defined

in the basis for the Ginna Technical specifications. No changes to the Technical Specifications were identified as a result of the CRFC replacement. The peak fuel cladding temperatures still satisfy the 10CFR50.46 requirement of not exceeding 2200°F. Off-site doses due to a design basis LOCA still satisfy the requirements of 10CRF100. Control Room doses due to a design basis LOCA still satisfy the requirements of NRC General Design Criteria 19 related to Control Room Habitability. Peak calculated containment pressures during design basis pipe ruptures are still below the containment design pressure of 60 psig.

SEV-1115
REMOVAL OF CRDMGA AND CRDMGB
REVERSE POWER PROTECTION

UFSAR Section 7.7.1.2.5.1 "Alternating Current Power Connections" takes credit for tripping out an MG set on a reverse power condition. The Control Rod Drive System original design included reverse power protection. This protection was removed (reference TSR 91-167, TM 93-031, and EWR 10322) due to several occurrences of undesired inadvertent tripping of the MG sets. However, this TSR, TM, and EWR neglected to adequately document the 10CFR50.59 evaluation of the removal and to update the UFSAR.

The probability of occurrence of an accident previously evaluated in the SAR is not increased. The change does not introduce any new failure modes or effects into the Control Rod Drive system (not already reviewed in the accident analysis).

The consequences of an accident previously evaluated in the SAR is not increased by this change. The change does not introduce any new failure modes or effects into the Control Rod Drive system.

The probability of occurence of a malfunction of equipment important to safety previously evaluated in the SAR is not increased. The change does not introduce any new failure modes or effects into the Control Rod Drive System.

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR is not increased. The change does not introduce any new failure modes or effects into the Control Rod Drive System.

The possibility for an accident of a different type than any evaluated previously in the SAR is not created. The proposed change does not create any new equipment interactions.

The possibility of a malfunction of equipment important to safety of a different type then evaluated previously in the SAR is not created. The proposed change does not create any new equipment interactions.

The margin of safety as defined in the basis for any technical specification is not reduced by this change. Reactivity control or the ability to drop the rods into the core (if required) is unaffected by this change.

SEV-1116
CHANGE MOV 7443 AND MOV 7444 FROM MOTOR ACTUATION TO MANUAL ACTUATION

MOV 7443 and MOV 7444 are motor operated containment leak test isolation valves. The valves do not require electrical actuation to perform their design function. Due to the increased maintenance associated with motor operated valves, the added cost of maintaining the motor actuators on valves 7443 and 7444 has no benefit for Ginna Station and increases the competitive price of product.

Electrical power for MOV 7443 and MOV 7444 will be removed per PCR 98-012. The motors will be abandoned in place. Electrical cables, conduit and components will be removed as practicable. A valve handle will be installed to allow manual actuation and a means for locking the handle will be provided to prevent tampering and/or mispositioning.

The probability of occurrence of an accident previously evaluated in the SAR is not increased due to removal of the control power and position indication for valves 7443 and 7444. The valves are not individually evaluated in accident mitigation. The valves are currently maintained in a closed position above mode 5 with position indication provided on the Main Control Board. After the modification the valves will continue to be maintained in the closed position however the valves will be locked closed due to the removal of the position indication. Plant configuration and piping remains unchanged and the containment integrity boundary is unaffected.

The consequences of an accident previously evaluated in the SAR will not be increased due to removal of the control power and position indication for valves 7443 and 7444. The valves are currently maintained in a closed position above mode 5 and this will not be changed by this modification. New locking valve handles will be installed to prevent mispositioning. Plant configuration and piping will not be changed by this modification therefore the integrity of the containment boundary is unaffected.

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased due to removal of the control power and position indication for valves 7443 and 7444. The valves will be placed in a locked closed configuration during operation above mode 5, which is consistent with current operational position. There is no change to the mechanical properties of the valves or piping therefore no new malfunctions are being added to the configuration.

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR is not created due to removal of the control power and position indication for valves 7443 and 7444. Since there is no change in mechanical properties and the valves will be maintained in a locked closed position for operation above mode 5 there are no new malfunctions to consider.

The possibility of an accident of a different type than any evaluated previously in the SAR is not created due to removal of the control power and position indication for valves 7443 and 7444. The valves will be procedurally maintained in a locked closed position for operation above mode 5, the same position which the valves are currently positioned.

The possibility of a malfunction of equipment important to safety of a different type then evaluated previously in the SAR is not created due to removal of the control power and position indication for valves 7443 and 7444. The valves will be placed in a locked closed position for operation above mode 5. New valve handles will be installed which have been evaluated by Mechanical Engineering with the determination that the additional weight (approximately three pounds) is negligible and will not cause a component failure during an earthquake. In addition, the handle will be welded to the stem therefore no seismic interactions exist.

The margin of safety as defined in the basis for any technical specification is not reduced due to removal of the control power and position indication for valves 7443 and 7444. Previous technical specification requirements which applied to the valves were per surveillance requirement SR 3.6.3.6 which required verification of proper actuation of the automatic containment isolation function in the control circuitry. This function will be removed and the valves will be maintained in a locked closed position above mode 5.

SEV-1117
INSTALLATION OF SPENT FUEL STORAGE RACKS AND RELATED MODIFICATIONS TO SPENT FUEL POOL

The proposed changes to the facility are as follows:

(a) Remove three old racks with no neutron absorber that currently constitute Region 1 of the spent fuel pool.

(b) Install seven new racks having Borated Stainless Steel as a neutron absorber. Two of the racks will be assigned to increase the capacity of Region 2 and the remaining five racks will be designated as the new Region 1.

(c) Remove obstructions as needed. Obstructions currently identified for removal are as follows:

- Four light funnels attached to the liner (two located on the north wall; one located on the east wall, and one located on the south wall). These light funnels will be shortened to approximately 1/4 in.

- Stubs attached to the liner (several stubs are located on the north, east and south walls of the spent fuel pool). These stubs will be shortened to approximately 1/4 in.

On removal of the old racks, other obstructions may be identified. These potential obstructions will be removed using the same procedures, tools, and administrative controls that are utilized to remove the above obstructions. This will ensure that the probability of puncturing the spent fuel pool liner is as low as reasonably achievable. In the event of a puncturing of the liner, there are procedures and administrative controls necessary to promptly inspect and repair any potential leaks.

(d) Install metal strips with a set of letter/number coordinates called "X-Y Indexing" on the edge of the pool to aid positioning of the spent fuel bridge during fuel shuffling. The X-Y Indexing plates will be bolted to the top of the concrete wall surrounding the spent fuel pool, on the north and south sides. The area at the top of the wall is that between the rail and the liner. Implementation guidelines will ensure that no rebar is cut. There will be tack welds applied on the outer edges of the bolts and the X-Y indexing.

(e) Relocate the support for the spent fuel handling tool further along the east wall to

a position closer to the south wall. This modification will entail removing the existing support for the spent fuel assembly handling tool that is welded to the liner and installing a new support that consists of a horizontal plate supported by a bracket over the curb. The horizontal plate that supports the spent fuel handling tool is identical to the existing one. The bracket will have a bolt on the outer side of the curb.

The scope of this safety evaluation is to primarily address any of the possible temporary configurations of the racks during the installation (a temporary configuration is defined as the geometrical arrangement of any number of racks on the pool floor that is different from the final layout achieved after the end of the installation).

In general, temporary configurations are not explicitly described in the NRC Safety Evaluation (NRC SE) issued by the U.S. NRC to RG&E on July 30, 1998. The NRC SE addresses the final configuration and establishes safety requirements applicable during the installation (e.g. criteria for heavy loads, criticality, radiological, summary of occupational exposure during the installation).

This safety evaluation will provide the basis for determining that the conclusions in the NRC SE are bounding with respect to any of the possible temporary configurations that could develop during the installation, and will also provide the basis that there are no additional unreviewed safety questions by implementing the modifications described above.

Removal of Old Racks and Installation of New Racks:

The NRC SE documents the evaluation of design basis accidents applicable during and after the installation. Training prior to the installation, adherence to procedures, and administrative controls will ensure that the probability of occurrence of the applicable design basis accidents, including drop of heavy loads, will not increase. The probability of occurrence of any of the design basis accidents already documented in the SAR and the NRC SE has not been increased.

This evaluation provides the basis for determining that the consequences of the design basis accidents documented in the NRC SE are bounding with respect to any of the possible temporary configurations that could develop during removal of the old racks and installation of the new racks. All limits and requirements will be met during the modification. The consequences of accidents previously evaluated in the SAR and the NRC SE have not been affected.

The NRC SE outlines the requirements for movements of heavy loads during and after the installation. These requirements will be met during the installation. There is no impact on the malfunction of equipment important to safety. Therefore, the probability of occurrence

of a malfunction of equipment important to safety previously evaluated in the SAR remains unchanged.

The NRC SE outlines the requirements for movements of heavy loads during and after the installation. These requirements will be met during the installation. There is no impact on the malfunction of equipment important to safety. Therefore, the consequences of a malfunction of equipment important to safety previously evaluated in the SAR remain unchanged.

The NRC SE documents the evaluation of design basis accidents applicable during and after the installation. This evaluation provides the basis for determining that the design basis accidents documented in the NRC SE are bounding and still applicable with respect to any of the possible temporary configurations that could develop during removal of the old racks and installation of the new racks. There are no new accidents introduced during the modification. Therefore, the possibility of an accident of a different type than any evaluated previously in the SAR and in the NRC SE is not created.

The NRC SE outlines the requirements for movements of heavy loads during and after the installation. These requirements will be met during the installation. Equipment important to safety will not be physically affected by removal of the old racks and installation of the new racks. There is no impact on the malfunction of equipment important to safety during the modification. Therefore, the possibility of a malfunction of equipment important to safety of a different type than evaluated previously in the SAR is not created.

The NRC SE documents the evaluation of design basis accidents applicable during and after the installation. This evaluation provides the basis for determining that the evaluation of the basis accidents documented in the NRC SE is bounding and still applicable with respect to any of the possible temporary configurations that could develop during removal of the old racks and installation of the new racks. All regulatory requirements and limits set forth in the SAR, the NRC SE, and the Technical Specifications are met during the modification. Therefore, the margin of safety as defined in the basis for any technical specification is not reduced.

Removal of Obstructions:

Training, procedures, and administrative controls are established to ensure that the probability of puncturing the spent fuel pool liner is as low as reasonably achievable. The probability of occurrence of a breach of the liner resulting in a damage similar to that a tornado missile puncturing the liner documented in the UFSAR has not increased.

Training, procedures, and administrative controls are established to ensure that the

probability of puncturing the spent fuel pool liner is as low as reasonably achievable. The consequences of any potential breach of the liner during removal of obstructions are bounded by the consequences of a hypothetical tornado missile puncturing the liner as documented in the tornado missile design basis accident. The consequences of accidents previously evaluated in the SAR have not been affected.

Maintaining the structural integrity of the spent fuel pool liner does not impact the malfunction of equipment related to safety. Therefore, the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR remains unchanged.

Maintaining the structural integrity of the spent fuel pool liner does not impact the malfunction of equipment related to safety. Therefore, the consequences of a malfunction of equipment important to safety previously evaluated in the SAR remain unchanged.

Any potential breach of the spent fuel pool liner during removal of obstructions is bounded by the consequences of a hypothetical tornado missile puncturing the liner as documented in the tornado missile design basis accident. The proposed modification does not introduce a new failure mode not documented in the SAR. Therefore, the possibility of an accident of a different type than any evaluated previously in the SAR is not created.

Maintaining the structural integrity of the spent fuel pool liner does not impact the malfunction of equipment related to safety. Therefore, the possibility of a malfunction of equipment important to safety of a different type than evaluated previously in the SAR is not created.

Any potential breach of the spent fuel pool liner during removal of obstructions is bounded by the consequences of a hypothetical tornado missile puncturing the liner as documented in the tornado missile design basis accident. Therefore, the margin of safety as defined in the basis for any technical specification is not reduced.

X-Y Indexing:

The concrete structure of the spent fuel pool will not be degraded by installing the X-Y Indexing plates. The probability of occurrence of the design basis accidents documented in the SAR and the NRC SE for the spent fuel pool structure has not increased.

The concrete structure of the spent fuel pool will not be degraded by installing the X-Y Indexing plates. The consequences of accidents previously evaluated in the SAR and the NRC SE have not been affected.

The concrete structure of the spent fuel pool will not be degraded by installing the X-Y Indexing plates. There is no impact on the malfunction of equipment important to safety. Therefore, the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR remains unchanged.

The concrete structure of the spent fuel pool will not be degraded by installing the X-Y Indexing plates. There is no impact on the malfunction of equipment important to safety. Therefore, the consequences of a malfunction of equipment important to safety previously evaluated in the SAR remain unchanged.

The X-Y Indexing plates are an attachment to the spent fuel pool structure. The spent fuel pool structure has been evaluated under normal and abnormal conditions as documented in the SAR. The proposed modification does not introduce a new failure mode not analyzed in the SAR. Therefore, the possibility of an accident of a different type than any evaluated previously in the SAR is not created.

The concrete structure of the spent fuel pool will not be degraded by installing the X-Y Indexing plates. There is no impact on the malfunction of equipment important to safety. Therefore, the possibility of a malfunction of equipment important to safety of a different type than evaluated previously in the SAR is not created.

The concrete structure of the spent fuel pool will not be degraded by installing the X-Y Indexing plates. Therefore, the margin of safety as defined in the basis for any technical specification is not reduced.

Relocation of the Support for the Spent Fuel Handling Tool:

The design of the proposed support is similar to the existing one. The probability of a drop of the spent fuel handling tool on the racks has remained unchanged. In the conservative direction, the tool support has been positioned further away from spent fuel racks. The probability of occurrence of the design basis accidents documented in the SAR and the NRC SE for the spent fuel pool structure has not increased.

The consequences of a drop of the spent fuel handling tool on top of spent fuel racks are bounded by the consequences of the Fuel Handling Accident (FHA) documented in the NRC SE.

The consequences of a drop of the tool support on top of the spent fuel racks are bounded by the consequences of the Tornado Missile Accident documented in the NRC SE.

The consequences of accidents previously evaluated in the NRC SE have not been

affected.

The drop of the spent fuel handling tool and/or its support has no impact on the malfunction of equipment important to safety. Therefore, the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR remains unchanged.

The drop of the spent fuel handling tool and/or its support has no impact on the malfunction of equipment important to safety. Therefore, the consequences of a malfunction of equipment important to safety previously evaluated in the SAR remain unchanged.

Identified accidents are the drop of the spent fuel handling tool and/or its support in the spent fuel pool. These accidents are bounded by accidents documented in the NRC SE. The proposed modification does not introduce a new failure mode not analyzed in the SAR and the NRC SE. Therefore, the possibility of an accident of a different type than any evaluated previously in the SAR is not created.

The drop of the spent fuel handling tool and/or its support has no impact on the malfunction of equipment important to safety. Therefore, the possibility of a malfunction of equipment important to safety of a different type than evaluated previously in the SAR is not created.

The consequences of a drop of the spent fuel handling tool on top of the spent fuel racks are bounded by the consequences of the Fuel Handling Accident (FHA) documented in the NRC SE. The consequences of a drop of the tool support on top of the racks are bounded by the consequences of the Tornado Missile Accident documented in the NRC SE. Therefore, the margin of safety as defined in the basis for any technical specification is not reduced.

SEV-1118
SEVERE ACCIDENT MANAGEMENT GUIDANCE (SAMG)
IMPLEMENTATION

The purpose of this change is to implement the Severe Accident Management Guidance (SAMG) at Ginna Station. The SAMGs are designed for use in extreme accident circumstances when the Plant EOPs are no longer effective and core damage is progressing. The new guidelines address the accident management changes necessary to mitigate the consequences of a severe accident that have progressed beyond the plant's design basis. Therefore, the actual SAMGs are likewise considered to be beyond design basis documents and are not subject to 10CFR50.59 review. The procedures addressed by this safety evaluation outline the administrative guidance for implementation and maintenance of the SAMG program, as well as the EOP transitions to the SAMGs.

The SAMGs are not put into use until an accident has progressed beyond the design basis of the plant. Because the SAMGs do not direct any plant alterations until after the normal accident mitigation procedures (EOPs) are exhausted, the probability of the occurrence of a previously evaluated accident is not increased.

Because the SAMGs are actually designed to minimize the consequences of an accident that has progressed beyond the design basis after the mitigation efforts directed by the EOPs have been exhausted, the consequences of a previously evaluated accident will not be increased.

The change addressed by this review simply establishes the administrative aspects of the SAMG program. The equipment configuration, functions or methods of performing those functions as described in the UFSAR are not affected.

The consequences of previously evaluated equipment failures are not affected by the administrative aspects of the SAMG program because equipment operation or configuration is not addressed in these documents.

The purpose of the SAMG program is minimization of the public dose consequences from accidents that have progressed beyond the plant's design basis. Because the SAMG process does not change any normal, off normal or design basis event mitigation equipment configuration or fundamental interactions, the use of the process cannot lead to a previously unevaluated accident. If a previously unevaluated accident should occur, the SAMGs should provide some guidance in dealing with the situation and thereby providing the plant staff with a tool to perform their primary function of protecting the public.

The administrative aspects of the SAMG program does not deal with equipment operation, configuration or functionality issues. Because the SAMG program does not result in any equipment

design function changes, the possibility of an unevaluated equipment failure is not increased. The SAMG's do provide equipment lineups and operational suggestions but only after the design basis accident mitigation procedure set is determined to be ineffective. The SAMGs are considered beyond design basis documents and, as such, will be maintained as guidelines and not subject to 50.59 review.

The implementation of the SAMGs is an industry commitment to the NRC and beyond the scope of Tech Specs. As SAMGs deal with beyond design basis events, Tech Spec bases is not affected.

SEV-1119
EVALUATION OF ADJUSTABLE TRAVEL STOP SET POSITION FOR HCV-624 AND HCV-625

An adjustable valve travel stop will be added to the actuators for RHR discharge control valves HCV-624 and HCV-625. The travel stop consists of a top mounted handwheel, mounted onto the existing actuator top cover. The handwheel has the capability to either manually close the valve or be used as a limit to upward travel of the actuator, thereby limiting the open position of the valve. The handwheel will be installed under PCR 98-068, and the desired position of the valve set during a flow test planned as part of PT-2.10.10, during the initial stage of the refueling cavity filling during the 1999 outage. Following the setting of each valve actuator in position, the handwheel will be chain locked in place. The modification will not prevent the valves from being throttled in the closed direction. The valves' open position will be limited to a position less than full open as determined by analysis and be set during the flow test. No further adjustment of the valve is needed for any mode of operation.

The probability of occurrence of an accident previously evaluated in the SAR is not increased, because the affected valves, HCV-624/625, do not change position following a postulated transient. They remain in their open position. The extent of their open position is being changed, and the new position will still ensure the required low head safety injection flow for the duration of the transient.

The consequences of an accident previously evaluated in the SAR are not increased, because the required flow rate listed in the COLR will continue to be maintained during the injection phase. Providing a limit on system flow will also ensure, under conditions resulting in maximum injection flow, that RHR pump runout conditions do not exist. In the longer term, following switchover to the sump recirculation phase, the modification provides a limitation on RHR flow rate, while assuming a loss of instrument air, thereby preserving NPSH margin. Therefore, core cooling can continue with no loss of function.

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR is not increased, because the valve travel stop is a physical stop against which the actuator stem rests. The valve actuator is not called upon to move following a postulated accident so there is no increase in probability of a malfunction. Should a loss of instrument air occur, the travel stop will prevent movement of the valve, since the stem of handwheel assembly rests against the diaphragm preventing further opening.

During non-accident modes of the RHR system when throttling is necessary using HCV-624/625, the travel stop will not interfere with the throttling of the valves in the closed direction. There is currently no need to throttle the valves more open than the travel stop position will be set.

Administrative limits currently exist on RHR flow (1500 gpm) which limit the flow rate to a value less than the travel stop would allow.

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR is not increased, because the valves' function will remain fail open on loss of instrument air. Since the valves are normally maintained open while the plant is at power, there are no times when the fail open on loss of instrument air function would be called upon. There are no malfunctions that would cause the valves to fail closed since the spring in the actuator is a passive device not dependent on external controls, and the valves are routinely tested and calibrated. The travel stop cannot cause the valves to move in the closed direction, since its design only restricts motion in the upward direction. Therefore, LHSI flow rate will still meet the COLR values and no increase in consequences can occur due to reduced core cooling assumed in the accident analysis.

The possibility of an accident of a different type than any evaluated previously in the SAR is not created, because the travel stop does not interfere with the operation of HCV-624/625 over the range of travel these valves are assumed to maintain.

The possibility of a malfunction of equipment important to safety of a different type than evaluated previously in the SAR is not created, because the travel stop is designed to provide sufficient flow to preserve LHSI capability under the limiting assumptions previously assumed, while limiting flow sufficiently to preserve RHR pump NPSH margin during the sump recirculation phase. The valve actuator handwheel will be locked in place so that manually opening the valve more than its setpoint cannot be inadvertently performed. Operation of the valve from the control room and operators use of the valves will be unaffected.

The margin of safety as defined in the basis for any technical specification is not reduced, because no changes are being made to the functions of the valves, and the LHSI system capability will continue to be maintained in excess of COLR flow requirements in the limiting case.

SEV-1120

REMOVAL OF DEWPOINT MEASURING INSTRUMENTATION FROM THE SEISMIC AND METEOROLOGICAL INSTRUMENTATION SYSTEM

The Ginna Station Seismic and Meteorological Instrumentation System (SMI) is made up of a variety of components. Included is a dewpoint measurement system. The Instrumentation & Control Special Projects group has requested to remove the dewpoint measuring system because of the maintenance requirements of the system and lack of requirements for its use. The environment in which the dewpoint transmitter is required to operate (increased frequency of airborne dirt particles due to a fairly constant breeze) is not conducive to the sensitivity of the dewpoint transmitter. The dewpoint transmitter senses humidity via a lens which is frequently fouled with dirt and grime resulting in recurring problems and inaccurate data.

The dewpoint monitoring system does not interact with any equipment used to mitigate accidents or transients. In addition, the data gathered by the dewpoint monitoring system is not used in the decision process for mitigation of accidents or transients.

The dewpoint monitoring system is functionally unrelated and physically independent of any System, Structure or Component important to safety. The independence of the dewpoint measuring system from any System, Structure or Component important to safety ensures that the proposed modification can not introduce a failure mechanism which would increase the probability of occurrence of an accident previously evaluated in chapter 15 of the UFSAR. The dewpoint monitoring system is not required per Reg. Guide 1.97. This modification will not affect the meteorological monitoring system design limits nor reduce system reliability.

The dewpoint monitoring system is functionally unrelated and physically independent of all equipment used for the mitigation of accidents and transients. The independence of the dewpoint measuring system from any System, Structure or Component important to the mitigation of accidents and transients ensures that the proposed modification can not introduce a failure mechanism which would increase the consequences of an accident previously evaluated in chapter 15 of the UFSAR. The modification does not impact or increase the calculated radiological dose to the general public for any event evaluated in the UFSAR. The dewpoint monitoring system is not presently required per Reg. Guide 1.97 and is not used as an input to other dose calculations.

The dewpoint monitoring system is functionally unrelated and physically independent of any System, Structure or Component important to safety. The independence of the dewpoint measuring system from any System, Structure or Component important to safety ensures that the proposed modification can not introduce a failure mechanism which would increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in chapter 15 of the UFSAR. The modification will not degrade the performance of the meteorological monitoring system. The dewpoint monitoring system is not interconnected to any System, Structure

or Component important to safety.

The dewpoint monitoring system is functionally unrelated and physically independent of all equipment used for the mitigation of accidents and transients. The independence of the dewpoint measuring system from any System, Structure or Component important to the mitigation of accidents and transients ensures the proposed modification will not introduce a failure mechanism which would increase the consequences of a malfunction of equipment important to safety previously evaluated in chapter 15 of the UFSAR. The modification does not impact or increase the calculated radiological dose to the general public for any event evaluated in the UFSAR.

The dewpoint monitoring system is functionally unrelated and physically independent of all equipment used for the mitigation of accidents and transients. The independence of the dewpoint measuring system from any System, Structure or Component important to the mitigation of accidents and transients ensures that the proposed modification will not introduce a failure mechanism which would increase the probability of an accident of a different type than any previously evaluated in chapter 15 of the UFSAR. There are no adverse affects upon other systems, nor any new failure modes induced.

The dewpoint monitoring system is functionally unrelated and physically independent of all equipment used for the mitigation of accidents and transients. The independence of the dewpoint measuring system from any System, Structure or Component important to the mitigation of accidents and transients ensures the proposed modification will not introduce a failure mechanism which would increase the consequences of a malfunction of equipment important to safety of a different type than previously evaluated in the UFSAR. The power source for the Ginna Station Seismic and Meteorological Instrumentation System is from both off-site power and in-plant non-1E sources. The physical location is such that damage to the structure(s) itself will not affect equipment important to safety. The modification does not degrade the meteorological monitoring system.

The dewpoint monitoring system is functionally unrelated and physically independent of all equipment used for the mitigation of accidents and transients. The independence of the dewpoint measuring system from any System, Structure or Component important to the mitigation of accidents and transients ensures the proposed modification will not introduce a failure mechanism which would reduce any margin of safety as defined in the basis of any Technical Specifications. The required functions and characteristics of the Ginna Station Seismic and Meteorological Instrumentation System remain unchanged.

SEV-1121
PCN # 98-4517 SAFETY EVALUATION, CHANGES TO ATT-2.1, ATTACHMENT MIN SW, TO ADDRESS ACTION REPORT 98-1042 CONCERNS

ACTION Report 98-1042 identified a concern with guidance provided in Revision 4 of procedure ATT-2.1, "ATTACHMENT MIN SW". This attachment is used to align the service water system for the recirculation phase of a LOCA with one operable SW pump. ATT-2.1 instructs the operators to fully open the service water globe valve on the discharge side of the CCW heat exchanger to be aligned (V-4619 or V-4620).

During a reconstitution of the service water system hydraulic model, an error was found in the hydraulic loss coefficient used to represent each CCW heat exchanger. The coefficient used in the calculation was significantly higher than the value that would be back-calculated from either vendor supplied pressure drop data or actual test data.

Since the actual hydraulic resistance is lower than originally modeled, the service water flow rate to the applicable CCW heat exchanger would be considerably higher than originally predicted if V-4619 or V-4620 were opened completely with a single service pump in service. As a result, the flow rate to the CRFCs and EDG coolers could be significantly lower than predicted in previous hydraulic models, and the service water pump margin to runout would be reduced.

The following changes to Revision 4 of ATT-2.1 are proposed:

> Delete step 3 which has operations request that the TSC evaluate isolation of SW loads in containment. The step will be replaced with explicit instructions to isolate inoperable containment loads (CRFCs and Reactor Compartment Coolers) by closing the service water isolation valve on the discharge of each line. This step will be preceded by a note stating that these isolations are to be performed as soon as possible after sump recirculation has been established.
>
> Add a new step containing the guidance previously in step 3 regarding TSC evaluation of closure of the Bus 17-18 cross-tie and startup of a second service water pump. This change is considered inconsequential and will not be addressed in this safety evaluation.
>
> Break out the step that isolates service water to the SFP heat exchangers and place it prior to the step that adjusts service water flow to the applicable CCW heat exchanger. There is no reason that this step has to be done after restoring service water to the Auxiliary Building, and moving this step minimizes complications during alignment of service water to the applicable CCW heat exchanger, such as the effect of service water flows from SFP HXB on the FIA-2005 reading, which is used to set V-4619 or V-4620 position. Since the current attachment revision already isolates SFP cooling, and the attachment must be

completed in entirety prior to going into recirculation, relocation of this step has no implications on the timing of the transfer to recirculation. Therefore, relocation of this step is considered inconsequential and will not be addressed in this safety evaluation.

Modify step 5 to throttle the SW outlet valve on the operating Auxiliary Building service water loop to between 2750 gpm and 3250 gpm.

Add a note to inform operators that EDG cooling may be affected while adjusting the service water flow to the CCW heat exchanger and to reduce load or refer to ER-D/G.2, ALTERNATE COOLING FOR EMERGENCY D/Gs, should an EDG temperature alarm occur. This change is considered inconsequential and will not be addressed in this safety evaluation.

Add new step to notify TSC of all loads that were isolated. This change is considered inconsequential and will not be addressed in this safety evaluation.

Implementation of these steps will address the issue raised in ACTION Report 98-1042. Additionally, explicit isolation of inoperable containment building loads will increase the heat removal rates from containment, increase the margin to vapor locking in the CRFCs due to flashing in the downstream service water piping, and provide increased service water flow rates to the EDG coolers.

The proposed procedure changes apply during the recirculation phase of a LOCA only. They will not increase the probability of occurrence of an accident previously evaluated in the SAR since the accident will have already occurred prior to usage of the procedure.

The proposed procedure changes will not increase the consequences of an accident previously evaluated in the SAR. The analyses outlined in the functional impact section of this safety evaluation provide adequate justification that the changes have only positive effects with respect to the capability to deal with and the consequences of a LOCA, which is the only impacted accident.

The proposed changes are beneficial with respect to equipment reliability during the recirculation phase of a LOCA and therefore it is reasonable to conclude that the probability of occurrence of a malfunction of equipment important to safety is reduced. Specifically:

The margin to overheating of the Emergency Diesel Generators is increased due to the increase in service water flow to the EDG coolers.

The margin to vapor locking in the CRFCs is increased due to the increase in service water flow through the CRFC coolers. This is also beneficial with respect to cooling the

atmosphere in containment.

The margin to runout of the single operating service water pump is increased by the increase in system back pressure.

The EQ temperature and pressure profiles which were used to qualify equipment in containment for post-accident conditions are met. Isolation of inactive containment loads increases overall containment heat removal so is marginally beneficial in terms of equipment reliability.

The consequences of a malfunction of equipment important to safety are not increased by the proposed changes. This statement is justified in the sub-sections below:

Note that the licensing basis single-active failure will already have occurred prior to usage of ATT-2.1, since this is required to get to one service water pump operation; therefore, any additional failure will be beyond the design basis of the plant.

The consequences of any failure that results in the loss of the sump heat sink will be reduced. Loss of this heat sink will result in an increase in containment temperature; the increased flow to the CRFCs provided by the proposed changes beneficially increases the heat removal from the containment atmosphere and increases the margin to vapor locking in the CRFCs.

The consequences of the loss of a CRFC are reduced since the remaining CRFC(s) will have higher flow rates and therefore greater heat removal.

The consequences of the loss of an EDG, assuming only one was originally operating, are no more severe than they would be if the proposed changes were not implemented since the end result in either case is a complete loss of active heat sinks.

The consequences of the loss of the operating service water pump are no more severe than they would be if the proposed changes were not implemented since the end result in either case is a complete loss of active heat sinks.

The possibility of an accident of a different type than any evaluated previously in the SAR is not created. The proposed changes are intended to help mitigate the consequences of an accident that has already occurred, and a second accident is not assumed to occur coincidentally during recovery from the first.

The proposed changes do not change the configuration of the plant prior to the occurrence of a design-basis LOCA and therefore will not create the possibility of a different type of malfunction.

As discussed previously, the proposed changes ultimately have beneficial impacts on equipment reliability during the recirculation phase of the LOCA.

The only margins that could potentially be challenged by the changes are the maximum containment pressure and the EQ profiles. The proposed changes will have no impact on the peak pressure since this occurs prior to the transfer to recirculation. Further, it has been shown that the proposed changes do not challenge the profiles assumed for equipment qualification. Since no margins of safety have been challenged, the margin of safety as defined in the basis for any technical specification will not be challenged.

SEV-1123
SPENT FUEL PIT LEAKAGE RELEASE PATH ASSESSMENT

There have been numerous USNRC Inspection Reports dealing with the presence of water leakage into various plant structures. Analyses of some of the leakage has indicated the presence of boric acid and radionuclides that are also present in the spent fuel pool (SPF) and transfer canal. With this finding, the NRC has expressed a concern on the potential for a radionuclide release off-site.

USNRC Inspection Report 95-015-01 initiated the concern of a radiological release of the Spent Fuel Pit (SPF) water into the environment. Since that inspection, several measures have been initiated to (1) assess the leakage source, (2) determine the most probable groundwater flow direction, and (3) initiate a monitoring program for tracking any potential off site releases.

Based on sampling and testing, it has been determined that some leakage is occurring from the transfer canal.

This evaluation is to assess the potential for such a release and demonstrate that leakage from the transfer canal will be controlled and processed as required to conform to the appropriate NRC and EPA regulations.

A sudden increase in SFP Liner leakage would be the accident/event of concern which, is not presently addressed in the UFSAR accident analysis. Therefore, the probability of occurrence of an accident previously evaluated in the UFSAR is not increased.

As stated above, the accident in question is not evaluated in the UFSAR. Therefore, the consequences of an accident previously evaluated in the SA is not increased.

The equipment in question would be the RHR and RCDT Pumps in the Auxiliary Building sub Basement. The suspected leakage of SFP water into the RHR room is believed to be originating through incomplete or defective seal welds of the liner to the embedment of the refueling canal. Should there be a complete failure of these welds, an unrestricted flow of the canal inventory into the RHR room is precluded by the concrete/bedrock interface. In addition, the increased frequency of Auxiliary Building sump Pump actuations would alert the Operators, providing an opportunity to take corrective actions. Therefore, there is no increase in the probability of occurrence of a malfunction of equipment previously evaluated in the SA.

The consequences of the event described above, (failure of the RHR/RCDT Pumps) would remain the same regardless of the failure mechanism and therefore, would not be increased.

The flow path described above, does not lend itself to a rapid outflow of water from the SFP. The leakage of borated water into the RHR room has been determined to be from the refueling canal and has been quantified to be very small (~0.001 gal/min). The leak path is through the interface of the refueling canal concrete foundation and bedrock. Both the concrete foundation and bedrock are impervious to water and, as such, erosion/failure of either, which could establish a potential flood path is not possible. This restriction of flow would allow ample time for mitigating actions such as installing the weir gate, closing of the transfer tube gate valve and/or draining the transfer canal. Even in the unlikely event of a rapid outflow of water from a failure in this area, the height of the weir gate path would preclude the uncovering of spent fuel in the pit. Therefore, the possibility of an accident of a different type than that evaluated in the UFSAR is not created.

Based on the above discussions of leak rates the operability of the RHR/RCDT pumps is not jeopardized by this condition. Therefore, the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the SA is not created by this condition.

The issue of SFP leakage is not addressed in any technical specification, therefore, there is no affect on margins of safety as defined in the bases of the Technical Specifications.

SEV-1124
VALVE STEM PACKING IMPROVEMENT PROGRAM SAFETY EVALUATION
CHANGES TO DESIGN CRITERIA - EWR 4859

This Safety Evaluation was prepared to replace Revision 1 to the Safety Analysis (Revision 1 was never approved) for EWR 4859 to evaluate the addition of Expandable Valve Stem Packing (EVSP) of the "cup and cone" design as a packing system alternative.

This analysis covers the live-loading of gland followers and/or replacement of valve stem packing of certain selected valves.

Valve stem packing leakage is a widespread problem that impacts overall nuclear power plant operation and maintenance. In some cases, even minor stem packing leakage has far reaching implications in terms of radiation exposure, load reduction and housekeeping problems. In 1984, Electric Power Research Institute (EPRI) established a program to study the root causes of valve stem packing leakage and to identify, develop and evaluate means of corrective action. As a result, two improvements were identified by recent EPRI studies. These improvements, when retrofitted, have the potential to greatly alleviate the maintenance burden associated with the valve steam packing leakage. These improvements are:

- Replacement of traditional woven asbestos packing with die-formed square flexible graphite packing or Expandable Valve Stem Packing (EVSP) of the "cup and cone" design.

- Live-loading of gland followers to compensate for stress relaxation, aging, consolidation or thermal cycling of the packing material.

As part of the preventive maintenance program, Ginna Station Maintenance Department has decided to replace the asbestos stem packing with die-formed square graphite stem packing or EVSP for several existing and new valves. Some of the valves shall also be retrofitted with live-loading.

The proposed modification would not increase the probability of occurrence of an accident previously evaluated in the UFSAR since this change only allows replacing approved packing materials and methods with improved alternatives that will reduce the potential for packing leakage.

The proposed modification would not increase the consequences of an accident previously evaluated in the UFSAR since the expandable valve packing is an improvement in valve packing systems with less potential and, subsequently, less consequences for leakage.

The proposed modification would not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR because this program provides the criteria for the replacement and upgrade of packing materials and methodology in safety-related valves resulting in an increase in reliability for affected valve operation.

The proposed modification would not increase the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR because the failure of packing (existing or replaced by this program) would not violate the equipment's pressure boundary function.

The proposed modification would not create the possibility of an accident of a different type than any previously evaluated in the UFSAR because the potential for failure related to packing materials and methods currently exist in the licensing basis and will remain with, although mitigated by, new improved packing systems.

The proposed modification would not create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the UFSAR because the change in packing material and methodology incorporates improved technology which will result in a greater valve packing system reliability.

The proposed modification would not reduce any margin of safety as defined in the basis of any technical specification because equipment reliability will be increased upon modification by the Valve Stem Packing Improvement Program.

SEV-1125
STATIONARY BATTERY REPLACEMENT

Battery A (BTRYA), Battery B (BTRYB) and the spare battery cells (BTRYSP) are being replaced during the 1999 refueling outage due to aging concerns initially identified in ACTION report 97-1110. The batteries have not degraded to the point where discharge testing indicates replacement is required, however the physical signs of aging, plus the need to replace the cells prior to 2009 have been factored in the decision to replace both batteries at this time.

Electrical Engineering Specification EE-168 was prepared to outline the design and performance requirements for the new battery cells. Nuclear Logistics Incorporated will be providing new batteries manufactured by GNB Technologies meeting the design and performance requirements of EE-168. RG&E requested quotes for 1200 amp-hour and 1495 amp-hour battery capacity in order to determine the marginal cost of increasing the margin between battery capacity and design basis load. The 1495 amp-hour battery was chosen as the replacement.

The existing batteries are GNB model NAX-1200 and NAX-17 (1200 amp-hour). The cells dimensions are: length 7.38 inches, width 14.5 inches and height 22.13 inches. Weight is 245 pounds.

The new batteries will be GNB model NCN-21 (1495 amp-hour). These cells are larger than the existing cells. Length 9.25 inches, width 14.5 inches and height 22.5 inches. Weight is 301 pounds which is 56 pounds heavier than the existing cells.

The added size and weight of the new battery cells would require modification of the existing racks. An initial evaluation determined no cost benefit between modification or replacement therefore new structural racks will be installed, designed to meet the seismic forces of the Battery Rooms. The spare battery cell racks will be modified as necessary to accommodate the larger cells. The Battery Rooms are located in the basement of the Control Building.

The probability of occurrence of an accident previously evaluated in the SAR is not increased by this modification. The station batteries are used to mitigate the consequences of accidents. They have no failure modes or effects which directly lead to the occurrence of any accident previously evaluated in the SAR. After the proposed change is complete the batteries will continue to have the independence and separation which they are required to have, therefore there are no new functional interactions which affect the previously evaluated accidents.

The new batteries will be seismically mounted on new racks and will be operated in the exact configuration as the existing system. Several changes to float voltage and equalize time will be placed into effect through procedural control, however these changes will not result in the occurrence of an accident. Other than the batteries, intercell connectors and racks, no new

equipment is being added. No existing equipment needs to be functionally modified as a result of this proposed change.

The consequences of an accident previously evaluated in the SAR is not increased by this modification. The new batteries have a greater capacity than the existing batteries therefore they have the ability to mitigate any design basis events which the old batteries have been qualified to mitigate. This includes the 4 hour station blackout coping period. The increase in amp-hour capacity is based on a difference in battery design, however these design differences will not increase the consequences of any accident previously evaluated.

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR is not increased by this modification. The new batteries are being purchased as Class-1E equipment from a qualified supplier to the design conditions of the Battery Rooms. The new battery racks are being purchased from the same supplier with seismic qualifications to the requirements of Ginna's Battery Rooms. The batteries will be bounded by the same maximum voltage (140 VDC) as the existing batteries, therefore the operability of all equipment connected to the DC distribution systems will be maintained with the new batteries installed. The float voltage will be set at a new higher value in order to minimize the amount of equalize charges which have to be performed on the batteries. There is no increase in the probability of a malfunction of any equipment important to safety connected to the DC distribution systems due to this modification.

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR is not increased by this modification. There is no increase in the consequences of a malfunction of any equipment due to this modification. If one of the batteries were to fail there would be no increase in the existing consequences resulting from the loss of a battery.

The consequences of a malfunction of a piece of equipment other than the batteries or racks will not be increased by this modification. Electrical evaluation including coordination and short circuit protection demonstrate that the protection of the electrical system will not be degraded due to this modification. An evaluation of the hydrogen generation capability of the new batteries and a change in the Battery Room combustible load demonstrate that installation of the new batteries will not exceed the ability of the HVAC system to remove hydrogen from the Battery Rooms nor will the amount of combustible load increase beyond the maximum allowable combustible load for the fire zones in which the batteries are contained.

The possibility of an accident of a different type than any evaluated previously in the SAR is not created by this modification. No new equipment is being added to the DC distribution system due to this modification, therefore there is no potential of an accident of a different type than any previously evaluated. The batteries are being replaced with new cells with increased capacity and new seismic racks are being installed. The new batteries are functionally equivalent to the existing batteries and no new failure modes will be introduced due to this modification.

The possibility of a malfunction of equipment important to safety of a different type then evaluated previously in the SAR is not created by this modification. This modification will install new Class-1E battery cells and new seismic battery racks to replace existing equipment. The new batteries will be lead calcium, which is different than the existing batteries which are lead antimony. Antimony and calcium are metals added to the grid design to increase strength. There is no possibility of a failure of a different type due to differences in battery design than previously evaluated for the batteries or racks.

Evaluations of the electrical system, HVAC system and fire loading demonstrate that the new batteries will be capable of operation without impacting the operability of any systems supporting the Battery Rooms or the DC distribution system.

The margin of safety as defined in the basis for any technical specification is not reduced by this modification. This modification will install new seismic battery racks designed for the seismic conditions of the Battery Rooms and the loads of the new battery cells. The new batteries have a greater capacity than the existing batteries being replaced therefore the margin of safety is not being reduced by this modification. The impact of having a larger battery connected to the DC distribution system has been evaluated.

SEV-1127
DIESEL GENERATOR SUPPLY BREAKER TIME DELAY RELAYS

During safety injection, if a single safeguards bus has an undervoltage actuation, either degraded or loss of voltage, its supply breaker will trip and the sister bus supply breaker on that train will also trip. Each bus will start a 1.3 second timer that upon timing out closes the bus diesel generator (DG) supply breaker. The sister bus UV system will not actuate since the minimum time required is the loss of voltage relay definite time delay of 2.4 seconds.

If the sister bus is 14 or 16, all loads that were sequenced on prior to diesel generator closure would be block loaded. SI sequencer would not be reset.

If the sister bus is 17 or 18, service water would be loaded out of sequence. It is also possible to load two service water motors onto the diesel generator. This would exceed design loading during SI.

This modification will install time delay relays in the control circuits of the diesel generator supply breakers to the safety related 480 VAC busses 14, 16, 17 and 18.

The two time delay relays, set for 0.5 and 3.5 second delay pickups, in each safeguards diesel generator supply breaker control circuit shall actuate the respective bus UV system upon coincident opening of both the normal and diesel generator supply breakers. Logic shall allow for live bus transfers and bus restoration using bus tie breakers.

The new configuration will operate as follows:

Upon coincident open normal and diesel generator supply breakers to a bus, the new relays begin timing. After 0.5 seconds the first relay's NO contacts close to actuate the bus UV system. After 3.5 seconds, the second relay times out and the 0.5 second delay relay is de-energized. The DG supply breaker closes when the DG frequency and voltage are acceptable and the UV system resets. If the existing 1.3 second delay relay times out and the DG supply breaker closes, the 0.5 second relay is de-energized and the UV system resets.

The new configuration performs the following:

- Activates a bus UV system before UV relays actuate when a bus is de-energized by a normal supply breaker trip.

- The 3.5 second delay pickup bridges the gap between the supply breaker opening and the UV loss of voltage relay time out, 2.75 second Technical Specification limit.

- Allows operators to restore bus voltage through use of bus-tie breakers after a 3.5 second delay.

- Allows bus transfer from DG supply breaker to normal supply breaker as currently performed in emergency procedures. (The 0.5 second delay allows the DG breaker to open and the normal supply breaker to close without a UV system actuation.)

The probability of occurrence of an accident previously evaluated in the SAR will be changed due to the implementation of this modification. The diesel generators are used to supply power to the 480 VAC busses to mitigate the consequences of accidents. After the proposed change is complete the diesel generator supply breakers will continue to have the independence and separation which they are required to have to perform their safety related functions.

A failure of the 0.5 second relay contacts to open introduces a new failure mechanism in the safeguards undervoltage systems. A relay failure increases the frequency per reactor year of a safeguards bus failure by 1.91E-6. A single relay failure renders a bus inoperable by maintaining the bus undervoltage system in the trip mode. The bus would be energized but the undervoltage actuation would prevent bus loading.

The failure of a relay's contacts to open is a single failure and does not affect the redundant train. The potential failure of a safeguards 480 volt train due to the existing configuration is not a single failure. Combined with a single failure of the redundant train's diesel generator the existing configuration could result in a station blackout during SI.

The proposed control configuration significantly decreases the frequency of a loss of a 480 volt bus safeguards train during an SI. However, it increases the frequency per reactor year of an inoperable safeguards bus. The net change in frequency of a safeguards bus loss is a decrease of two orders of magnitude, 1E-4 decrease versus 1.91E-6 increase. Therefore the proposed configuration increases overall plant safety.

The consequences of an accident previously evaluated in the SAR is not increased by this modification. This modification will install new time delay relays in the control circuits of the diesel generator supply breakers. Any failure of the breakers to actuate due to the new relays will be bounded by previously assumed failures of the undervoltage system to actuate and failure of breakers to open/close. No new equipment is being installed and no plant configuration changes are being performed which will increase the consequences of any previously evaluated accidents.

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR is not increased by this modification. As discussed above, there is a net overall decrease in the loss of a safeguards bus due to this modification. A new failure mechanism

will be introduced to the undervoltage system due to this modification, however the increased probability of the loss of a safeguards bus due to the installation of the new time delay relays will be offset by a decreased probability of the loss of a 480 volt bus safeguards train during an SI.

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR is not increased by this modification. There is no increase in the consequences of a malfunction of any equipment due to this modification. If a bus supply breaker failed to close or an undervoltage system failed to actuate there would be no increase in the existing consequences.

The possibility of an accident of a different type than any evaluated previously in the SAR is not created by this modification. No new equipment is being added to the diesel generator bus supply breaker control circuits which can result in an accident of a different type than any previously evaluated. The new relays are being installed within the control circuits of the diesel generator bus supply breakers, in parallel with existing time delay relays. The function performed by the breakers is not being changed.

The possibility of a malfunction of equipment important to safety of a different type then evaluated previously in the SAR is not created by this modification. This modification will install new time delay relays in the diesel generator bus supply breakers. The relays are safety related and are seismically qualified for use at Ginna Station. A failure of any new equipment will not create the possibility of a malfunction of equipment important to safety of a different type due than previously evaluated for the breakers or undervoltage system.

The margin of safety as defined in the basis for any technical specification is not reduced by this modification. This modification will install new time delay relays in the control circuits of the diesel generator supply breakers. There is no impact on Technical Specifications due to this modification therefore there is no change in the margin of safety.

SEV-1128
SERVICE AIR SYSTEM UPGRADE PHASE B

The scope of this modification is to replace the existing Service Air Compressor with a more efficient and reliable oil free, air cooled, two stage rotary screw compressor and heatless regenerative desiccant dryer. This modification also includes the addition of a cross-connect between the instrument air system and the service air system downstream of the new dryer. A check valve and an automatic isolation valve on the service air side of the cross-connect will function to prevent service air backflow into the instrument air system and prevent a loss of service air pressure from degrading the instrument air system. Although no system or component name changes are involved, this proposed change realistically reconfigures the instrument air system into a four compressor system which takes advantage of its increased compressor capacity to supply air to the service air system.

This modification will improve the reliability of both the service air and instrument air systems and thereby aid in meeting the NRC requirements of the Maintenance Rule. The change installs a very dependable backup to the instrument air system as well providing a low maintenance source of air capacity for use as service air.

This modification also includes reconfiguring door F28 such that it can be left open during periods of high ambient temperature in the turbine building. The door closer will be redesigned so that a fusible link will allow automatic closure in the event of a fire in the turbine building or all volatile treatment room.

The instrument and service air systems have no failure modes or effects which are precursors to accidents evaluated in the SAR. The proposed change does not introduce any new failure modes or effects to the air systems or any other system or component which is a precursor to an accident. Because the proposed change has no interaction with any system which, if failed, leads to an accident the proposed change can not increase the probability of an accident previously evaluated in the SAR.

Pneumatically operated components required for accident mitigation have backup systems which provide valve operator mode of force in the event the station air systems are lost or degraded. The proposed change has no functional interaction with the backup systems thus the change can not cause any equipment failures which would reduce the availability of equipment relied upon for accident mitigation. Because the change does not affect the equipment set used to mitigate accidents the change can not increase the consequences of an accident.

As described the proposed change provides isolation between the service and instrument air systems. After the change the instrument air system will have better capacity and equipment reliability then before the change. It can be concluded that the proposed change reduces the

probability of occurrence of a malfunction in a risk significant system (instrument air).

The accident analysis already assumes the unavailability of the station air systems. The proposed change does not introduce any new failure modes or effects into the station air systems. This proposed change can not alter the consequences of a loss of instrument air.

The accident analysis already assumes the unavailability of the station air systems. The proposed change does not introduce any new failure modes or effects into the station air systems. Because the change does not affect any system that can act as an accident precursor it can not create the possibility of an accident of a different type the previously reviewed.

Pneumatically operated components required for accident mitigation have backup systems which provide valve operator mode of force in the event the station air systems are lost or degraded. The proposed change has no functional interaction with the backup systems thus the change can not cause any equipment malfunctions which would reduce the availability of equipment relied upon for accident mitigation. Because the proposed change only interacts with non-safety equipment it can be concluded that the change does not increase the possibility of a malfunction of equipment important to safety.

The instrument and service air systems are not credited as inputs to the accident analysis nor are they factors in the basis for any margin of safety addressed in the technical specifications.

SEV-1129
CONTROL ROOM HVAC UPGRADE PHASE 1

The purpose of this evaluation is to determine if an unreviewed safety question exist with the planned phase 1 modification to upgrade the reliability of the Control Room HVAC system by providing a redundant filtration train for the Control Room Emergency Air Treatment Sub-system (CREATS). Phase 1 of the modification consist of :

1) Adding tie in connections to the existing CRHVAC duct work with blind flanges to allow future connection of a second charcoal filter train.

2) Replacing MCC K spare breaker 1KK with an upgraded breaker

3) Swapping MCC K DC control power from train B to train A, removing 4KV test cabinet load from Train A Main DC distribution panel and placing it on the Turbine Building DC distribution panel.

4) Adding a backdraft damper to the outlet of the Control Room Air Handling unit Supply Fan(AKD27).

5) Replace degraded CRHVAC flex duct connectors

6) Connect a spare cable to one of the spare contacts on SI relays SI-16X and SI-26X for future use. One cable to each relay.

7) Disconnect or cap the supply and return ducts to the MUX room and remove doors to the Relay Room

8) Cutting the existing face plate in half and adding a support member on the Aux Bench Board in the Control Room to allow future on-line replacement for new train B controls.

The probability of occurrence of an accident previously evaluated in the SAR is not increased since the source terms during a LOCA will not be increased by this modification, it does not affect the RCS , Containment Filtration or the ECCS. The reliability of the AC and DC systems will not be decreased since the electrical changes meet the original plant design and construction standards. The SI system testing will be done in a plant mode when the SI system is not required to be operable and procedural actions are in place to prevent an inadvertent SI actuation. The provisions provided for the protection against fires will not be impacted.

The consequences of an accident previously evaluated in the SAR is not increased by this

proposed change since the performance of the CREATS system, AC and DC systems will not be degraded from what is assumed in the accident analysis. The SI system functional will not be affected by testing existing spare wiring and contacts.

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR is not increased since the new backdraft damper is similar in construction to existing active dampers in the system and therefore will have the same type of failure mechanisms and probabilities. The new breaker, cables and flexible duct connectors are the same or equal to the original equipment the are replacing and therefore their probability of occurrence or malfunction will not be increased over what was originally assumed. No new equipment is added to the SI system.

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR is not increased. A failure of the CREATS system filtering ability was already assumed in the original accident analysis therefore the addition of the backdraft damper will not affect any consequences. The performance of the CREATs system will not be degraded and therefore, the consequences of an radioactive release will not be increased. The additional DC load on the A Batteries will not reduce the systems ability to cope with a loss of all AC for four hours, therefore the consequences of a SBO will not be increased: The SI system testing will be done in a plant mode when the SI system is not required to be operable and procedural actions are in place to prevent an inadvertent SI actuation.

The possibility of an accident of a different type than any evaluated previously in the SAR is not created. The proposed change creates no new functional interactions with existing plant equipment nor does it introduce new failure modes or mechanisms which could lead to reactor core damage or fission product release. No new equipment is added to the SI system. The SI contact testing will not change any system configuration and the SI system testing will be done in a plant mode when the SI system is not required to be operable with procedural actions in place to prevent an inadvertent SI actuation. .

The possibility of a malfunction of equipment important to safety of a different type then evaluated previously in the SAR is not created since the cable, damper, breaker and duct material are all currently used at the station and therefore will not create a different type of malfunction. Train A DC power is already used will all other train A MCC loads and there are no train B loads on MCC K. The loss of MCC K has already been considered and therefore, the use of train B control power is already bounded by the existing anaysis. The new spare SI cable will meet exist seperation criteria and fire protection requirements and is therefore bounded by existing analysis.

The margin of safety as defined in the basis for any technical specification is not reduced by this proposed modification since the capability or the requirement for the CREATS system to detect

radiation, isolate the control room, filter 2000 CFM of control room air will not be affected. The A train battery capacity will not be reduced by this modification. The SI contact testing will not change any system configuration.

SEV-1130
NEW PROCEDURE AP-CVCS.3, LOSS OF ALL CHARGING FLOW

New procedure AP-CVCS.3, Loss of All Charging Flow, has been developed to deal with the unique problems associated with that event. A loss of charging takes away the ability for RCS makeup at normal system pressure. Even if CVCS Letdown is isolated, the RCS continues to lose inventory through the RCP seals. The RCP seals are protected from high temperature conditions by CCW flow through the thermal barrier. However, if action is not taken, the RCS will continue to lose inventory until the pressurizer empties and pressure control is lost. A brief description of the procedure's high level actions are as follows:

a. Attempt recovery of Charging Pumps. Exit procedure if successful.

b. Reduce load at 5%/minute per AP-TURB.5, Transfer 4160 Volt loads, then trip the turbine at 15 Mw.

c. Shutdown the reactor.

d. Cooldown the RCS at <100F/hour to 530°F (provided two RCPs are operating), to allow for RCS depressurization. This is a shutdown margin issue for single loop operation.

e. Depressurize RCS to <1950 psig and block Safety Injection.

f. Restore pressurizer level by starting a Safety Injection Pump and further depressurizing the RCS to approximately 1400 psig.

g. Energize pressurizer heaters to maintain the pressurizer saturated.

h. Maintain RCS at stable temperature, pressure and inventory.

The probability of occurrence of an accident previously evaluated in the SAR is not increased because the new procedure is designed to shutdown the plant and re-establish a means of inventory control in a controlled manner. Continued operation of the CCW System ensures RCP seal integrity.

The consequences of an accident previously evaluated in the SAR are not increased because, without operator intervention, a loss of charging event would eventually terminate with auto safeguards actuation with the RCS in a more degraded condition. The new procedure actually reduces consequences of this transient.

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR is not increased because the new procedure is actually mitigating the consequences of just such an event (loss of charging). Other equipment is operated within previously established parameters, with the exception of the pressure/temperature limit curve for RCP NPSH. Although not recommended for routine operation, the limits established by this procedure for RCS pressure/temperature relationships are sufficient to support safe RCP operation.

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR is not increased because the new procedure addresses this type of event without impacting the failure consequences of other plant equipment.

The possibility of an accident of a different type than any evaluated previously in the SAR is not created because the intended function and operation of plant systems is not affected. In addition, the plant is not placed in a configuration which is not analyzed, and this evolution is intended to prevent a more significant transient from occurring.

The possibility of a malfunction of equipment important to safety of a different type than evaluated previously in the SAR is not created because the procedure addresses issues of equipment operational limits. For example, the operator is directed to observe starting duty of Safety Injection Pumps when cycling the pumps to stabilize RCS inventory.

The margin of safety as defined in the basis for any technical specification is not reduced because, although a loss of all charging may result in entry into certain LCOs, the procedure is designed to stabilize the plant and restore volume/pressure control, thereby maintaining the margin of safety.

SEV-1131
CYCLE 28 RELOAD

The reload for Cycle 28 consists of 44 new fuel assemblies labeled as feed regions 30A and 30B. This safety evaluation is valid for an End-of-Cycle 27 burnup of 15,509 to 16,517 MWD/MTU and for a Cycle 28 burnup that does not to exceed 18,160 MWD/MTU without additional analysis. The fuel assemblies to be loaded in the core for Cycle 28 are mechanically the same as the Cycle 27 fuel assemblies except for the following:

1. The fuel rod clad is fabricated with ZIRLO, an alloy similar to Zircaloy-4.

2. The thimble guide tubes are fabricated with ZIRLO.

3. The instrumentation tubes are fabricated with ZIRLO.

Compared with Zircaloy-4, ZIRLO has no Chromium (Cr) and reduced contents of Tin (Sn) and Iron (Fe). In addition, ZIRLO has incorporated a nominal amount of Niobium (Ni). The purpose of changing the chemical composition of Zircaloy-4 to that of the ZIRLO alloy is to improve corrosion resistance and dimensional stability under irradiation.

Westinghouse has designated the design characteristics of fuel assemblies with ZIRLO as VANTAGE +, and those of fuel assemblies with Zircaloy-4 as OFA.

The Cycle 28 reload will not increase the probability of occurrence of an accident previously evaluated in the SAR because the reload core does not affect accident initiators or equipment operation. The reload core does not cause a pipe to break or equipment to malfunction. Therefore, the reload core can not increase the probability of an accident previously evaluated in the SAR.

The change to ZIRLO as the material for fuel rod cladding, guide tubes, and instrumentation tubes, is not directly related to the probability of any accident previously evaluated in the UFSAR. The use of ZIRLO as a material does not impact the mechanical integrity of the fuel rod, or the structural integrity of the fuel assembly or the core under normal or accident conditions. All design criteria, applicable standards, and safety limits are met. Because of this, there are no new challenges to components and systems that would increase the probability of any previously-evaluated accident. Furthermore, the use of ZIRLO as fuel cladding improves corrosion performance and dimensional stability under normal conditions.

The fuel design changes satisfy existing design criteria; therefore, the probability of failure does not increase. Gap reopening does not affect accident initiators.

The Cycle 28 reload does not increase the consequences of an accident previously evaluated in the SAR because the core characteristics are bounded by parameters assumed in the accident analyses. When deviations occurred, reanalysis was performed to show that the acceptance criteria was still satisfied.

The mechanical changes to the fuel assemblies do not degrade performance. The ZIRLO material used in the fuel rod cladding, guide tubes, and instrumentation tubes has similar physical and mechanical properties to that of Zircaloy-4. All design criteria, applicable standards, and safety limits for the fuel rod cladding, guide tubes, and instrumentation tubes using ZIRLO material are met. Therefore, mechanical and structural integrity of the fuel assemblies will be maintained under normal and accident conditions.

An analysis of all fuel with gap reopening demonstrates compliance with the corrosion limit set forth in 10 CFR 50.46 for all of Cycle 28.

The radiological consequences of accidents with fuel assemblies using ZIRLO material are the same as those documented in the UFSAR for fuel assemblies using.

Based on the above, the radiological consequences of accidents previously evaluated in the SAR have not increased.

The Cycle 28 reload and fuel assembly changes do not create an accident of a different type than any evaluated previously in the SAR because (a) the core parameters are bounded by those assumed in accident analyses, and (b) the design parameters are still within the assumed ranges.

The required SDM is met for all power levels and at any time during the core life with the control rods above the insertion limits in the COLR.

The fuel assemblies of the Cycle 28 reload with ZIRLO material meet the same design criteria, applicable standards, and safety limits as those fuel assemblies with Zircaloy-4 material in the remainder of the core. No new single failure mechanisms have been created under normal or accident conditions.

The Cycle 28 reload does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR because the reload core does not affect equipment operation. The reload core does not cause equipment to malfunction.

All design criteria, applicable standards, and safety limits are met for the fuel assemblies with fuel rod cladding, guide tubes, and instrumentation tubes fabricated with ZIRLO material. Meeting design criteria and applicable standards precludes new challenges to components and systems that could increase the probability of malfunction. No new failure modes or limiting single failures have

been created. The fuel design changes satisfy all design criteria, applicable standards, and safety limits; therefore, the probability of failure does not increase.

Gap reopening is not expected to lead to fuel failure. Violating the gap reopening SAFDL criteria does not result in exceeding the 17% oxidation limit.

The Cycle 28 reload does not increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR because the core characteristics are bounded by parameters assumed in the accident analyses. When deviations occurred, a reanalysis was performed to show that the acceptance criteria was still satisfied.

The mechanical changes to the fuel assemblies do not degrade fuel performance. All design criteria, applicable standards, and safety limits are met for the fuel assemblies with fuel rod cladding, guide tubes, and instrumentation tubes fabricated with ZIRLO material. Meeting design criteria and applicable standards precludes new challenges to components and systems and/or a challenge to the integrity of the fuel rod cladding. No new failure modes or limiting single failures have been created. The doses documented in the UFSAR remain unaffected by the change in material to ZIRLO in the fuel assemblies.

Gap reopening does not affect the consequences of equipment malfunction.

The Cycle 28 reload, with the associated mechanical changes to the fuel assemblies, does not create the possibility of a new type of malfunction of equipment important to safety not previously evaluated in the SAR because (a) the core parameters are bounded by those assumed in accident analyses, (b) the design parameters are still within the assumed ranges, and (c) the limitations imposed by the MSLB are within normal ranges of operation.

All design criteria, applicable standards, and safety limits are met for the fuel assemblies with fuel rod cladding, guide tubes, and instrumentation tubes fabricated with ZIRLO material. Meeting design criteria and applicable standards precludes new challenges to components and systems and/or a challenge to the integrity of the fuel rod cladding. No new failure modes or limiting single failures have been created.

Previous analyses assumed no gap reopening for simplicity. Analyses with gap reopening show acceptable consequences. This condition is acceptable provided that continued compliance with the 17% oxidation limit is maintained. This compliance has been confirmed with a cycle-specific corrosion analysis.

The Cycle 28 reload, with the associated mechanical changes to the fuel assemblies and the identified limitations, does not reduce the margin of safety as defined in the basis for any technical specification because it meets all design criteria, applicable standards, and safety limits set forth in

the licensing basis.

With respect to the fuel assemblies, the two design types, OFA (Zircaloy-4) and VANTAGE+ (ZIRLO), meet all the design and safety limits.

SEV-1133
MINIMUM AUXILIARY FEEDWATER TEMPERATURE OF 32°F

The purpose of this evaluation is to determine if an unreviewed safety question exist with reducing the minimum temperature assumed in accident analysis for auxiliary feedwater from 50°F to 32°F. The only components and systems affected by the proposed change are the Condensate Storage Tanks (CSTs) and Auxiliary Feedwater System (AFW) which provide auxiliary feedwater to the steam generators (SGs). Normally, this is during plant startup/shutdown or following a reactor trip.

The function of the CSTs and AFW is to supply feedwater to the SGs for decay heat removal or during periods of low power operation when the main feedwater system is secured and/or the turbine is not latched. Reducing the minimum temperature increases the heat removal of the auxiliary feedwater which is beneficial except for over cooling transients, which have been evaluated.

Implementation of this change does not increase the probability of occurrence of an accident previously evaluated in that the change does not introduce any effect that could increase the probability of an accident. The systems affected by the change have been evaluated for the new minimum temperature, and it is within there design.

The affected accidents have been reevaluated with the new lower temperature. Implementation of this change does not increase the consequences of an accident previously evaluated in that the consequences meet the required acceptance criteria; thus the consequences are acceptable.

Implementation of this change does not increase the probability of occurrence of a malfunction of equipment important to safety in that the temperature reduction does not cause equipment or piping to operate outside its design temperature range.

Implementation of this change does not increase the consequence of a malfunction of equipment important to safety previously evaluated in that the change does not impact the capability to meet the accident analysis nor does it adversely impact the ability of any equipment to perform their intended safety function.

This change does not introduce the possibility of an accident of a different type than previously evaluated in that the change affects only the parametric value used by the current analysis.

This change does not introduce the possibility of an equipment malfunction of a different type than previously evaluated in that the change affects only the parametric value used by the current analysis.

This change does not reduce the margin of safety as defined in the basis for the Technical

Specification in that the accidents affected still meet the required acceptance criteria. Since the acceptance criteria are met there is no reduction in the margin of safety.

SEV-1134
REACTOR INTERNALS BAFFLE BOLT REPLACEMENT

The purpose of this safety evaluation is to examine the acceptability of 1999 baffle-former-bolt replacement strategy.

A newly developed body of information regarding baffle bolts concludes that plants can be safely operated and shutdown, during normal, upset and faulted conditions, with fewer baffle-former-bolts than are initially installed in the formers provided that physically sound bolts are configured in a pattern which provides for the correct structural resistance to the credible forces applied.

The Nuclear Regulatory Commission (NRC) has evaluated the acceptability of performing baffle-bolt replacements under the auspices of 10CFR50.59. (NRC Safety Evaluation of Topical Report WCAP-15029, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions", TAC No. MA1152.)

In the review of WCAP -15029 the NRC concluded that the general methodology presented was acceptable provided that : 1) the limiting baffle bolt loading will be determined by analysis for a class of plants and a specific break; 2) the noding to be used in the representation of the loading is demonstrated to be adequate by performing nodalization sensitivity studies or by some other acceptable methodology. The review furthermore required the demonstration of conservatism in projected bolting material properties and for the accounting of limitations in the inspection methods used to detect flaws.

The Rochester Gas and Electric (RG&E) baffle-former-bolt replacement strategy differed slightly from the generic Westinghouse Owners Group (WOG) strategy (WCAP-15036, revision 1). RG&E performed ultrasonic testing of all the inspectable bolts. This examination identified 59 bolts with flaw indications. The replacement program changed 40 of these bolts along with 9 bolts that could not be tested and 7 bolts classified as suspect. All 19 bolts with defect indications in Former Plate levels 2 through 6 were replaced regardless of their location within the WOG pattern. All bolts that exhibited defects in Former levels 1 and 7 were left in service. Of the 6 bolts which exhibited flaws in positions required to conform to the WOG pattern in levels 1 and 7, sufficient adjacent bolts outside of the pattern were verified to be present to structurally compensate for the potential defects. The RG&E pattern of analysis thus incorporated both new and aged, but verified acceptable, bolts. The end result of this methodology yielded an output which provides reasonable assurance that all the fasteners relied upon within the analysis will function as intended.

The technical evaluations performed for RG&Es baffle-former-bolt replacement program demonstrate that the reactor internals package will function as required under faulted load conditions, as well as comprehensively addressing the NRCs criteria for making the change under 10CFR50.59, as delineated in their review of the WCAP-15029.

The baffle-former-barrel and baffle-former-bolts react to the stresses caused by the accidents evaluated in the SAR but do not have a failure mode that is a precursor to any analyzed accident. Because the units do not have a failure mode that leads to an accident the former-bolt replacement can not increase the probability of occurrence of a accident previously evaluated in the SAR.

After completion of the change the baffle-former-bolting arrangement will have sufficient structural integrity to resist the force loading caused by the analyzed accident set. Because the as-left condition will perform its function in accordance with the design requirements the consequences of an accident previously evaluated in the SAR will be unchanged.

The baffle-formers physically and functionally interact with the reactor core barrel and provide lateral fuel support and reactor coolant flow direction. The baffle-former-bolting replacement does not add any new functional interactions with equipment important to safety. The analysis of the as-left baffle-former-bolting configuration demonstrates that the units have sufficient integrity to resist all credible design basis loading conditions. After the change application of the design accident loading forces to the units will not cause them to physically fail, or otherwise deform, to an extent which could cause fuel damage or preclude the ability to cool the fuel following an accident. Accordingly, it can be concluded that the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR is unchanged.

The analysis of the as-left baffle-former-bolting configuration demonstrates that the units have sufficient integrity to resist all credible design basis loading conditions. After the change application of the design accident loading forces to the units will not cause them to physically fail, or otherwise deform, to an extent which could cause fuel damage or preclude the ability to cool the fuel following an accident. After the baffle-bolt replacements the reactor core internals will respond to the stress of accidents and malfunctions the same way as before the change. Therefore the consequences of a malfunction of equipment important to safety previously evaluated in the SAR is unchanged.

The baffle-former-bolting replacement program does not alter the design, fit, form, or function of the reactor internals. There are no new functional interactions created by bolt replacements. After completion of the change the reactor internals package will function as before the change. Because this change does not introduce any new design or functional interactions it can not possibly introduce the potential for an accident of a different type then previously evaluated in the SAR.

The baffle-former-bolting replacement program does not alter the design, fit, form, or function of the reactor internals. There are no new functional interactions created by bolt replacements. The analysis of the as-left baffle-former-bolting configuration demonstrates that the units have sufficient integrity to resist all credible design basis loading conditions. After the change application of the design accident loading forces to the units will not cause them to physically fail, or otherwise deform, to an extent which could cause fuel damage or preclude the ability to cool the fuel

following an accident. After the baffle-bolt replacements the reactor core internals will respond to the stress of accidents and malfunctions the same way as before the change. This change does not introduce any new design or functional interactions therefore it can not possibly introduce the potential for a malfunction of a different type then previously evaluated in the SAR.

Technical specifications presumes that the reactor vessel internals are configured to function as designed and licensed. The baffle-formers and baffle-former-bolts are not specifically addressed in the technical specifications. Because, after the bolt replacements, the vessel internals will function as technical specifications assumes it can be concluded that the margins of safety presumed in the bases for technical specifications will not be reduced.

10CFR50.59 SAFETY REVIEW FOR PCN 98-3013 TO PROVIDE GUIDANCE ON THE TWO OPERATOR CONCURRENCE RULE

RG&E responses to IE Bulletin No. 79-06A, REVIEW OF OPERATIONAL ERRORS AND SYSTEM MISALIGNMENTS IDENTIFIED DURING THE THREE MILE ISLAND INCIDENT, state that administrative procedures were modified to require that two licensed operators shall agree on any overriding before the overriding action is executed on any safeguard system active component. The change modifies that commitment to now require that the action be approved by the SRO that has the command function. This change still meets the intent of the NRC guidance which is to have the control board operators not stopping active components without thoroughly evaluating the conditions and consulting other individuals (ie, the SRO).

The change will result in the procedural direction remaining in compliance with Technical Specification 5.1.2, which states that the Shift Supervisor or another SRO in his absence shall be responsible for the control room command function.

The change will not alter the description in the UFSAR which states that the Shift Supervisor is responsible for the performance of all personnel assigned to his shift who could affect plant safety, regardless of specialty affiliation. The UFSAR also states that the operating shift crews conform to the requirements for shift complement as specified in 10 CFR 50.54 (l), where an individual licensed as a senior operator is to be responsible for directing the licensed activities of licensed operators. 10CFR50.54 (x) and (y) also state that a licensee may take reasonable action that departs from a license condition or a Technical Specification in an emergency when, as a minimum, it is approved by a licensed senior operator.