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FACILITY CHANGES, TESTS AND EXPERIMENTS

CONDUCTED WITHOUT PRIOR APPROVAL

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

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SEV-1000
UNDERVOLTAGE PROTECTION SYSTEM DESIGN

The purpose of this safety evaluation is to determine if there are any unreviewed safety questions related to modifying the undervoltage protection system (UVPS) for Buses 14, 16, 17, and 18. This modification is necessary to eliminate a potential single failure point.

The UVPS for each safeguards bus consists of two channels with each channel containing a degraded voltage and loss of voltage relay. Either relay will activate the channel; however, both channels must actuate to produce an UV signal on the bus. In addition to the degraded voltage and loss of voltage relays (i.e., 27 relays), the UVPS contains logic that is used to trip relays (i.e., 27X relays) which pick up contacts and initialize load shedding on the bus. The UVPS logic is energized from power supply converters in the UV Control Cabinets which receive power via Instrument Bus A (for Buses 14 and 18) and C (for Buses 16 and 17). These power supply converters are used to transform 120 VAC power to 12 VDC power. All devices in the UVPS logic fail safe upon loss of power (i.e., generate an UV signal) with the exception of opto-isolators which are energized by power supply converter PS-2. The loss of power to PS-2 will fail the UVPS for the respective safeguards bus under all conditions.

The instrument buses supplying the UVPS logic normally receive power from either Battery A (for Instrument Bus A) or B (for Instrument Bus C) via inverters. A failure of the inverter coincident with an undervoltage condition or loss of offsite power (LOOP) would deenergize PS-2 and prevent an UV signal because the opto-isolators are de-energized. Since the loss of an inverter is a credible single failure, and a LOOP is a design basis event, it is desirable to correct this design deficiency.

The proposed modification will replace the current 120 VAC instrument bus power supply to PS-2 in the UVPS Control Cabinets with 125 VDC power from the batteries. Power supply converters PS-1 and PS-4 will be removed from the UVPS Control Cabinets and all UVPS logic devices, including the opto-isolators, will now be powered from PS-2. Power supply converters PS-1 and PS-2 in the UVPS Relay Cabinets and PS-3 in the UVPS Control Cabinets only supply power to indicating lights and diagnostic systems which are not required to operate the system.

The new power supply converter PS-2 will utilize an existing 125 VDC automatic throwover switch to enable either battery train to provide power. This throwover switch is already used to power other components contained in the UVPS design. The modification will also replace the existing 120 VAC to 12 VDC power supply converter PS-2 with a new 125 VDC to 12 VDC converter due to the use of a new electrical source.

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 modification. The UVPS is only used to mitigate a loss of voltage to 480 VAC safeguard buses and is not normally considered with respect to initiating an accident. In addition, the modification will increase the reliability of the UVPS following a LOOP by eliminating the failure of the UVPS to perform its safety function resulting from a single failure of the inverters and instrument buses. Consequently, there is no increase in the failure probability of any equipment important to safety. The modification does not directly involve the mitigation of radiological consequences of an accident nor any of the fission product barriers. Therefore, the modification will not increase the calculated radiological dose to the general public for any event evaluated in the UFSAR.

The possibility for an accident or malfunction of a different type than any evaluated previously in the UFSAR will not be created by the proposed modification. The modification will increase the reliability of the UVPS following a LOOP by eliminating the failure of the UVPS to perform its safety function resulting from a single failure of the inverters and instrument buses. Since the DC throwover switches are utilized in the current UVPS design, using the throwover switch as a source of power for the UVPS logic does not create a new failure mechanism.

The margin of safety as defined in the bases for any technical specification is not reduced by the proposed modification. The UVPS will remain capable of performing its function to monitor voltage on the four 480 VAC safeguard buses under all accident conditions. The timing response of the UVPS will also remain the same such that no margin of safety limits are affected.

SEV-1001
RCP OIL LEVEL INSTRUMENTATION UPGRADE - PHASE TWO

EWR 4534B is the second phase of a modification which replaces existing RCP motor oil level monitoring instrumentation. Phase two addresses RCP A only; RCP B and control room work is complete.

The new oil level transmitters and level switches used for this modification have been installed by the RCP vendor during RCP motor refurbishment. This scope of work proposed under this modification involves the installation of Namco quick disconnect plugs and receptacles as well as the flexible conduits necessary to support the new configuration. No new power supplies or changes to existing power supplies are proposed for this modification.

The new transmitters are the functional equivalent of the existing ones. The purpose of installing new transmitters is to increase instrument accuracies and ease of RCP maintenance. The Namco plugs and receptacles proposed for installation during this modification are designed to facilitate calibration and maintenance. The new plugs will be spliced into the existing transmitter cables.

This change is being proposed as a modification which will increase transmitter accuracies and ease of calibration and also supply plant operators with specific alarm indication as to which RCP motor bearing has reached an alarm setpoint.

This proposed modification does not change the critical design parameters of Reactor Coolant pumps. Because this modification does not affect RCP operation it does not alter the probability of the loss of a RCP causing a loss of flow type accident.

Should the failure of a RCP cause a Loss of Flow accident, the Reactor Protection System activates a reactor trip and, in conjunction with RCP flow coastdown, prevents fuel damage. These actions utilize equipment independent of this modification. The consequences of a Loss of Flow accident is dependent on these features occurring within a specified time and the amount of heat generated in the core. Because the proposed modification does not input to the above features, it does not alter the consequences of a Decrease in Reactor Coolant System Flow.

The proposed modification addressed in this modification does not interact with any safety related power supplies or instrumentation and control circuits. This modification is a functional equivalent of the existing system. Because this modification does not interact with equipment important to safety and only provides indication and does not alter any system functions, this modification will not create the possibility of a new type of accident.

The margins of safety associated with Decrease in Reactor Coolant System Flow are factored into fuel design and Reactor Protection System sensed parameters and their corresponding actuation setpoints. Because this modification does not involve the Reactor Protection System or its setpoints, no reduction to a margin of safety will result.

SEV-1002
LLRW INTERIM STORAGE FACILITY

EWR 10018 is a request to design, locate, procure and install a controlled facility for the temporary storage of low level, solid, radioactive waste pending the shipment of the waste off site to a long-term storage facility or disposal site. The location of this facility will be on site, in the Northeast quadrant of the plant site, on the open area between the Radwaste Storage Building and the All Volatile Treatment (AVT) Building. This Low Level Radioactive Waste (LLRW) interim storage facility will provide temporary storage for handling up to five (5) years of projected solid radioactive waste, and include any additional operating margin to store the waste material generated by the Ginna Nuclear Power Plant.

The addition of this facility does not increase the probability of an occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in that the facility is remotely located several hundred feet away from all plant safeguards equipment and functions.

The results of the design analysis demonstrates that the results of an HIC accident are bounded by a fuel handling accident. Therefore, the addition of this facility does not create the possibility of an accident or malfunction of a different type than previously evaluated in the safety analysis report in that possible events are limited to a potential solid radioactive waste spill with consequences less than ten (10) per cent of the 10CFR100 limits.

The addition of this facility does not reduce the margin of safety as defined in the basis for any Technical Specification. The LLRW facility is not within the scope of the Technical Specifications.

Therefore, the addition of this facility does not invalue any unreviewed safety question.

SEV-1003
CCW AND SW PUMP DISCHARGE CHECK VALVE REPLACEMENT

This phase of EWR 5284 will replace the Component Cooling Water (CCW) and Service Water (SW) pump discharge check valves. Each of these check valves have been evaluated for their present condition of integrity and it has been determined that replacement is necessary.

The valves being replaced during the 1994 outage are:

CCW - 723A and 723B
SW - 4601 and 4602

The valves being replaced at a future date are:

SW - 4603 and 4604

The partial replacement of the SW valves described above has been evaluated for hydraulic impact and has been found to be acceptable.

The reliability of the existing CCW check valves is in question because on several occasions during ASME Section XI testing the valves did not immediately close when the opposite header pump was tripped. In all cases, subsequent testing did demonstrate that the valves properly responded. Additionally, it is suspected that existing valves cause a significant waterhammer as they close, possibly contributing to evidence of cracked flooring in the vicinity of the CCW pumps.

The SW system check valves are being replaced because of significant degradation caused by various corrosion mechanisms which are prevalent in the service water system. Previous inspections of these valves have discovered a variety of corrosion defects which may ultimately render the valve inoperable.

The pump check valves must be exercised in their full open and closed position to verify fulfillment of its safety function performance, in accordance with ASME B&PV Code, Section XI, Article IWV-3522. The safety functions of check valves are to prevent reverse flow of the pumps when one is inadvertently or intentionally stopped and maintain pressure integrity of the system. In this way the flow from the running pump is maintained through the system providing adequate cooling to essential safety related components.

Replacement of these check valves will utilize an enhanced design known as a "nozzle check valve". A nozzle check valve is distinguished by its valve disc being structurally fitted into the middle of the flow stream. The advantages of this design are:

1) Minimal flow velocity required to keep the valve fully open.
2) Rapid response to reverse flow.
3) Minimum maintenance requirements.
4) Spring assist for more reliable closure.

These features will improve system design by:

1) Minimizing water hammer when closing.
2) Minimize wear by being fully open and stable during normal operation.
3) Reliably responding to reverse flow.

The proposed modification would not increase the probability of occurrence of an accident evaluated previously in the UFSAR because:

a. The equipment proposed would not introduce any likely ignition sources that could start a fire.

b. New piping and valves interfacing with the Component Cooling and SW System would be sized, specified, and installed in accordance with the existing piping classifications and code designations for design, material, and construction.

c. The CCW and SW Systems are used to mitigate accidents and consequently, is not expected to affect the probability of occurrence of an accident.

The proposed modification would not increase the consequences of an accident previously evaluated in the UFSAR because:

a. The replacement check valve would not degrade the ability of the CCW or SW pumps to deliver their required flow during a design-basis accident since a design analysis will confirm that the minimum flow requirements are satisfied.

b. All equipment, piping and conduit will be anchored and supported so that it does not affect safety related equipment during a seismic event.

The proposed modification would not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR because:

a. New piping and valves would be sized, specified and installed in accordance with existing piping classifications and code designations for design, material and construction.

b. The valve being a replacement for an existing valve would be required to meet the original system design bases and performance criteria.

c. The new check valve would in fact lessen the probability of occurrence of a malfunction of equipment important to safety. The new valve will be better designed for the system conditions thereby providing for more reliable valve operation due to prevention of damage to internal working parts. This will also ensure better leak tightness when the valve is closed.

The proposed modification would not increase the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR because:

a. The replacement check valve would be required to meet the original system design bases and performance criteria.

b. Plausible accidents have been evaluated in the UFSAR. Failure of the replacement check valve would result in the same consequences as would the failure of the existing check valve, which was evaluated.

c. Since the proposed modification does not impact any other systems, the replacement of the check valve would not degrade fire protection, separation, power supply availability, seismic integrity or any other assumptions in the UFSAR accident analysis.

The proposed modification would not create the possibility of an accident of a different type than any previously evaluated in the UFSAR because:

a. The modification involves replacement of an existing valve with a functionally similar valve. The replacement valve will be designed (including material selection), fabricated, inspected and tested to the same codes and standards as the existing valve such that the replacement valve is technically equivalent or superior to the existing valve.

b. The new check valve will be required to meet the original system design basis and performance criteria of the existing valve.

c. All plausible accidents for the existing check valve were evaluated in the UFSAR. Since the new valve is a suitable replacement, the proposed modification would not create the possibility of an accident of a different type than any previously evaluated.

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:

a. The replacement check valve would be required to meet the original system design basis and performance criteria.

b. Replacement of the check valve will not create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated since the valve is technically improved and functionally equivalent to the existing valve.

The proposed modification would not reduce margins of safety as defined in the basis for any technical specification because:

a. The check valve is a replacement for an existing valve and is required to meet system design bases and performance criteria. The valve is technically better than the original valve. Therefore, CCW and SW pump delivery will not be significantly altered. The proposed new valve has a slightly lower C_v than the existing valve and consequently a higher differential pressure. This difference will be evaluated to demonstrate that the system design margins are not exceeded.

Based on the preceding, the proposed modification does not involve a change in the Technical Specification or the UFSAR and is not an unreviewed safety question.

SEV-1004
CONTAINMENT PLATFORM UPGRADE - PHASE 3

The proposed modifications detailed below constitute personnel safety enhancements only. This Safety Evaluation will document that there are no changes to the facility as described in the Safety Analysis Report as a result of these modifications.

A lifeline system (MANSAFE) will be installed on the east side of the reactor cavity approximately 5'-3" above the operating floor. It will run from the north end of the shield wall approximately 12 feet south with supports anchored directly to the shield wall. The MANSAFE system consists of stainless steel cable and stainless steel brackets which are anchored to structural elements.

A MANSAFE system will be installed on the west side of the reactor cavity approximately 4'-6" above the operating floor running from column 102 to the south end of the shield wall. It will be supported at column 102 and anchored directly to the shield wall.

A second MANSAFE system will be installed on the west side of the reactor cavity approximately 3'-6" above the operator floor. It will run from column 107 approximately 15 feet north with supports attached to column 107 and existing cable tray supports.

Installation of handrail and kickplate along a portion of the catwalk at elevation 295' near the A steam generator.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR is not increased. There are no accidents, transients, or malfunctions analyzed in the UFSAR that involve the MANSAFE system and the system is not utilized in the mitigation of any events analyzed. The modification design and installation involves personnel safety, reducing the risk of injury due to personnel falling.

The possibility for an accident or malfunction of equipment of a different type than any previously evaluated in the UFSAR is not created. The mounting of the MANSAFE system and structural integrity of the concrete shield walls and cable trays to which the MANSAFE is attached is governed by existing design methodology previously applied for the attachment of the supports. The possibility of a missile being created by the failure of the MANSAFE system during a seismic event does not exist, because the design requirements require analysis to show the anchored structure capable of withstanding all design loads.

The margin of safety as defined in the basis for any technical specification is not reduced. The design and installation of the MANSAFE system involves enhancements to ensure personnel safety by reducing the possibility of injury due to falling while working near the reactor cavity. The modification does not

involve any systems or equipment required by Technical Specifications.

SEV-1005
SERVICE WATER FOULING (PHASE 3)

The purpose of this safety evaluation is to evaluate the changes proposed under EWR 4658C with respect to the original safety considerations of the plant design basis and to determine if NRC approval is required (if the modification involves an unreviewed safety question or affects the plant Technical Specifications).

EWR 4658C is the third phase of the Service Water Fouling modification project (EWR 4658). This phase of the project addresses issues summarized below.

AFW/SAFW Pump Suction Piping Flushing Capability

The scope of work proposed under this portion of the modification involves replacing five existing 3/4" and 1" piping connections (and attached drain valves) with new 2" connections and valves to permit higher flushing capacities to break up and remove the silt buildup conditions. Other than providing higher flow capacities, the new drain valve configurations are the functional equivalent of the existing ones.

Component Cooling Water (CCW) Heat Exchangers Piping

The scope of work proposed under this portion of the modification involves the installation of two new thermowells and temperature elements on the shell-side (CCW) outlets of each of the two CCW heat exchangers. The CCW temperatures will be used to perform independent thermal performance testing of each CCW heat exchanger without isolating the parallel heat exchanger.

The new temperature connections and associated instrumentation are only intended to provide local indication during periodic surveillance testing. No new control features or changes to any existing system/component controls are introduced as a result of this modification. Also, no new power supplies or changes to existing power supplies are proposed for this modification.

CRFC Fan Motor Cooler Piping

The scope of work proposed under this portion of the modification involves the installation of a new thermowell and temperature element on the connecting SWS piping attached to the outlets of each of the four Containment Recirculating Fan Cooler (CRFC) fan motor cooler coils.

CRFC fan motor cooler coil exit temperatures will be used to perform a heat balance to, subsequently, determine an accurate indication of SWS flow through each CRFC fan motor cooler coil.

This instrumentation is necessary to support a performance testing program for safety-related heat exchangers as defined in Generic Letter 89-13. The instrumentation will be used to determine the heat removal capability of these SWS heat exchangers and to monitor/trend the heat removal capability over time via periodic performance testing.

Certain safety related heat exchangers have already been equipped with new pressure and temperature instrumentation under phases 1 and 2 of the EWR 4658 modification. A Safety Analysis was previously completed to evaluate the changes performed under the scope of the first two phases.

The new temperature connections and associated instrumentation are only intended to provide local indication during periodic surveillance testing. No new control features or changes to any existing system/component controls are introduced as a result of this modification. Also, no new power supplies or changes to existing power supplies are proposed for this modification.

This modification does not increase the probability of occurrence (or consequences) of an accident, or a malfunction of equipment important to safety, previously evaluated in the SAR. The modifications alters only piping pressure-boundary items and are independent of accident mitigation features.

This modification does not create the possibility of an accident, or a malfunction of equipment important to safety, of a type different from any previously evaluated in the SAR. The modifications affect only the pressure-retaining function of valves, and piping components, and are independent of accident mitigation features.

This modifications will not reduce the margin of safety as defined in the bases for any technical specifications. The modifications do not affect any automatic actuation signals or the operability of any of the components involved, nor will the functions that those components currently perform be altered.

Consequently, this modification does not involve an unreviewed safety question.

SEV-1006
RCS COLD LEG TEMPERATURE RECORDING

The purpose of this safety evaluation is to determine if there are any unreviewed safety questions related to the work scope of EWR 10106, "RCS Cold Leg Temperature Recorder." This modification is necessary to achieve a configuration which is consistent with RG&E's position regarding implementation of Supplement 1 of NUREG 0737 at Ginna Station.

The existing configuration of RCS cold leg temperature monitoring instrumentation consists of the following:

1) Two Class 1E instrument channels, T409B-1 and T410B-1, which provide Main Control Board indication and input to RVLMS.

2) Two Non Class 1E instrument channels, T450 and T451, which provide Main Control Board chart recording, annunciator alarms and inputs to the Plant Process Computer System (PPCS).

RCS cold leg temperature is classified as Category 1, Type A variable by RG&E. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident", recommends that Category 1 instrumentation include provisions for recording the RCS cold leg temperature variable. The two existing instrument channels which are qualified for Class 1E service do not provide a means of recording the temperature variable. RG&E is currently committed to the NRC to provide recording of the Class 1E RCS cold leg temperature channels during the 1994 refueling outage.

This EWR Shall modify this configuration as follows:

1) Input of the Non Class 1E instrument channels, T450 and T451, to MCB chart recorder RK-3 will be disabled.

2) Isolated inputs from Class 1E instrument channels, T409B-1 and T410B-1 will be provided to MCB chart recorder RK-3 and the PPCS.

3) MCB chart recorder RK-3 will be rescaled to display the 0 - 700°F temperature range of the Class 1E channels.

The proposed modification will not increase the probability of occurrence, or the consequence of, any accident or malfunction of equipment important to safety. The proposed modification is properly isolated from the safety related portions of the Tcold indication circuitry via qualified isolators. The isolators installed are within the loading capabilities of the racks in which they are installed. Therefore, the installation will not cause a malfunction of equipment important to safety.

The proposed modification does not introduce the possibility for an accident or malfunction of a different type than previously evaluated in the Safety Analysis Report. The modification only affects indication channels, and therefore, cannot be an accident initiator.

The proposed modification does not alter the function of any system used in accident mitigation. The modification enhances the ability of operators to mitigate accidents by providing recording of a post-accident Type A variable.

The margin of safety as defined by plant Technical Specifications is not affected by this modification. This modification is properly isolated from any instrumentation required by Technical Specifications, specifically the instrumentation specified in Table 3.5-3, "Accident Monitoring Instrumentation".

SEV-1007
RHR REDUNDANT FLOW LOOP

The design of the proposed modification addressed in this safety analysis is documented in EWR 4970, "Residual Heat Removal System Redundant Flow Loop."

The flow delivered to the reactor vessel by the RHR System during a design basis accident is a Category 1, Type A, post accident monitoring variable as defined by RG&E. RG&E's position regarding implementation of Supplement 1 of NUREG 0737 requires redundant Class 1E process measurement instrument channels for variables identified as Category 1, Type A. The existing post accident RHR flow measurement instrumentation consists of only a single Class 1E instrument channel without any redundancy provisions and is therefore inconsistent with RG&E's position regarding implementation of Supplement 1 of NUREG 0737 at Ginna Station.

EWR 4970 will install a redundant residual heat removal flow instrument channel. Installation of the redundant channel is necessary to complete a regulatory commitment with respect to Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."

The basic scope of this modification involves adding a new pressure transmitter, piping/tubing and valves to existing spare taps of RHR flow orifice FE-626, utilizing spare signal processing modules in the FOX1 Rack, changing the existing Main Control Board RHR flow indicator from single to duplex, and separating the RHR flow inputs to the Reactor Vessel Level Monitoring System (RVLMS). Also, an isolated output from the new channel will be provided to the Plant Process Computer System (PPCS) for display and recording capability.

This proposed modification represents a change to the facility as described in the Final Safety Analysis Report (FSAR) in that only one channel of RHR flow is currently addressed. These changes produce the benefit of providing diverse, independent, post accident qualified RHR flow indication and also provide the ability to maintain one channel of RVLMS operable when performing RHR flow instrument channel maintenance.

The proposed modification will not increase the probability of occurrence, or the consequence of, any accident or malfunction of equipment important to safety.

The proposed modification has no functional impact on the RHR system, and has no potential to introduce a failure which in and of itself can act as a precursor or initiating event for an accident analyzed in the UFSAR. Therefore, the proposed modification does not increase the probability or consequences of any evaluated accidents.

The proposed modification is designed such that the pressure boundary interface with the RHR system meets the system design requirements. Because the piping system interfaces are being made to an existing RHR flow element, a pressure boundary failure of the equipment installed under the proposed modification will have exactly the same RHR system effects as a failure of the existing equipment.

The electrical portion of the proposed modification is designed to meet safety grade (Class 1E) standards. Isolation is provided so that a failure in the new equipment will not propagate and introduce a failure to other, functionally unrelated, safety systems.

The proposed modification does not introduce the possibility for an accident or malfunction of a different type than previously evaluated in the Safety Analysis Report.

The proposed modification does not alter the function of any system used in accident mitigation. The design characteristics of the modification ensures no new, unbounded failures will be introduced to plant systems. Because no plant functions are changed as a result of this modification, there is no possibility of creating new system interactions which might lead to a previously unanalyzed accident or an unbounded malfunction.

The margin of safety as defined by plant Technical Specifications is not reduced as a result of this proposed modification.

The RHR system parameter critical to the margin of safety is RHR flow (Tech Spec 4.5.2.1). The proposed modification does not alter system flow and therefore does not reduce the margin of safety associated with the host system.

SEV-1008
STEAM LINE MONITORS R31, R32

As a result of NUREG 0737 Radiation Monitors R-31 and R-32 were installed and the open/close status of the Main Steam Safety valves (3508, 3509, 3510, 3511, 3512, 3513, 3114 and 3515), Steam Generator Atmospheric Relief valves (3410 and 3411) and the Turbine Driven Auxiliary Feedwater Pump Steam Supply valves (3504A and 3505A) were monitored and the output recorded on strip chart recorders (RK-47A, RK-47B and RK-47C) in the Control Room. This was done so that radioactive releases could be quantified by integrating the product of steam activity and the exhaust flow rate with respect to time.

In response to the 1989 Ginna Station EOP audit finding 89-80-04, the steam line monitors display capabilities were evaluated. An apparent weakness in the Control Room operators' ability to discern main steam line radiation monitor (R31 and R32) readings from the RK-47A, RK-47B and RK-47C chart recorders located in the RMS rack was identified. Per RG&E IOC, the NRC observed operators on the simulator and noted that there was confusion relating to the steam line radiation monitors and their use in event diagnosis.

Presently, RK-47A records steam line radiation as detected by radiation detectors RE-31 and RE-32. RK-47B records the positions of valves associated with the B Steam Generator (3508, 3510, 3512, 3514, 3410 and 3504A). RK-47C records the positions of valves associated with the A Steam Generator (3509, 3511, 3513, 3515, 3411 and 3505A). The recorders turn on and the PPCS alarms when main steam line activity in either steam line goes above 0.1 mR/hour. The Main Control Board displays the positions of valve 3504A, 3505A, 3410 and 3411. The Plant Process Computer System displays and maintains a record of the positions of valves 3504a, 3505A, 3410 and 3411 and steam line radiation as detected by radiation detectors RE-31 and RE-32. The duration of time the valves are not closed is used with steam line radiation to calculate the amount of effluent released through the steam lines during accidents.

The scope of this modification is to remove recorders RK-47A, RK-47B and RK-47C. Two independent indicators with alarm audible will be installed in the space vacated by the recorders to provide continuous indication and alarm capability of steam line radiation as detected by radiation detectors RE-31 and RE-32. An audible alarm will sound when main steam line activity in either steam line goes above the alarm setpoint. Main Control Board valve indications will remain unchanged. The information presently available on the PPCS will remain intact and information will be added to the PPCS. The positions of the A and B Main Steam Safety valves (3508 to 3515) will be diverted from the recorders to the PPCS via the MUX racks. New data points, with the calculated effluent release, will be added to the PPCS.

This modification does not affect the operability of any of the components involved. Only the method of monitoring the status or output of the equipment is being changed. Thus, this modification will not increase the probability of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

The components involved in this modification will be installed to monitor the consequences of accidents, but, do not themselves impact the operability of the components. Even if the indicators and the PPCS were to simultaneously malfunction, it would not alter the consequences of an accident though it would affect the dose assessment capabilities. Therefore, this modification will not alter the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Since the circuits for the steam line monitors, ARVs and TDAFW pump supply valves are already in place and connected to the PPCS, the only change will be the routing of the output of the Main Steam safeties from the present recorders to the PPCS. The present position switches are isolated from the valve and do not breach the pressure boundary. This modification will not create the possibility of an accident or malfunction of a different type than any previously evaluated in the UFSAR.

This modification will not reduce the margin of safety as defined in the bases for any technical specifications. There is no impact on the operability of any equipment important to safety as a result of this modification.

SEV-1009
RECORDER REPLACEMENT PHASE 2

The purpose of this safety evaluation is to determine if there are any unreviewed safety questions related to the removal of Accident Monitoring Recorders RK-47D and RK-47E and replacement of Turbine Metal Temperature Recorder RK-30B. These recorders have become obsolete and difficult to repair since spare parts are no longer available for them. The recorder that will replace RK-30B will record the same variables as the present recorder and has the same features.

The proposed modification is changing the way Sump A Level, Sump B Level, Sub Cooling Margin (T_{SAT} - T_{HOT}) and Containment Vessel Wide Range Pressure are recorded and displayed. Operators will be able to retrieve this data from the PPCS. The above parameters are currently displayed on chart recorders RK-47D and RK-47E. Operators will still be able to verify the information from indicators on the Containment Isolation Reset Panel and/or the Main Control Board. The PPCS will maintain a record of the information for trending and historical purposes. All of the variables currently recorded on RK-47D and RK-47E are presently available in the PPCS with the exception of Containment Vessel Wide Range Pressure which will be added to the PPCS by utilizing two spare circuits and two spare computer taps.

The PPCS is considered to be an acceptable means of fulfilling the requirements of Reg. Guide 1.97 for recording these parameters. RG&E's position regarding implementation of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," stipulates that the variables presently recorded on RK-47D and RK-47E will be available in the PPCS but do not need to be recorded on a chart recorder.

The Sump A Level, Sump B Level and Sub Cooling Margin instrumentation loops do not provide logic signals to any of the systems needed to mitigate the consequences of any of these accidents. The Containment Vessel Wide Range Pressure loop provides a logic signal to the ESFAS system. Specifically, actuation of steam line isolation on hi-hi containment pressure "a" and containment spray on hi-hi containment pressure "b". Within that loop the control signal splits from the indicating signal at a qualified isolator. The proposed modification is changing the way these instrumentation loops are indicated. Operators will be able to retrieve this data from the PPCS instead of from a recorder. They will still be able to verify the information from indicators on the Containment Isolation Reset Panel and/or the Main Control Board. Therefore, the proposed modification does not affect any of the previously evaluated accidents, transients and special events as described in the UFSAR.

The Sump A Level, Sump B Level and Sub Cooling Margin instrumentation loops provide indication only and do not provide logic signals to any of the systems needed to mitigate the consequences of accidents. The Containment Vessel Wide Range Pressure loops provide a logic signal to Containment Spray and Steam Line Isolation. This modification does not affect the automatic actuation of these logic signals. All indication presently available to the operators will continue to be available. The functions that these components currently perform will not be altered because the modification involves only portions of the electrical loops which are isolated from the signals that actuate the containment spray pumps and close the main steam isolation valves.

Removal of recorders RK-47D and RK-47E will eliminate any potential failure mode associated with the recorders themselves, their power supplies and isolation of these recorders from the upstream instrument loops and actuation signals. Therefore, this modification will not increase the consequences or probability of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

The modification involves indication only and is independent of accident mitigation features. The re-routed circuits for the Containment Vessel Wide Range Pressure will be isolated from the indicator and from the ESFAS logic signal and thus will not create the possibility of an accident or malfunction of a different type than any previously evaluated in the UFSAR.

This modification will not reduce the margin of safety as defined in the bases for any technical specification. All indication presently available to the operators will continue to be available, both on the PPCS and on the Main Control Board or the Containment Isolation Reset Panel. This modification does not affect automatic actuation signals or the operability of any of the components involved, nor will the functions that these components currently perform be altered.

SEV-1010
UFSAR CHANGE ON CCW PUMP MATERIAL

Identified Deficiency Report (IDR) 0021-92 identified a concern on the Ginna Station Component Cooling Water (CCW) Pumps.

The IDR noted that statements, contained in the Ginna Station UFSAR and other various documentation such as letters to the NRC during the Systematic Evaluation Program, contained what appeared to be conflicting statements on the material used to construct the CCW pump casings.

The intent of this Safety Evaluation is to document the resolution of this discrepancy. No physical changes to plant systems are being performed.

The UFSAR section 3.2.2.1.3 will be revised to show the correct material used to fabricate the pump, that is, Cast Iron which is consistent with UFSAR section 9.2.2.3, Table 9.2-3 and additional testing performed by RG&E. Cast Iron is also consistent with the original Westinghouse "E" spec for the pump. The Westinghouse System description RGE-200/C/4 dated 12/27/67, p. 10 is believed to incorrectly represent the material as carbon steel. This evaluation supports UFSAR change 10/133.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR is not increased. The change proposed has been determined to have been a misrepresentation of the material for the CCW casing as carbon steel in lieu of the actual material of cast iron. The original design specified material was cast iron. An RG&E letter in response to an NRC request for information inadvertently listed the material as carbon steel. In a subsequent NRC letter the NRC referenced this material as carbon steel. The basis for the NRC's safety evaluation, however, did not involve the material type, but rather the fact that Ginna has redundant CCW pumps and backup capability which can be provided in a relatively quick period of time. It has also been established that, in concert with the NRC's statement, fracture toughness requirements were not imposed as part of the design requirements. Therefore, there is no increased probability or consequences of an accident or malfunction previously evaluated.

The possibility of an accident or malfunction of a different type than any evaluated previously in the UFSAR is not created. The fact that the ASME code does not require fracture toughness for this component, and the fact that the material was originally specified as carbon steel, justifies the conclusion that a failure of this component due to its material type is not a technical concern.

The margin of safety as defined in the basis for any Technical Specifications is not reduced. The CCW casing material is not associated directly with any of the Tech Spec requirements on the CCW system. The material selection of the casing has been established to have been properly the correct one used, since the original Westinghouse spec called for cast iron, and that material is utilized in the current installation.

SEV-1011
THROTTLING OF CCW TO RHR FLOW CONTROL VALVES 780 A AND 780 B

The proposed change involves the additional throttling of manual butterfly valves 780 A&B, which are the CCW system outlet valves from the RHR heat exchangers. These valves are to be throttled in order to reduce the CCW flow through the CCW heat exchangers (shell side) from approximately 3000 gpm to 2500 gpm and concurrently reduce CCW flow through the RHR heat exchangers (shell side) from approximately 2775 gpm to 1800 gpm. The basis for this change is to minimize the potential for flow induced vibration (FIV) in these heat exchangers caused by excessive inlet flow velocities, while continuing to provide all heat exchanger requirements to meet normal plant operation and post-accident functions.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased. There are no new components being added by the proposed change. It involves a change to the open position of valves 780 A and B by adjustment of valve position. The valves perform no function during normal power operation, because the RHR heat exchanger inlet valves MOV-738 A and B are closed. Therefore, the probability of an accident or malfunction occurring at power or the corresponding consequences as analyzed are not increased. During modes when the RHR heat exchanger receives flow from the CCW system, these valves serve to control (limit) the flowrate through the CCW system. Design Analyses have demonstrated that sufficient heat removal capability from RHR and CCW heat exchangers continues to provide margin for normal plant cooldowns and for long term cooldown during the recovery phase (recirculation phase) post-LOCA.

The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not increased. There are no new failure modes introduced, because additional throttling is being performed for a valve that currently functions in the throttled position. The degree of throttling to be performed is within the range for the valve type. The proposed change is being made to reduce potential malfunctions resulting from damage to the CCW and RHR heat exchangers due to flow induced vibration and follows the recommendations by both heat exchanger vendors. The proposed change is not creating any additional openings in primary (or secondary) systems nor adding additional piping or electrical controls. Therefore, there are no new accidents or malfunctions created.

The margins of safety as defined in the basis for any technical specification is not reduced. The valves being modified are not specified in the Technical Specifications and their functions are not credited in the basis for any technical specifications. Component Cooling water heat exchangers are required operable by 3.3.3.1b. and RHR heat exchangers operable by 3.3.1.1e. By reducing the maximum flowrate through these heat exchangers to minimize the effects of flow induced vibrations, the operability and need to repair these components can be expected to be improved.

SEV-1012
SAFETY EVALUATION FOR USE OF ETHANOLAMINE

Ethanolamine (ETA) is an alternate amine which is being increasingly used at PWR's for secondary system pH control. ETA is being considered for use on a trial basis at Ginna. Per the request of SFEWR 10167, the purpose of this Safety Evaluation is to determine if use of ETA at Ginna would create a change to the Technical Specifications or create an unreviewed safety question at Ginna. Ethanolamine is also equivalently referred to as monoethanolamine. Alternate amines such as 1,2 diaminoethane (DAE), 2-amino 2-methylpropanol (AMP), 3-methoxypropyl amin (MPA), 3-hydroxy quinuclidine (3HQ), and morpholine are not within the scope of this safety evaluation.

Ginna currently uses All Volatile Treatment (AVT) with ammonia for pH control and hydrazine for dissolved oxygen control. ETA has a lower relative volatility than ammonia (relative volatility is similar to the inverse of solubility). Since ETA tends to condense with the water phase more than ammonia, ETA is better suited to protect wet steam piping from corrosion by creating a higher pH in these areas. The low relative volatility may also allow increased condensate polisher run times since the amine remains in the heater drain system longer than ammonia.

ETA is an amine similar to morpholine. Morpholine was used at Ginna prior to 1974 when the station used phosphate secondary chemistry. Morpholine was used on a trial basis in Ginna's steam generators for wetlayup during the Spring 1992 Refueling Outage. ETA is a breakdown product of morpholine and was therefore present in the Ginna Secondary system prior to 1974 and in the steam generators in 1992.

ETA is recommended by EPRI's PWR Secondary Chemistry Guidelines. ETA has been used at Beaver Valley Unit 1, Davis Besse, Catawba, Prairie Island Unit 1, North Anna, Palo Verde, Diablo Canyon, Millstone Unit 2, ANO-2, Crystal River, and Connecticut Yankee.

ETA storage and injection designs for Ginna have not been prepared. The following assumptions are made for the purpose of this evaluation. Any deviations from these assumptions in detailed designs will require an additional safety review.

ETA will be stored in the Turbine Building north of the turbine centerline on the basement or mezzanine levels or outside the Turbine building within 50 feet of the ammonia storage tank.

ETA will be stored and injected at a concentration not to exceed 80% solution strength and conforming with the MSDS or an equivalent MSDS.

The ETA will be stored in nitrogen pressurized stainless steel tanks and will be transferred with compatible pipe/tubing. ETA storage will not exceed 1000 gallons.

ETA injection point(s) will be in the non-safety related portion of the condensate or feedwater systems. ETA concentrations in the secondary system will not exceed 15 ppm.

Plant Chemistry will establish ETA air monitoring methodology and toxicity limits to ensure the requirements of SC-6.3 will be met in the case of an ETA spill.

ETA storage and processing system will be contained such that any spill will not come into contact with plant materials that are not compatible with 80% ETA and any spill will not come within 50 feet of a control room air in-take.

This activity does not increase the probability of an occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in that the activity does not adversely impact any existing plant equipment or systems nor affect existing accident analyses.

This activity does not create the possibility of an accident or malfunction of a different type than previously evaluated in the safety analysis report.

This activity does not reduce the margin of safety as defined in the basis for any Technical Specification in that the activity is not within the scope of the Technical Specifications.

SEV-1013
BORIC ACID PIPING RELIEF VALVE ADDITION

Two relief valves are planned to be installed in the boric acid piping. The relief valves are to provide overpressure protection of the line between valves 826A/826B and 826C/826D in the event that heat tracing is turned on with these pairs of valves closed. Vent valves 1821 and 1822 are planned to be changed from normally closed to normally open. The safety classification of the lines between 826A/826B and 826C/826D is planned to be changed from Safety Class 2 to Safety Significant. Reference Technical Specification Amendment #57 and SFEWR 10178 for further background. The purpose of this Safety Evaluation is to determine if these changes create a change to the Technical Specifications or create an unreviewed safety question.

The addition of the relief valves does not increase the probability of an occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in that the relief valves serve a piping section that performs no safeguards functions.

The addition of the relief valves does not create the possibility of an accident or malfunction of a different type than previously evaluated in the safety analysis report in that the effected their placement are limited to non safeguards equipment and systems.

The addition of the relief valves does not reduce the margin of safety as defined in the basis for any Technical Specification in that the change is to non-safeguards equipment as described by Technical Specification Amendment #57.

SEV-1014
TEMPORARY REMOVAL OF PRESSURIZER COMPARTMENT BLOCKS

The pressurizer compartment has removable blocks in its roof design to allow for access to the upper portions of the pressurizer area. The original plant design provided these blocks to prevent the ejection of missiles from the compartment. The motive force behind these missiles is the energy available to the pressurizer subsystem. The pressurizer compartment is slightly southwest of the "B" steam generator.

Periodically, between refueling outages, access is required to the pressurizer compartment to perform various equipment surveillance inspections. Due to the difficulty in reaching the upper portions of the compartment, it is proposed that the removable blocks be repositioned during these surveillances to allow for personnel access not to exceed 192 hours (8 days) per year.

This temporary change does not increase the probability of an occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the safety analysis. The probability of an initiating event, combined with the short removal time of the removable blocks, is low enough that there is no significant impact in the probability of occurrence and therefore, consequences of an accident or malfunction of equipment.

This temporary change does not create the possibility of an accident or malfunction of a different type than previously evaluated in the safety analysis in that the temporary removal of the blocks does not change or create any new initiating events. In addition, the change does not impact the ability to meet Technical Specification requirements. Missiles are not a new event since SEP Topic III-4.c performed a review of missile effects.

This change does not reduce the margin of safety as defined in the basis for any Technical Specification in that the removable blocks are not within the scope of the Technical Specifications.

Therefore, this change does not involve an unreviewed safety question.

SEV-1015
REMOVAL OF PRESSURIZER COMPARTMENT BLOCKS FOLLOWING REFUELING

The pressurizer compartment has removable blocks in its roof design to allow for access to the upper portions of the pressurizer area. The original plant design provided these blocks to prevent the ejection of missiles from the compartment. The motive force behind these missiles is the energy available to the pressurizer subsystem. The pressurizer compartment is slightly southwest of the "B" steam generator.

During restart after refueling, there is a need to perform inspections in the pressurizer compartment. Due to space limitations, it is necessary to reposition one of the removable blocks to allow for access by plant personnel. It is proposed to leave this block out of position until inspections are completed prior to criticality, reducing boron concentration or moving a control rod from the full insert position.

This change does not increase the probability of an occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated because the removal of the pressurizer blocks do not provide an initiating mechanism and the radiological consequences of a small LOCA have been analyzed. The blocks do not interact with any equipment important to safety therefore no increase in malfunctions are created.

This change does not create the possibility of an accident or malfunction of a different type than previously evaluated in the safety analysis because the consequences of the event are bounded by existing MSLB analysis. Missiles are not a new event, since SEP Topic III-4.c performed a review of missile effects for Ginna.

This change does not reduce the margin of safety as defined in the basis for any Technical Specification in that the removable blocks are not within the scope of the Technical Specifications and do not impact any conditions required to meet the Technical Specifications.

Therefore, this change does not involve an unreviewed safety question.

This report discusses the use of BWI replacement steam generators (RSGs) at the R.E. Ginna Nuclear Power Plant (Ginna). It demonstrates that the RSGs meet the existing UFSAR criteria and can be operated in accordance with Nuclear Regulatory Commission (NRC) requirements set forth in 10CFR50.59, without prior NRC approval.

The RSGs are manufactured by Babcock & Wilcox International (BWI) in Cambridge, Ontario, Canada. The RSGs are designed, manufactured and tested in accordance with the 1986 Edition of Section III of the ASME Code and will be N-stamped by BWI prior to shipment to Ginna. The design, procurement, and manufacturing process is performed under a Quality Assurance Program that complies with the requirements of Appendix B to 10CFR50 and with the current NRC requirements that relate to steam generator design.

The RSGs occupy the same physical envelope as the Original Steam Generators (OSGs). There are no changes to interfaces with the reactor coolant, main feedwater or main steam systems, or to major component supports or piping supports. Differences between the OSG and RSG designs include (1) a small weight increase, (2) addition of an integral flow restrictor in the main steam nozzle, (3) increased heat transfer area, (4) use of Alloy 690 tube material, (5) reduced tube diameter and wall thickness, (6) slightly increased primary side water inventory and (7) slightly increased full-power secondary side water inventory. Evaluations of the differences between the RSGs and OSGs are presented in this report. These evaluations confirm that use of the RSGs meets the existing UFSAR licensing acceptance criteria, does not require a change to the plant Technical Specifications or their bases, and does not create an Unreviewed Safety Question.

The three criteria identified in 10CFR50.59 have been factored into seven questions. These are addressed in Subsections 1 through 7. Assessment of each question as it pertains to use of the RSGs demonstrates that use of the RSGs does not create an unreviewed safety question.

1) Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the UFSAR?

This question is addressed in three steps: First, the accidents and transients evaluated in the UFSAR for which the probability of occurrence could be affected by use of the RSGs are identified. Second, the differences between the RSGs and the OSGs that could affect the probability of occurrence of an accident or transient are identified. Third, the identified differences are evaluated to determine if the probability of an accident or transient is increased by use of the RSGs. These steps are addressed below.

Step 1 - Identification of Accidents or Transients for Which the Probability of Occurrence is Potentially Affected by Use of the RSGs

Most of the accidents and transients evaluated in the UFSAR are initiated by failures or inadvertent actuations of equipment and systems that are not related to use of the RSGs. All accidents and transients presented in the UFSAR were evaluated with the RSGs in place of the OSGs to identify any for which the probability of occurrence could be affected by use of the RSGs. The Steam Generator Tube Rupture (SGTR) was identified as the only accident or transient for which the probability of occurrence could be affected by use of the RSGs.

Step 2 - Identification of RSG Differences That Could Affect Accident Probability

Differences between the RSG and OSGs that could affect the probability of occurrence of an SGTR are the RSG tube bundle design, tube support system, and tube-to-tubesheet joint design. These differences are evaluated in Step 3.

Step 3 - Evaluation of Differences

The RSG and OSG tube bundle configurations are similar, however the RSG tubes are smaller in diameter, constructed of Alloy 690 instead of Alloy 600, and supported by stainless steel lattice grids instead of the drilled carbon steel plates used in the OSGs. The RSG upper tube bundle shape consists of tubes with continuous, smooth, long-radius bends. The U-bends of the innermost RSG tubes are skewed to provide a longer minimum bend radius than those in the OSG design. The RSG tube bundle configuration is used to evaluate the structural and vibrational adequacy of the RSG tubes. The tube bundle design is analyzed to determine tube vibration characteristics and the effectiveness of lattice grids in suppressing vibration. The results verify that lattice grids are superior for strength and vibration restraint. This minimizes wear due to fretting. Therefore, the RSG tubes and tube bundle configuration are as capable of controlling harmful modes of vibration as are the OSG tubes and tube bundle configuration, and the RSG tube bundle configuration does not increase the probability of an SGTR.

Average tube inside and outside diameters are smaller for the RSG than for the OSG, and the RSG tube wall is thinner than that of the OSG. Nevertheless, the Alloy 690 material used in the RSG tubes is stronger than the Alloy 600 tubes used in the OSGs, the RSG tubes are nominally 0.007 inches thinner than the OSG tubes (consistent with the smaller inside diameter of the RSG tubes). The RSG tube material (Alloy 690) has a higher allowable stress

than does the Alloy 600 tubing used in the OSGs. Because the critical rupture pressure of the RSG tubes exceeds that of the OSG tubes, the reduced RSG tube wall thickness does not increase the probability of an SGTR.

Additionally, the RSG tube material is more resistant to primary and secondary side corrosion and cracking than is the OSG tube material. Factors that contribute to increased corrosion resistance are higher chromium content and higher grain boundary carbide decoration. Because the RSG tube material resists corrosion better than the OSG tube material, the difference in tube materials does not increase the probability of an SGTR.

The RSG tubes are joined to the tubesheets by welding and hydraulic expansion. The RSG tubes are hydraulically expanded through the full depth of the tubesheet, whereas the OSG tubes are not. Elimination of the tubesheet crevice in the RSG protects against secondary side stress-corrosion cracking in this area. Therefore, differences in the OSG and RSG tube-to-tubesheet joint designs do not increase the probability of an SGTR.

The paragraphs above describe review of the accidents and transients evaluated in the UFSAR and identify the SGTR as the only accident or transient for which probability of occurrence could be affected by use of the RSGs. The differences between the RSG and OSG that could affect the probability of occurrence of an SGTR are identified, and each is evaluated to show that use of the RSGs does not increase the probability of occurrence of this event. Because the differences between the RSG and OSG tube bundle design that could affect the probability of an SGTR have been identified and shown not to increase the probability of an SGTR, use of the RSGs does not increase the probability of an SGTR and use of the RSGs does not increase the probability of occurrence of accidents or transients previously evaluated in the UFSAR.

2) Does the proposed activity increase the consequences of an accident previously evaluated in the UFSAR?

Consistent with NSAC-125, consequences of an accident are considered to be dose consequences. This question is addressed in three steps: First, accidents and transients previously evaluated in the UFSAR for which dose consequences could be affected by use of the RSGs are identified. Second, the differences between the RSGs and OSGs that could affect the consequences of these accidents and transients are identified. Third, the RSG and OSG differences are evaluated to determine if the consequences of these accidents and transients are increased by use of the RSGs. These steps are addressed below.

Step 1 - Identification of Accident for Which the Consequences are Potentially Affected by Use of the RSGs

Dose consequences from LOCAs are calculated independent of any steam generator parameters, and are therefore not potentially effected by the RSGs.

Step 2 - Identification of RSG Difference That Could Affect Accident Dose Consequences

The parameters important to offsite does consequences are (1) the integrated leakage into the secondary system and (2) the RCS activity concentration. The integrated leakage into the secondary system is a function of the break area. The RCS activity level used in the UFSAR calculations corresponds to the Technical Specification limit, which is not affected by steam generator replacement. Therefore, the only difference between the RSGs and OSGs that could affect SGTR does consequences is the RSG tube inside diameter. The effect of this difference on the SGTR dose consequences is evaluated below.

Step 3 - Evaluation of Difference

The nominal RSG tube inside diameter is 0.111 inches smaller than the nominal OSG tube inside diameter. Because the RSG broken tube flow area is smaller, the break flow will be lower. However this lower flow will cause a slower depressurization of the RCS, and consequently a later reactor trip. The combined effect of these differences produces approximately the same integrated leakage, and that the existing Ginna calculation bounds the Ginna plant with the RSGs with respect to offsite dose.

Because the dose consequences for the SGTR (the only UFSAR accident that could be affected by use of the RSGs) are shown to be bounded by the existing calculation, use of the RSGs does not increase the consequences of any accident previously evaluated in the UFSAR.

3) Does the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR?

This question is addressed in three steps: First, the differences between the RSGs and OSGs that could affect malfunction of the equipment assumed to function in the accident analysis are identified. Second, the equipment that could be affected by these RSG differences is identified. Third, the effect of the differences identified on the probability of a malfunction of the equipment identified is evaluated. For this evaluation, meeting the existing structural acceptance criteria is considered to demonstrate that the probability of a structural malfunction is not increased. These steps are addressed below.

Step 1 - Identification of RSG Differences That Could Affect Equipment Malfunction

The probability of a malfunction of the equipment and protection features that are assumed to function in the UFSAR accident analyses could be affected if the equipment or protection features are required to operate outside their design conditions. The RSGs are designed to the same temperatures and pressures as the OSGs. The only RSG differences identified that could change the operating conditions of equipment or protection features assumed to function in the UFSAR accident analyses are the RSG weight and center of gravity. The RSGs are approximately six percent heavier than the OSGs and their centers of gravity at operating conditions are approximately 12 inches higher than those of the OSGs.

Step 2 - Identification of Equipment Affected by RSG Differences

The loads imposed on the RSG supports and attached piping are slightly higher than for the OSGs because the RSGs are slightly heavier and have a higher center of gravity than the OSGs. Evaluation of the increased RSG loads on supports and attached piping is discussed in Step 3 below.

Step 3 - Evaluation of Differences

The effects of a six percent increase in weight and a 12-inch higher center of gravity for the RSGs are evaluated. The evaluation shows that the existing supports and attached piping loads meet the UFSAR acceptance criteria with the RSGs in place of the OSGs. The component, piping and support loads and stresses remain below the existing allowable values. Because the existing allowable loads are shown not to increase beyond the acceptance criteria with the RSGs in place of the OSGs, the probability of a malfunction of this equipment is considered to be equivalent for the RSGs and OSGs. Therefore, use of the RSGs does not increase the probability of a malfunction of component supports or attached piping.

The above paragraphs identify increased RSG weight and higher center of gravity as the only aspects of use of the RSGs that could affect the probability of a malfunction of equipment important to safety. Steam generator supports and attached piping are identified as the only equipment that could be affected. The evaluation shows that the increase in weight and higher center of gravity do not affect the probability of malfunction of this equipment. Therefore, use of the RSGs does not increase the probability of a malfunction of equipment important to safety previously analyzed in the UFSAR.

4) Does the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR?

This question is addressed in three steps: First, the differences between the RSGs and OSGs that could affect the consequences of a malfunction of the equipment assumed to function in the UFSAR accident analysis are identified. Second, the equipment affected is identified. Third, the effect of the differences on the consequences is evaluated. These steps are presented below.

<u>Step 1</u> - Identification of RSG Differences That Could Affect Equipment Malfunction Consequences

The RSGs could affect the consequences of a malfunction of equipment by (a) affecting the sequence of events or thermal-hydraulic response of an accident, by (b) affecting assumed operator actions, or by (c) causing operation of systems important to safety outside their operating limits. Differences between the RSG and OSG that potentially increase the consequences of a malfunction of equipment assumed to function in the UFSAR are addressed in turn below.

a. The RSGs are designed for the same conditions as the OSGs. The acceptance criteria for the accidents and transients evaluated in the UFSAR are met with the RSGs. Because the design conditions for the RSGs and OSGs are equivalent; and because the acceptance criteria for the accidents and transients evaluated in the UFSAR are met, the equipment assumed to function in the UFSAR accident analysis functions within its operating limits with the RSGs in place of the OSGs. Therefore, no RSG differences are identified that affect the consequences of a malfunction of equipment assumed to function in the UFSAR.

b. The operator actions assumed in response to the accidents and transients analyzed in the UFSAR are documented in the plant emergency operating procedures. Evaluation of plant emergency operating procedures, identified no required changes. Therefore, the actions prescribed in these procedures, and the operating

setpoints remain appropriate for the RSGs. Because use of the RSGs does not affect the emergency operating procedures, no RSG differences are identified that affect the consequences of a malfunction of equipment assumed to function in the UFSAR.

c. Review of the plant Technical Specifications with respect to the accident analyses presented in the UFSAR, showed that the operating limits used for the OSGs are not changed for the RSGs. Because use of the RSGs does not change the operating limits of systems important to safety, no RSG differences are identified that affect the consequences of a malfunction of equipment assumed to function in the UFSAR.

The paragraph above shows that use of the RSGs does not (a) exceed any acceptance criteria for the accidents and transients evaluated in the UFSAR, (b) affect the emergency operating procedures, or (c) cause operation of systems important to safety outside their operating limits. Therefore, there are no RSG differences that affect the consequences of a malfunction of equipment assumed to function in the UFSAR.

<u>Step 2</u> - Equipment Affected by Differences

Since there are no differences identified that affect the consequences of a malfunction of equipment assumed to function in the UFSAR, no equipment is affected.

<u>Step 3</u> - Evaluation of Differences

Because there are no RSG differences identified that affect the consequences of a malfunction of equipment assumed to function in the UFSAR, and because no affected equipment is identified, there are no differences to evaluate.

Because there are no differences between the RSGs and OSGs that could affect the consequences of a malfunction of equipment assumed to function in the UFSAR, use of the RSGs does not increase the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR.

5) Does the proposed activity create the possibility of an accident of a different type than any previously evaluated in the UFSAR?

This question is addressed in two steps: First, the differences between the RSGs and OSGs that could have the potential to create an accident of a different type than any previously evaluated in the UFSAR are identified. Second, any differences identified in the first step are evaluated for their potential to create an accident of a different type than any previously evaluated in the UFSAR.

Step 1 - Identification of RSG Differences That Could Create a Different Type of Accident

The RSGs connect to the same piping (except for minor small-bore piping relocations), instrument connections, and supports as do the OSGs. Because the RSGs connect to and use the same piping, instrument connections, and supports as do the OSGs, no differences are created that could result in the possibility of an accident of a difference type than any previously evaluated in the UFSAR.

Step 2 - Evaluation of Differences

Because use of the RSGs does not create differences that could result in the possibility of a different type of accident, there are no differences to evaluate.

Because use of the RSGs does not create any differences that could result in the possibility of an accident of a different type, their use does not result in the possibility of an accident of a different type than types already evaluated in the UFSAR.

6) Does the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated int he UFSAR?

This question is addressed in two steps: First, the differences between the RSGs and OSGs that could have the potential to create a malfunction of a different type than any previously evaluated in the UFSAR are identified. Second, the differences identified in the first step are evaluated for their potential to create a malfunction of equipment important to safety of a different type than any previously evaluated in the UFSAR.

Step 1 - Identification of RSG Differences That Could Create a Different Type of Malfunction

The RSGs connect to the same piping (except for minor small-bore piping relocations), instrument connections, and supports as do the OSGs. Because the RSGs connect to and use the same piping, instrumentation, and supports as do the OSGs, no differences are created that could result in the possibility of a malfunction of a different type than any previously evaluated in the UFSAR.

Step 2 - Evaluation of Differences

Because use of the RSGs does not create any differences that could result in the possibility of a different type of malfunction, there are no differences to evaluate.

Therefore use of the RSGs does not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the UFSAR.

7) Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specifications?

This question is addressed in four steps: First, a review of the UFSAR accident analyses, structural analyses, and other issues identifies areas where use of the RSGs that could reduce the margin of safety as defined in the Technical Specifications. (Reduction in the margin of safety is defined as the value of any acceptance criteria parameter exceeding the value that is reported in the UFSAR or modified by the NRC's SER.) Second, the differences between the RSGs and OSGs that could affect each area identified in the first step are identified. Third, each difference is evaluated to determine its effect on the margin of safety. Fourth, the Technical Specifications are reviewed to identify any changes that are required by use of the RSGs, and to assess the effect of any changes on the margin of safety. These steps are addressed below.

<u>Step 1</u> - Identification of Areas Where Use of the RSGs Could Affect the Margin of Safety

The results of the UFSAR accident analysis evaluation showed that use of the RSGs was bounded by the UFSAR accident analyses for all events except items 1, 2 and 3 below.

Evaluation of the containment pressure response with the RSGs in place of the OSGs, showed that use of the RSGs is bounded by the existing UFSAR calculations except for item 4 below.

Evaluation of the Ginna structural analyses with the RSGs in place of the OSGs, identified item 5 below as the only structural aspect of use of the RSGs that could affect the plant structural analyses.

Evaluation of other issues with the RSGs in place of the OSGs, identified items 6, 7 and 8 below as potentially affected by use of the RSGs.

1. Combined Steam Generator Relief Valve and Feedwater Control Valve Failure.

2. Loss of External Electric Load.

3. Steam Generator Tube Rupture.

4. LBLOCA containment response.

5. RSG supports and attached piping loads and stresses.

6. Overpressure protection during normal operation.

7. Low temperature over-pressure.

8. ATWS

Step 2 - Identification of RSG Differences

The RSG differences that could affect the Items identified in Step 1 are shown in corresponding Items 1 through 8 below.

1. Combined Steam Generator Relief Valve and Feedwater Control Valve Failure identifies secondary side operating pressure (that slightly reduces generator over-feeding and increases Atmospheric dump valve flow) as an aspect of use of the RSG that could affect this event.

2. Loss of External Electric Load identifies steam production rate and the pressure drop from the downcomer to the steam exit nozzle as aspects of the RSG that could affect the peak secondary side operating pressure. Initial secondary side pressure is an aspect of the RSG that could affect the peak primary side pressure.

3. Steam Generator Tube Rupture identifies integrated steam generator leakage and RCS activity level as parameters that could affect SGTR dose consequences. The only difference between the RSGs and OSGs that could affect the integrated steam generator leakage is the tube break area. RCS activity level is based on the Technical Specification limit, and therefore is not affected by use of the RSGs. Inside tube diameter and secondary side volume are aspects of the RSG that could affect overfill.

4. LBLOCA containment pressure response identifies primary side volume and heat transfer area as aspects of the RSG that could affect peak containment pressure.

5. RSG supports and attached piping loads and stresses identifies weight and center of gravity as aspects of the RSG that could affect RSG support and attached piping loads.

6. Overpressure protection during normal operation identifies heat transfer area as an aspect of the RSG that could affect peak RCS pressure and peak main steam pressure.

7. Low temperature over-pressure identifies secondary side liquid mass and primary side volume as aspects of the RSG that could affect these results.

8. Results of WCAP-8404 identifies secondary side liquid mass and primary side volume as aspects of the RSG that could affect these results.

Step 3 - Evaluation of Differences

The differences identified in Step 2 are evaluated below for their effect on the margin of safety for the corresponding areas identified in Step 1.

1. Combined Steam Generator Relief Valve and Feedwater Control Valve Failure

Because this event was not shown to be bounded by evaluation, it was analyzed with the RSGs in place of the OSGs. The analysis showed that the higher RSG secondary side operating pressure causes slightly less feedwater to reach the RSG than reaches the OSG during this event. This reduces (improves) plant response to this event, and the UFSAR analysis bounds use of the RSGs. Therefore use of the RSGs does not reduce the margin of safety.

2. Loss of External Electric Load

Because this event was not shown to be bounded by evaluation, it was analyzed with the RSGs in place of the OSGs. The analysis shows that primary and secondary side pressures, and minimum DNBR remain within the acceptance criteria for this event with the RSGs. Therefore use of the RSGs does not reduce the margin of safety.

3. Steam Generator Tube Rupture (SGTR)

Because the SGTR was not shown to be bounded by evaluation, it was analyzed with the RSGs in place of the OSGs. The analysis discusses two aspects of the SGTR. These are offsite dose and steam generator overfill. The analysis showed that the UFSAR analyses bound the RSGs for dose consequences and for steam generator overfill. Therefore use of the RSGs does not reduce the margin of safety.

4. LBLOCA containment response

Analysis of the LBLOCA shows that the peak containment pressure following a LBLOCA would be slightly higher with the RSGs than with the OSGs. However, the peak containment pressure would not exceed the containment design pressure (the acceptance criteria for the LBLOCA event). Therefore use of the RSGs does not reduce the margin of safety.

5. RSG supports and attached piping loads and stresses

   An analysis discusses the structural analysis of the Ginna plant with the RSGs in place of the OSGs. It shows that the UFSAR acceptance criteria are met with the increased RSG weight and higher center of gravity. Therefore use of the RSGs does not reduce the margin of safety.

6. Overpressure protection during normal operation

   Analysis of this event shows that the calculated primary and secondary side pressures meet the acceptance criteria with the RSGs in place of the OSGs. Therefore use of the RSGs does not reduce the margin of safety.

7. Low Temperature Over-Pressure

   Analysis of this event shows that the calculated primary and secondary side pressures in the calculation of record meet the acceptance criteria with the RSGs in place of the OSGs. Therefore use of the RSGs does not reduce the margin of safety.

8. ATWS

   Evaluations of the effects of use of the RSGs on the ATWS events described in WCAP-8404 were evaluated. These evaluations show that use of the RSGs does not alter the conclusions of WCAP-8404. Therefore use of the RSGs does not reduce the margin of safety.

Evaluation of the UFSAR accident analyses shows that all existing UFSAR acceptance criteria are met with the RSGs. The results of these evaluations are documented. Therefore, use of the RSGs does not reduce the margins of safety in the UFSAR accident analyses. Evaluation of the use of the RSGs on the peak containment pressure that results from an LBLOCA or MSLB is evaluated and shown to be below the existing design pressure. Therefore, use of the RSGs does not reduce the margins of safety in the peak containment pressure analyses. Increased support loads and stresses that result form use of the RSGs are shown to remain below the existing acceptance criteria. Therefore, use of the RSGs does not reduce the margins of safety in the UFSAR structural evaluations. Other issues that could be affected by use of the RSGs are evaluated. The existing acceptance criteria are shown to be met with the RSGs in place of the OSGs. Therefore, use of the RSGs does not reduce the margin of safety with regard to the issues identified.

Step 4 - Evaluation of Technical Specifications and Bases

Review of the Ginna Technical Specifications showed that there are no changes required to the Technical Specifications by use of the RSGs. Therefore, the existing Technical Specifications remain applicable for the RSGs. The results of this review are documented. Therefore, the margin of safety in the Technical Specifications is not reduced by use of the RSGs.

Because the acceptance criteria in the UFSAR accident and structural analyses are met with the RSGs in place of the OSGs, because the acceptance criteria for the other issues identified are met with the RSGs in place of the OSGs, and because changes to the Technical Specifications are not required, it is concluded that use of the RSGs does not reduce the margin of safety defined in the basis for any Technical Specifications.

Old Steam Generator Storage Facility

The Old Steam Generator Storage Facility (OSGSF) is a reinforced concrete building which will provide long-term onsite storage of the two old steam generators and the attached insulation material. The design life of the OSGSF is 40 years and the facility is located outside the existing security perimeter fence. The OSGSF is constructed of a 21-inch reinforced concrete roof with 3-inch metal decking and single span steel beams supported on 30-inch reinforced concrete walls with a strip footing foundation. The old steam generators and saddle beams will be supported by reinforced concrete piers on spread footings. A floating slab will be used for the floor area of the building. Grading will be performed to direct surface water run-off away from the protected area.

The OSGSF is a stand alone facility and will have no interface with permanent plant structures. The OSGSF will interface with the old steam generators and their supports. Furthermore, the OSGSF will be provided power from the non-safety related electrical power distribution system.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

The OSGSF is classified as a nonsafety-related, non-seismic structure and is not physically connected or immediately adjacent to an existing site structure. The OSGSF interfaces with the old steam generators, their supports and nonsafety-related electrical distribution systems. The OSGSF does not interface with any existing plant instrumentation and control systems. The design of this facility ensures that no safety-related structures, systems, or components are impacted.

Due to their massive weight, a steam generator will not become a tornado missile. Penetration of the concrete resulting from a tornado missile is precluded by the 30 inch reinforced concrete walls. Therefore, the effects of a tornado strike or tornado missiles is not expected to result in a radioactive release.

The consequences resulting from a postulated failure of the OSGSF onto the stored steam generators or a postulated drop of a steam generator from the storage position has been addressed as part of the haul route evaluation presented in safety evaluation SEV-1020 and determined to be acceptable.

Therefore, based on the above, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

The design of the OSGSF ensures that there is no impact to safety-related structures, systems, or components in the event of a design basis seismic event. This facility is designed for limited personnel access and occupancy. Grading of the area adjacent to the OSGSF will be performed to ensure that surface water runoff is directed away from the protected area. Therefore, this activity will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

The OSGSF is a non-seismic, nonsafety-related structure located outside the existing security perimeter fence. This facility is a stand alone facility which is not physically connected or immediately adjacent to any existing site structure. The OSGSF will not affect the integrity of any safe shutdown structures, systems, or components. Therefore, the margin of safety as defined in the Bases section of the Technical Specifications will not be reduced.

<u>Containment Access Facility</u>

The following modifications will be constructed to provide an efficient means of ingress and egress to the plant for the SGR.

A temporary, pre-engineered, Containment Access Facility (CAF) will be erected west of the service building to provide an efficient means of plant access for workers during the SGR and will be removed following the SGR.

A new permanent door, with security electronics, will be installed in the west wall of the intermediate building for access to the existing containment personnel lock.

Temporary walkways will be installed on the service building roof for access to 1) the new door installed in the intermediate building and 2) the containment dome area via door S56 and the stairwell located in the northwest corner of the intermediate building.

Temporary modifications will be made to permanent stairways and landings, and temporary stairways and landings will be installed inside the intermediate building to improve worker ingress and egress to and from the containment personnel lock. These temporary modifications consist of removing the existing up and down stair treads to Elevations 298'-4" and 278'-4" as well as the associated landing, and providing temporary down stair treads only. The existing stairway and landing will be reinstalled upon completion of the steam generator replacement outage.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

The CAF is erected adjacent to nonsafety-related plant structures and is not directly connected to any existing plant structures. The design of the CAF ensures that no safety-related structures, systems, or components are impacted.

The new intermediate building door is not designed to withstand internally or externally generated missiles. However, safety-related equipment and components are sufficiently protected from the effects of internal and external missiles.

The materials utilized in the design of the temporary walkways are similar to other temporary and permanent site structures and are no worse than existing materials which may become missiles. The equipment within the intermediate building and in the vicinity of the new door, which is required for safe shutdown, has been identified and is protected from the effects of missiles.

The existing stairways and landings are seismic category II/I. Modifications to these structures will not adversely affect any safety-related structures, systems, or components in the event of a design basis seismic event. The temporary stairways are located sufficiently away from safety-related equipment such that should a failure of the temporary stairways and landings occur due to a seismic event, no adverse impact to the equipment would occur.

The performance of existing fire protection equipment is not adversely affected by the CAF interface with the existing site fire water system. The CAF fire protection system has been sized to ensure that the demand of the CAF fire protection system is within the capacity of the yard loop system. The CAF sprinkler system will be provided in accordance with NFPA 13.

Modification to the existing intermediate building stairway required to provide improved ingress and egress will restrict access to El. 298'-4". Access to this elevation will remain available via the stairway located in the northeast corner of the intermediate building.

During installation of the new door in the intermediate building to support the steam generator replacement, a temporary cover will be provided over the opening, as required, to minimize any perturbation in the intermediate building ventilation. Thus, no adverse impact to the existing fire detection system will result.

Therefore, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

The design of the CAF, the temporary stairways, and modifications to existing stairways and landings ensures that no safety-related structures, systems, or components are adversely impacted in the event of a design basis seismic event. The installation, use, and removal of the CAF will not adversely affect the underground commodities in the vicinity. All underground commodities beneath the CAF have been evaluated and will not be adversely affected by CAF surcharge loadings. The gas line beneath the CAF will be encapsulated within a guard pipe which will be vented above grade to prevent the buildup of natural gas beneath the CAF in the inadvertent event of a gas line failure. The materials utilized for the temporary walkways are no worse than existing materials which may become missiles. Although the new intermediate door is not designed to withstand internally or externally generated missiles, safety-related equipment and components are sufficiently protected from postulated missiles.

Therefore, the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR is not created.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

The CAF, temporary walkways, stairways, and the intermediate building door are designed to preclude adverse impact to safety-related structures, systems, and components during a seismic event. This ensures that systems required to be operable following a seismic event are unaffected by these modifications. The CAF and walkways can be installed, used, and removed during any mode of plant operation. Therefore, the margin of safety as defined in the Technical Specifications will not be reduced.

CONTAINMENT CONSTRUCTION OPENINGS

This section of the safety evaluation evaluates the design and analysis of the containment dome reinforced concrete and liner plate during and after the steam generator replacement (SGR). This safety evaluation also evaluates the physical designs and construction activities used to remove and re-install the containment dome reinforced concrete, liner plate, and associated components.

Rigging and handling of heavy loads, including the steam generators, liner plate sections, and construction equipment to be mounted on the containment dome, is addressed in either SEV-1020, Steam Generator Rigging and Handling or SEV-1024, Containment Dome Rigging & Handling.

To facilitate the removal and replacement of the steam generators, two construction openings will be cut in the containment vessel dome. The construction opening for steam generator A is located directly over the steam generator. The construction opening for steam generator B consists of a slotted opening originating above the steam generator and connecting to a larger opening located toward the northeast of the containment dome.

The following is a description of the planned construction sequence to create and restore the construction openings in the containment dome. The construction sequence is divided into 5 stages and provides a reference basis for the containment dome structural evaluation approach. For completeness, the major activities associated with each stage are described below. However, the evaluation of activities associated with each stage or portion of a stage may appear later in this document. Where the evaluation of activities occurs in other sections of this document, reference to the appropriate section of the document has been provided.

Prior to shutdown for the steam generator replacement (SGR) outage with the plant in an Operating mode, the following construction activities will be accomplished (Stage 1):

- Temporary structural steel platforms and structures will be installed on top of the containment dome to support the automated hydraulic jackhammers, craft personnel, and miscellaneous construction equipment and materials.
- Preliminary concrete scoring and drilling operations may commence. These preliminary concrete operations are performed to facilitate establishment of the cut line boundaries for the manual and automatic jackhammering activities and will be limited to an approximate depth of eight inches of concrete. Concrete excavation will not occur during this stage. Automated hydraulic jackhammers shall not be used during these operations.

With the plant in the Cold Shutdown or Refueling Condition, with fuel in the reactor vessel and/or in the process of being offloaded to the spent fuel pool, the following construction activities will be accomplished (Stage 2):

- Supplemental drop protection will be added between the containment bridge crane beams. This material will be sized to prevent any material dropped from overhead work from reaching the cavity.

- Concrete excavation operations will commence using the automated hydraulic jackhammers, manual jackhammers, and rock drills. As the rebar is exposed, it will be cut into manageable lengths to facilitate removal. Excavation activities will be completed down to the top of the liner plate. During these plant conditions the liner plate will not be penetrated and its leaktight integrity will be maintained.

- All layers of reinforcing bar located within construction openings "A" and "B" will be cut.

- Interfering commodities on the inside of the dome (i.e., containment spray piping, HVAC ductwork, and painter's trolley rails) will be temporarily removed.

With the plant in a "no-mode" condition with all fuel offloaded to the spent fuel pool and all necessary isolation of the spent fuel pool and supporting containment systems achieved, the following construction activities will be accomplished (Stage 3):

- Concrete excavation operations may still be on-going. Once completed, the steel liner plate will be cut in sections, lifted from the containment dome, and moved to a preparation area onsite to prepare it for re-installation.

- The old steam generators will be removed and the new steam generators will be installed. (This activity is addressed in the Safety Evaluation for DC-10034B, Steam Generator Rigging and Handling.)

- The old liner plate sections or new patch plate sections, if required, will be prepared, rigged into place, welded, leak chase channels installed, NDE performed, and leak-tested.

- The interfering commodities on the inside of the dome will be reinstalled and tested as appropriate.

- The majority of reinforcing bars will be installed and cadwelded. Splicing of rebar by welding may also be performed, if necessary.

With the plant returned to a Refueling or Cold Shutdown Condition, with fuel in the reactor vessel and/or in the process of being onloaded from the spent fuel pool, the following construction activities will be accomplished (Stage 4):

- Installation and testing of the interfering commodities on the interior of the containment dome may still be ongoing.

- The final pieces of reinforcing bar will be set and cadwelded/stick-welded.

- All concrete will be placed/poured and cured. Strength tests will be conducted and acceptable results will be achieved.

- A limited containment structural integrity test (SIT) and an integrated leak rate test (ILRT) will be conducted, and based on acceptable test results, full containment integrity will be proven (Cold Shutdown only).

- The automated hydraulic and manual jackhammers, and any other large construction equipment, will be removed from the containment dome.

With the plant returned to an Operating Condition, the following construction activities will be accomplished (Stage 5):

- The temporary structural steel platforms and structures as well as miscellaneous construction equipment will be removed from the containment dome.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

The plant will be in Cold Shutdown and the primary coolant system depressurized before concrete can be excavated and reinforcing bar can be cut. Removal of the reinforcing steel and concrete down to the liner plate during Cold Shutdown or Refueling will not adversely impact previously reviewed accidents or malfunctions since the liner's leaktight barrier will be maintained in all occurrences where a significant radioactive release may occur. The load associated with the weight of the concrete rubble has been evaluated and does not adversely affect the design requirements of the containment concrete or the liner plate. Analysis has shown that the containment has sufficient capacity to support the construction loads.

Refueling of the reactor will not commence until the containment liner has been restored and local leak tests performed. The plant will not leave the Cold Shutdown condition until the containment dome is completely restored and the limited SIT/ILRT performed.

The functional capability of the containment dome to withstand a tornado missile impact with less than the designed concrete dome thickness in selected areas was evaluated and the probability of a tornado missile striking a section of the excavated concrete, penetrating the liner plate, and damaging safety related components during stages 2 and 4 is not considered credible. During stage 3 there is no provision for tornado missile protection since no equipment within containment is required with the reactor defueled.

Thus, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

The size and location of the containment openings have been selected based on Calculation 22225-C-0402-01 and the safety evaluation associated with Design Criteria DC-10034B (SEV-1020) to preclude any adverse impact on the spent fuel pool as a result of a heavy load drop.

The plant will be in Cold Shutdown and the primary coolant system depressurized before significant concrete is removed and reinforcing bar can be cut.

Breaching of the containment dome liner will not occur until the plant has been defueled, all fuel is safely stored within the spent fuel pool, and all necessary isolation of the spent fuel pool and supporting containment systems achieved.

The containment dome has been analyzed for all applicable loading configurations associated with the containment construction openings, to include the loading associated with the addition of temporary structures and equipment onto the dome, and has been demonstrated to be capable of performing its required functions under all modes of plant operation during which activities are conducted.

Based on the above, this activity will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

The containment construction openings do not involve a change to any technical specifications as defined in Sections 3.6 or 3.8 of the Ginna Technical Specifications. This activity has been reviewed against the design description presented in Section 5.2.1 of the Technical Specifications and has been determined to not impact this description.

The design bases and performance criteria of safety related equipment has been shown to remain consistent with the approved safety analyses.

The temporary configuration of the containment openings to include the additional loadings imposed by the structures and equipment utilized to construct the opening have been evaluated and determined to have no adverse impact on the structural capability of the containment.

All specifications associated with Technical Specifications 3.6, 3.8, and 5.2 have been reviewed and determined to not be impacted by implementation of the construction openings in the plant modes assumed in this safety evaluation.

Therefore, the margin of safety as defined in the Bases section of the Technical Specifications 3.6 and 3.8 will not be reduced.

<u>TEMPORARY PLATFORMS</u>

Various structural steel platforms and frames will be installed on and anchored to the containment dome to support the automated hydraulic jackhammers, craft personnel, and miscellaneous construction equipment and materials. These platforms are designed to facilitate the construction activities planned to be performed, provide a safe means of worker access to the dome openings, and to protect personnel and adjacent equipment from falling debris.

Temporary equipment and material laydown areas will be erected on the facade structure close to the dome openings. A temporary craft break area, stairwell(s), and personnel walkways will also be erected on the facade structure. The function of the temporary craft break area is to provide a break area and lunch room for personnel working on the containment dome.

These temporary structures may be erected prior to plant shutdown and may or may not be removed until after the plant has returned to operation following replacement of the steam generators. The temporary platforms and walkways have been designed to perform their functions under all modes of plant operation. To provide weather protection and help in containing the waste from the concrete removal operation and other construction activities, temporary weather enclosures will be placed over the dome openings. The temporary weather enclosures, walkways, and platforms will be attached to the containment dome and/or the facade structure.

Heavy load handling activities during installation and removal of these temporary structures will be performed in accordance with the requirements specified in the safety evaluations prepared for Design Criteria DC-10034B to ensure heavy loads are maintained within previously reviewed and approved safe load paths.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

The temporary structures are classified as safety significant, Seismic Category II/I. These structures have been evaluated to ensure that they do not fail and impact adjacent safety-related equipment during a seismic event.

The containment dome and facade have been evaluated for the additional loading configurations of the temporary structures and determined to be within the allowable loadings.

Based on the above, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

The containment structure and facade structure have been evaluated for the additional load of the temporary structures and determined to be capable of performing their required functions under all modes of operation. Therefore, it is acceptable for these modifications to be in place during any mode of operation.

The temporary structures located within the facade have been evaluated for either the normal or 132 mph tornado wind speeds, whichever condition governs. Therefore, these structures will remain intact during winds of this magnitude.

Based on the above, this activity will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

The permanent plant structures will not be modified as a result of the installation of the temporary structures. Containment integrity, as defined in Technical Specification 3.8, is not affected by this modification. Restoration will not modify the load carrying characteristics of any of these structures. All operating design conditions will remain unchanged. The containment dome and facade structure will continue to be capable of performing their required safety-related functions.

Therefore, the margin of safety as defined in the Bases section of the Technical Specifications will not be reduced.

<u>PERMANENT STEAM GENERATOR SUPPORTS</u>

The method for disconnection of the old steam generators and reinstallation of the new steam generators is addressed in Design Criteria 10034A and basically includes:

- Lock lower support columns in place with temporary structural restraints

- Disconnect the old steam generators from the upper support (ring) snubbers/struts and lower support columns

- Install the new steam generators, replacement upper supports (rings), and lower support components

- Connect the new steam generators to the support structures, including all shimming necessary to achieve RCS piping fitup.

During the steam generator replacement outage, new upper restraint rings and new portions of the lower support brackets will be installed.

The upper restraint ring will be fabricated in two halves, set in place in the field, and the halves joined by field welding. The new portions of the lower support bracket will be fabricated in two pieces, to allow ease of fit-up during reinstallation.

To allow removal of the old steam generators, the upper restraint ring snubber hydraulic tubing must be severed. Following the new steam generator installation, this tubing must be reattached to the snubbers.

To facilitate removal and reinstallation of the steam generators, new support bracket assemblies which connect the steam generator support foot pads to their respective vertical support columns will be designed, fabricated and installed. The existing one-piece lower support bracket assembly will be replaced with two pieces: an upper bracket assembly and a lower bracket assembly. This arrangement will allow the upper bracket assembly to be bolted and pinned to the steam generator support foot pad and the lower bracket assembly prior to the steam generator being installed in the cubicle. The upper and lower support bracket assemblies will subsequently be bolted together. Upon reinstallation, the lower support will satisfy the current design bases.

All modifications to these components will satisfy the original design requirements.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

New upper supports (rings) and lower steam generator support components will be fabricated and installed in accordance with the current design requirements.

The hydraulic tubing will be restored to satisfy the existing design requirements. Replacement materials, if required, will satisfy the original code requirements and will meet the existing installation specifications.

Based on the above, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

The new upper supports will be a like-for-like replacement with no change in the basic support configuration.

The existing one-piece lower support bracket assembly will be replaced with two pieces: an upper bracket assembly and a lower bracket assembly. All design basis loading conditions have been evaluated and determined to be acceptable. Upon reinstallation, the lower support will satisfy the current design bases.

The hydraulic tubing will be restored to satisfy the existing design requirements. No operational changes occur as a result of the tubing removal and replacement. Therefore, this activity will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

New upper supports (rings) and lower steam generator support components will be fabricated and installed in accordance with the current design. Activities associated with the steam generator supports will not be performed until the steam generators and supports have been declared inoperable. The new upper supports will be like-for-like replacements with no change in the basic support configuration. The existing one-piece lower support bracket assembly will be replaced with two pieces: an upper bracket assembly and a lower bracket assembly. Upon reinstallation, the lower support will satisfy the current design requirements. The activities required to remove and reinstall the hydraulic tubing will not require any changes to the Technical Specifications. Therefore, the margin of safety as defined in the Bases section of the Technical Specifications will not be reduced.

STEAM GENERATOR LOWER SUPPORT TEMPORARY RESTRAINTS

The steam generator lower supports are pinned columns which will require stabilization before being disconnected from the steam generators. The following is a description of the activities performed during the 1995 Refueling Outage and the remaining activities to be performed during the steam generator replacement outage:

- During the 1995 Refueling Outage new pipe clamps were installed on the existing steam generator support columns, and new tube steel members and new connection plates were installed on the existing platforms.

- During the SGR outage, the temporary stabilizing braces will be installed between the column brackets and tube steel. The stabilizing braces may be loosely installed during defueling but the bracing will not be rigidly connected to the column brackets until all fuel has been removed from the reactor and all necessary isolation of the spent fuel pool achieved. The turnbuckles will be tightened before the RCS piping is cut.

- The turnbuckles will be loosened and the stabilizing braces disconnected after 1/3 weld-out RCS pipe welding is completed and before fuel reload.

- The temporary bracing and column brackets will be completely disconnected from the steam generator support columns prior to entry in a mode above Cold Shutdown.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

The column brackets are classified as safety significant, Seismic Category II/I and have been analyzed to ensure they will not fail and adversely impact safety related equipment during a seismic event. This design approach ensures that the installation of brackets will not adversely impact safety-related structures, systems or components. The brackets shall be completely removed and the lower support columns restored to their pre-steam generator replacement configuration prior to entry into a mode above Cold Shutdown.

During the steam generator replacement outage, the temporary restraint bracing will be fully installed/tensioned after all fuel has been removed from the reactor vessel and will be removed before fuel reload. This administrative control ensures that the steam generator lower supports will continue to satisfy all design basis loading configurations.

Based on the above, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

As a result of the administrative controls and design of the brackets to seismic category II/I criteria, this activity will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

The seismic design of the brackets combined with the limits imposed regarding the plant conditions under which final installation of the restraints can be performed ensures that the structural integrity of the reactor coolant system and all design basis loading configurations are maintained. Therefore, the margin of safety as defined in the Bases section of the Technical Specifications will not be reduced.

REACTOR CAVITY DECKING

To provide a general work and construction laydown area inside the containment, a portion of the reactor cavity will be temporarily covered with steel decking. This decking will be installed in pre-fabricated panels after all fuel is removed from the containment, the reactor cavity drained, the reactor internals have been placed within the vessel, the reactor head has been re-set, and necessary isolation of the spent fuel pool and supporting containment systems is achieved. The reactor cavity decking shall be removed prior to fuel reload.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

The reactor cavity decking will be installed once the reactor vessel is completely defueled, the refueling cavity drained, the spent fuel pool isolated, all fuel safely stored within the spent fuel pool, the reactor internals have been placed within the reactor vessel, the reactor head has been re-set, and all safety-related systems required during the defueled condition are isolated from containment. The cavity decking will be removed prior to fuel reload.

All heavy load movements during installation and removal of the cavity decking will meet the requirements of RG&E Administrative Procedure A-1305 to ensure heavy loads are maintained within previously reviewed and approved safe load paths. With the plant in the defueled condition and essential safety-related systems isolated from containment, the impact of a construction incident is minimized.

Based on the above, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

As a result of the analysis performed confirming no impact on permanent plant components and the controls imposed regarding when decking may be installed and removed, this activity will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

As a result of the analysis performed confirming no impact on permanent plant components and the controls imposed regarding when the decking may be installed and removed, the margin of safety as defined in the Bases section of the Technical Specifications will not be reduced.

INTERFERING COMMODITIES

To facilitate the removal and replacement of the steam generators, two openings will be made in the containment vessel dome, one opening over steam generator A and a slotted opening near steam generator B. For these openings to be made, certain commodities must be temporarily removed from the inside surface of the containment dome as follows.

Sections of the containment spray ring piping at elevation 372'-8" will be severed, supports disconnected and temporarily removed from the inside of the containment dome at the construction openings over steam generators A and B to allow a clear rigging path for steam generator removal.

One section of the containment air recirculation system ducting will also be temporarily removed from the inside of the containment dome at the construction opening over steam generator B.

Sections of the painters trolley rail at El. 377'-10" will be temporarily removed at the construction opening over steam generators A and B.

All removed interferences will be reinstalled upon completion of the steam generator replacement to satisfy the current design requirements. Welding and non-destructive examination associated with welding to the containment liner, if required, will be performed in accordance with the Special Processes Manual. Non-destructive examination of the welds of the support plates to the liner plate will be confirmed to not adversely affect the integrity of the containment liner.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

The interfering commodities will be removed and re-installed during the steam generator replacement (SGR) outage. The design bases of the affected equipment will not be altered. All of the affected equipment will be re-installed to satisfy the current design basis.

The containment spray and containment air recirculation systems will not be removed until the plant has reached Cold Shutdown. With the plant in the Cold Shutdown condition, the design basis events for which the containment spray and containment air recirculation systems are required to be operable will not occur. Thus, after reaching Cold Shutdown, the spray system and the containment air recirculation system are no longer required to perform a safety function.

Reinstallation of the air recirculation ductwork and the containment spray piping in accordance with the current design requirements, including the applicable system testing, will be performed prior to entering a mode above Cold Shutdown. This ensures that the systems are available during plant modes in which they may be expected to perform their safety function.

Based on the above, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

While work is conducted in containment, drop protection and administrative controls will be in place to preclude the potential for an accidental load drop impacting the refueling cavity or other equipment that is performing a safety function.

As a result of the controls imposed regarding when the interfering systems can be removed and reinstalled, and since the systems will be reinstalled in accordance with the current design requirements, this activity will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

As a result of the controls imposed regarding when the interfering systems can be removed and reinstalled, the margin of safety as defined in the Bases section of the Technical Specifications will not be reduced. All requirements of Technical Specification Section 3.3.2 will be adhered to during the implementation of this modification.

STEAM GENERATOR ACCESS PLATFORMS

In order to facilitate the removal and reinstallation of the steam generators, portions of the existing steam generator secondary manway access platforms will be temporarily removed and re-installed during the steam generator replacement outage.

In order to facilitate access to the new steam generator secondary manways, permanent secondary manway access platforms will be designed and installed on the steam domes for both the "A" and "B" steam generators.

For the "A" steam generator, a walkway between the existing platform at elevation 310'-0" and the new platform will be provided.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

The new access platforms have been classified as safety significant, Seismic Category II/I and have been analyzed to ensure that the components will not fail and adversely impact safety related equipment during a seismic event. All installation and removal activities conducted with fuel in the containment building will satisfy the requirements of RG&E Administrative Procedure A-1305.1, as applicable, and the additional administrative controls imposed by this safety evaluation. The modifications associated with this design will be performed during the Cold Shutdown mode.

The addition of the access platforms and walkway will not adversely affect the containment spray coverage during a postulated accident. The spray system will continue to cover a maximum portion of the containment free volume. The additional miscellaneous steel added to containment is negligible when compared to the total amount of miscellaneous steel in containment. Therefore, the addition of the platforms will not adversely affect the heat sinks used in the accident analyses.

Modifications to the existing platforms to allow removal and reinstallation will be performed to ensure that the platforms will satisfy the existing design requirements. The existing platforms will be removed by unbolting existing connections or cutting platform members. Any platform members which are cut during disassembly, will be reinstalled using full penetration welds to ensure the structural integrity of the member is not affected. The platforms and walkway will be coated with coatings that satisfy the requirements of CE-125 to ensure that a coating system approved for in-containment usage is applied. Thus, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

As a result of the load handling administrative controls imposed by this safety evaluation, the design of the access platforms and walkway to withstand seismic loadings, and the method of disassembly/reinstallation of the existing platforms, this activity will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

The access platforms do not involve a change to any Ginna Technical Specifications. Adequate precautions in the design and installation/removal of the access platforms have been implemented to minimize adverse impact to structures, systems or components.

The steam generator access platforms are not specifically addressed in the existing station Technical Specifications.

All technical specification requirements applicable during Cold Shutdown and Refueling will be adhered to.

Therefore, the margin of safety as defined in the Technical Specifications will not be reduced.

HAUL ROUTES

To accomplish the steam generator replacement, various heavy loads will be transported around the site. The most significant of these heavy loads are: the Transi-Lift crane, mobile construction cranes, the new and old steam generators, and test loads.

The haul route for the Transi-Lift to and from its working location east of the containment is as follows:

The Transi-Lift will travel from its assembly and load test location in the south section of the main parking lot, proceed north through a removed section of the security fence, turn east, and proceed following the plant road to the working location east of the containment. The Transi-Lift will be returned to the main parking lot for disassembly along the same route but in reverse.

For movement of the Transi-Lift within the protected area along the haul route, the boom of the Transi-Lift is pointed West or South-West and shall not pass over the Contaminated Storage Building, the Standby Auxiliary Feedwater (SAFW) Building, or the Auxiliary Building. The Transi-Lift will be positioned with its rear crawler on the rear crawler foundation with the boom oriented to the southwest.

The Transi-Lift movement into the protected area may occur during either Cold Shutdown, Refueling or Defueled condition.

The haul route for the new steam generators to arrive at the site is as follows:

The transporter will head west on Lake Road from the Bear Creek barge facility, turn north onto the main plant service road, cross the bridge over Deer Creek, turn west into the main parking lot, proceed past the warehouse, and turn north into the warehouse parking lot where the steam generators will be temporarily stored. The transporting of the new steam generators to the temporary storage location may occur during any mode of plant operation.

The haul route for each of the new steam generators from the temporary storage location to the upending location east of the containment is as follows:

The transporter will exit the warehouse parking lot, turn east into the main parking lot, enter the site through the east security gate, turn east, and proceed following the plant road to the upending location. The transportation of the new steam generators from the storage location to the upending location will occur during the defueled mode of operation.

The haul route for each of the old steam generators from the containment to the old steam generator storage facility (OSGSF) is as follows:

From the downending location east of the containment, the transporter will proceed following the plant road, exit the site south through the east security gate, proceed west through the main parking lot, proceed past the warehouse, turn north into the warehouse parking lot, and proceed to the OSGSF located north of the warehouse. The transportation of the old steam generators from the downending location to the OSGSF will occur during the defueled mode of operation.

Various commodities along the haul route require temporary or permanent relocation to allow for passage of the prime mover, transporter, mobile construction cranes and the Transi-Lift. Additionally, protection of underground facilities may be required for those commodities within the influence of expected surcharge loads.

Various improvements along the haul route are required to adequately support heavy load movement. Improvements to the haul route may be performed during any mode of reactor operation.

Prior to assembly of the Transi-lift foundations in the southern most portion of the main parking lot, the reinforced concrete pipes in the laydown/assembly area will be replaced with corrugated metal pipe and the catch basins and handhole will be filled to preclude failure of the structure.

Prior to movement of the Transi-Lift along the haul route, all catch basins, manholes and handholes which will be effected by the Transi-lift surcharge loads will be filled to preclude failure of the structures. All at-grade commodities such as valve boxes and clean-outs will be protected by installing plywood, fill or planking adjacent to the item which will allow the Transi-Lift crawler to span over the item. High mast light tower number 7 will be removed to allow passage of the Transi-Lift. High mast light tower number 8 may be removed, if required, to allow passage of the Transi-lift to pass.

In addition, grading of the haul route will be performed to ensure all leveling requirements for movement of the Transi-Lift are met.

A portion of the Auxiliary Building screen wall and the liquid nitrogen tank foundation will be removed to allow passage of the prime mover, transporter, and steam generators during the Steam Generator Replacement.

The haul route will be load tested utilizing the prime mover and the rear crawler loaded with counterweights. Movement of the haul route test loads along the haul routes may be performed during any mode of reactor operation.

Haul route surfaces will be protected as required using plywood overlays or structural fill.

The existing 10-inch thick concrete slab located over the 34.5 kV ductbank and the 115 kV ductbank under the plant roadway will require modification as a result of the surcharge loadings associated with the Transi-Lift and has been evaluated in Safety Evaluation SEV-1023. The 34.5 kV ductbank and the 115 kV ductbank located under the South parking lot will not be adversely affected by the surcharge loads and therefore does not require modification.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

As described in the above sections, movement of components, including the Transi-Lift and mobile construction cranes, along the haul route have been reviewed and all buried commodities protected, if required, to ensure no adverse impact on safety-related or safety-significant systems or components. Furthermore, the haul route will be load tested prior to movement of the Transi-Lift or the new/old steam generators into the protected area.

UFSAR accidents and transients are not directly impacted by the movement of the Transi-Lift or the mobile construction cranes along the haul route. As a result of the controls imposed upon movements along the haul route, the probability of occurrence of the accidents and transients is not increased beyond that currently assumed. Therefore, these accidents and transients have not been adversely impacted and the mitigating measures described in the UFSAR have not been altered or affected.

The radiological consequences associated with a drop of the steam generator during transport to the old steam generator storage facility has been evaluated and demonstrated to be within the applicable regulatory guidelines and less than the limiting, and more permanent accidents currently evaluated in the UFSAR.

Based on the above, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

Surcharge loadings resulting from the movement of components, including the Transi-Lift and mobile construction cranes, along the haul route have been reviewed to identify the effects on the underground and adjacent commodities. The buried commodities are either unaffected by the impact loadings or temporary protection will be provided to ensure no adverse impact to the buried commodities. Above ground commodities were also reviewed and will be relocated, if necessary, to allow safe passage of components along the haul route. Temporary protective measures, such as temporary fire or security watches, will be invoked, where appropriate, to ensure no adverse reduction in the degree of protection or function of systems or components during the component relocation.

In the unlikely event of a steam generator drop from the transporter during movement along the haul route, a breach of the steam generator primary side could occur. This postulated drop of a steam generator is classified in the same category of accidents as a rupture of a tank containing radioactive material. The Gas Decay Tank Rupture is the limiting event currently evaluated in the UFSAR for accidental gaseous releases. The maximum calculated offsite dose from a postulated drop at the worst location along the haul route was 25.3 mrem whole body. This whole body dose is a small fraction of the 10CFR100 guideline values for accidental releases and the maximum whole body dose of 1.3 rem for the waste gas decay tank rupture.

Based on the above, this activity will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

This activity does not affect compliance with requirements of the Technical Specifications nor will these activities require a change to the Technical Specifications. Therefore, the margin of safety as defined in the Bases section of the Technical Specifications will not be reduced.

TEMPORARY CRANES AND RIGGING COMPONENTS

A Transi-Lift LTL-1200, Series 2A mobile crane with a 340-foot boom and a 120-foot stinger will be used to handle the new and old steam generators during the steam generator replacement. The Transi-Lift will be assembled and load tested in the main parking lot and then moved along an evaluated haul route to a working location east of the containment.

Foundations will be installed east of the containment to support the front and rear crawlers of the Transi-Lift. The installation of these foundations has been evaluated in SEV-1023.

Miscellaneous small hoists will be utilized to rig loads to the containment dome prior to cold shutdown. These rigging activities are addressed in SEV-1024.

Mobile construction cranes will be used prior to, during, and following the SGR outage. These cranes may be used to:

- Assemble and disassemble the Transi-Lift in the main parking lot (may be performed in all modes of plant operation).
- Assemble the Transi-Lift counterweight on the rear crawler for the haul route load tests (may be performed in all modes of plant operation).
- Stage and place counterweights for the foundation load test (may be performed in all modes of plant operation).

- Place counterweights on the Transi-Lift rear crawler when the Transi-Lift is in the steam generator rigging position (will be performed in the defueled mode of plant operation).

- Assemble the pivot stand in the alleyway (will be performed in the defueled mode of plant operation).

A self-leveling, multi-wheeled transporter and prime mover will be used to transport the new and old steam generators to and from the downending/upending location (will be performed in the defueled mode of plant operation).

Various rigging components will be used to attach the Transi-Lift to the old and new steam generators. These components will include a spreader beam, link assemblies, lifting trunnions, main steam nozzle plug (for the old steam generators), link plates, and associated pins and bolts.

Underground and adjacent commodities must be protected or relocated to allow for the installation, use, and removal of the temporary cranes and rigging components. This section evaluates the affect on the underground and adjacent commodities as a result of a postulated load drop/crane failure during the utilization of the temporary cranes and rigging activities in their installed positions.

A pivot stand will be used to upend the new steam generators and to downend the old steam generators. The pivot stand will be located in the alleyway east of the containment and positioned such that the stand is aligned with the Transi-Lift to allow downending/upending of the steam generators as well as being accessible to the site transporter.

The activities associated with the temporary cranes and rigging will be conducted in the following plant modes of operation.

Load testing of the Transi-Lift will be performed in the main parking lot. The Transi-Lift load test is anticipated to be performed with the plant in operation but may be performed during any mode of reactor operation.

Rigging of the steam generators to and from the containment may only be performed with the reactor vessel completely defueled, all necessary isolation of the spent fuel pool and supporting containment systems achieved, and the temporary spent fuel pool cooling system operable.

The pivot stand shall be installed, utilized and removed once the reactor vessel is completely defueled, all necessary isolation of the spent fuel pool and supporting containment systems is achieved, and the temporary spent fuel pool cooling system is operable.

The load test of the Transi-Lift foundations is anticipated to be performed with the plant in operation but may be performed during any mode of operation.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

These lifting and handling activities are performed with temporary cranes which have no direct interface with permanent plant systems or components. The design, material, and construction standards applicable to permanent plant SSCs are unaffected by the temporary installation of the Steam Generator Replacement cranes and associated heavy load operations. The cranes and rigging utilized for the one-piece steam generator replacement are engineered and certified in accordance with appropriate standards which ensure that acceptable factors of safety are utilized in the design of the cranes and provides reasonable assurance that the cranes will safely perform their intended function. The temporary cranes associated with the Steam Generator Replacement do not interact with the function of plant systems and do not affect the environment in which the systems operate. The defense-in-depth philosophy described in this safety evaluation, which includes engineered lifts, identification of safe load paths, localized protection of key shutdown safety functions, procedural controls of lifting activities as well as backup spent fuel pool cooling ensures the continuity of key shutdown safety functions. Furthermore, the lifting equipment used to accomplish the one-piece steam generator removal has been successfully utilized for numerous heavy load lifts, with many of these lifts at a lifted load in excess of the weight of a steam generator and associated handling equipment.

The temporary installation of the cranes and the associated heavy load handling operations do not change, degrade, or prevent actions described or assumed in any accident discussed in the UFSAR. Safe load paths prevent the movement of loads over required plant SSCs, with the exception of the Containment following reactor shutdown and cooldown.

As described in UFSAR Section 15.7.1, the rupture of the gas decay tank assumes the instantaneous release of the entire contents directly to the atmosphere at ground level. The steam generator drop at the worst location along the haul route results in the release of a Curie inventory significantly less than that postulated for a gas decay tank rupture. To determine the consequences of a steam generator drop at the worst location along the haul route, a new atmospheric dispersion factor was calculated. To calculate the offsite dose associated with a steam generator drop at the worst location along the haul route, this calculated accident dispersion factor was utilized to maximize the offsite dose. Utilizing this calculated accident dispersion factor, in the unlikely event of a steam generator drop, the estimated radiological consequences have been determined and are enveloped by the consequences of a gas decay tank rupture described in UFSAR Section 15.7.1, which is the highest offsite dose for a failure of any radwaste system currently analyzed in the UFSAR.

Although the gas decay tanks are not within the conservative zone-of-influence, in the further unlikely event that the dropped steam generator results in a gas decay tank rupture (in spite of intervening concrete walls and floors) concurrent with a breach of the steam generator integrity, the combined radiological consequences remain a small fraction of and well within the guidelines of 10CFR100. As specified in the Safety Evaluation prepared by the Division of Reactor Licensing, the AEC utilized the criteria of "well within 10 CFR Part 100 guidelines" to establish the acceptance limit for the waste gas decay tank failure. Thus, the offsite consequences of a combined release due to a breached steam generator concurrent with the failure of a gas decay tank is within the established acceptance limit for the gas decay tank rupture and, in fact, is significantly less than the 10 CFR Part 100 guidelines. Therefore, the lifting and handling activities associated with the one piece steam generator replacement will not increase the probability of occurrence of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR and identified above.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

The applicable accident types previously evaluated in the UFSAR are heavy load handling and liquid and gas waste system failures, including tank ruptures. The potential effects of a load drop are evaluated using the guidelines of NUREG-0612, consistent with the approach documented in the UFSAR for other heavy loads with the potential for interacting with the spent fuel or associated SSCs. Safe load paths are defined for crane movement and load handling that preclude heavy loads being moved over plant SSCs performing key shutdown safety functions. Appropriate plant restrictions are specified and backup systems are provided so that spent fuel pool cooling, inventory control, and reactivity control are maintained.

The potential radiological release from a dropped steam generator is evaluated in the same manner as the rupture of any other tank containing radioactive material and the consequences are enveloped by the failure of a gas decay tank. Even if the drop of a steam generator should cause the failure of a gas decay tank, the combined radiological consequences are well within, and are in fact, a small fraction of the 10CFR100 guidelines.

The proposed action does not add, delete, or modify any existing SSCs required to perform a key shutdown safety function. The ambient environment in which plant SSCs operate is not affected by the temporary installation of the cranes. Therefore, the lifting and handling activities associated with steam generator replacement will not create a possibility for an accident of a different type than any evaluated previously in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

The lifting and handling activities associated with steam generator replacement will not reduce any margin of safety as defined in the basis of any Technical Specifications. The installation and use of the temporary cranes does not adversely affect the operation of any safety related SSCs. Safe load paths are defined for the lifts to be made by the cranes that preclude loads being moved over required safety related components without first being analyzed or by providing adequate drop protection. The steam generators will only be lifted while the reactor is shut down and defueled. The provision of a temporary spent fuel pool cooling system satisfies the bases for Technical Specification 3.11.4. The proposed action does not affect any instrument accuracies or trip setpoints. Therefore, the margin of safety is maintained.

STEAM GENERATOR SUPPORT SADDLES

Support saddles will be used with the site transporter. These saddles will be used during transport operations and for the long-term storage of the old steam generators in the old steam generator storage facility.

Use of the steam generator support saddles may occur during any mode of reactor operation.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

The steam generator support saddles are classified as non-safety related, non-seismic and have been located to ensure their failure will not adversely impact safety-related structures.

The drop of an old steam generator from the support saddle, which results in a breach of the old steam generators, does not result in doses which exceed the dose limits of 10CFR100 and Section C.1.p of Regulatory Guide 1.29. This calculated dose is also less than the doses calculated for previously evaluated accidents in the UFSAR. Furthermore, due to their massive weight, the steam generators will not become tornado missiles.

Based on the above, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

This activity will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

No direct impact to Technical Specification systems occur as a result of the installation and use of the steam generator support saddles. All Technical Specifications will be adhered to during installation and use of the support saddles. Therefore, the margin of safety as defined in the Bases section of the Technical Specifications will not be reduced.

CONTAINMENT AUXILIARY CRANE

To facilitate the rigging of miscellaneous loads during the steam generator replacement outage, a temporary auxiliary crane (Allied Marine Systems S/N 1663 or 1664) will be installed on the containment floor at elevation 278'-4" adjacent to the northeast corner of the reactor cavity. The auxiliary crane and support tower will be installed, used, and removed during the steam generator replacement outage.

Anchor bolts for the support tower were installed during the 1995 refueling outage. The anchor bolts were evaluated in Safety Evaluation SEV-1022.

During the 1996 SGR outage, the auxiliary crane base and tower will be installed, used and removed during the Cold Shutdown, Refueling or defueled conditions.

Power for the auxiliary crane will be supplied from a temporary power distribution system installed inside containment for the steam generator replacement activities. This temporary power distribution system is addressed in the Safety Evaluation for DC-10034D, Temporary Utilities & Services.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

The auxiliary crane and support tower are classified as safety significant, seismic category II/I and have been analyzed to ensure that the crane and tower will not fail and adversely impact safety related equipment during a design basis seismic event. All installation activities conducted with fuel in the containment building will satisfy the requirements of RG&E Administrative Procedure A-1301.

The auxiliary crane and support tower will be installed, used, and removed during the steam generator replacement outage. Because the auxiliary crane and support tower are only installed, used and removed during the refueling outage in Cold Shutdown, Refueling, or defueled conditions the coatings are classified as non-safety related.

Therefore, the installation, use, and removal of the auxiliary crane and support tower will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

As a result of the load handling administrative controls imposed by RG&E Administrative Procedure A-1305.1 and design of the auxiliary crane and support tower to seismic category II/I criteria, this activity will not create the possibility of an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

The installation, use, and removal of the temporary auxiliary crane and support tower does not involve a change to any Ginna Technical Specification.

The temporary auxiliary crane and tower support are not specifically addressed in the existing station Technical Specifications. All Technical Specification requirements applicable during Cold Shutdown or Refueling/defueling will be adhered to. Therefore, the margin of safety as defined in the Technical Specifications will not be reduced.

SEV-1021
STEAM GENERATOR VESSEL, STEAM GENERATOR PIPING, AND INSULATION

STEAM GENERATOR VESSEL

This section of SEV-1021 evaluates the preparation of the steam generators for installation into containment including the removal of the nitrogen blanket, removal of welded closures on all piping nozzles (provided to prevent contamination of the generator internals during transit and storage), removal of shipping protection/supports, the machining of primary nozzles to allow fit-up with the existing reactor coolant piping, and the machining of secondary side nozzles to allow for fit-up with the existing secondary side piping.

All preparatory activities will be performed in accordance with the vendor recommendations, as applicable, to ensure that any assumptions utilized by the vendor in the design of the steam generators are not adversely affected.

Rigging and handling of the steam generators, to include staging of the steam generators inside the protected area, is addressed in the Safety Evaluation for DC-10034B, Steam Generator Rigging and Handling.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

The new steam generator preparatory activities only affect the steam generator vessels. All preparatory activities will be performed prior to the vessels entering containment, and will comply with the applicable codes and standards to ensure the existing design basis is satisfied. In addition, all preparatory activities will be performed in accordance with the vendor recommendations, as applicable, to ensure that any assumptions utilized by the vendor in the design of the steam generator are not adversely affected.

Once the nozzle shipping covers have been removed and when activities are conducted within the steam generators, appropriate measures will be taken to ensure that only materials compatible with the steam generators are utilized and that no foreign objects are introduced. All consumable materials utilized in the preparation of the steam generators for installation will be controlled in accordance with Procedure A-805, Control of Consumable Materials at Ginna Station. Procedural controls will be established to inventory all materials/tools entering the steam generator and to provide for a foreign object search following completion of internal steam generator activities.

Thus, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

Steam generator preparatory activities will only impact the replacement steam generators and will be performed in accordance with the applicable ASME codes to ensure no adverse impact to the steam generators. Procedural guidance for foreign object and consumable material control shall be in place to prevent deleterious materials from entering the steam generators. This activity will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

The preparation of new steam generators does not involve a change to the Technical Specifications. The design bases and performance criteria of the safety related steam generators has been shown to remain consistent with the applicable Design Criteria.

Therefore, the margin of safety as defined in the Bases section of the Technical Specifications will not be reduced.

STEAM GENERATOR PIPING

To replace the two existing steam generators, all connected piping/tubing must be severed to allow for removal of the steam generators. Following new steam generator installation, the piping/tubing must be reattached to the new steam generators. The systems affected are:

- o Reactor coolant hot and cross-over leg piping
- o Feedwater piping
- o Main steam piping
- o Steam generator blowdown and shell drain piping
- o Steam generator/main steam flow element instrumentation tubing
- o Loose parts monitoring system

Following severance of the old steam generator from the connected reactor coolant system piping, the extent of RCS pipe movement against machining and welding tolerances will be assessed, and, if unacceptable, the cold leg elbows may be replaced.

Due to the design of the feedwater and level instrumentation nozzles on the replacement steam generators, minor modifications to the piping/tubing is required. In addition, the existing condensate pot assemblies installed on the steam generator level instrumentation upper nozzles and the main steam flow elements, will be removed, discarded and replaced.

To facilitate non-destructive examinations, radiography access ports will be installed on the main steam and feedwater piping sections to be removed and replaced.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

All reactor coolant system and secondary side piping/tubing will be restored to satisfy the existing design requirements in accordance with ASME Section XI and III. Replacement materials, including all weld metal utilized, will satisfy the original code requirements and will meet the existing installation specification. No additional welds will be added to the reactor coolant system as the new welds will replace existing welds. However, the location of the safe end to elbow weld will change to improve the inservice inspection profile of the weld. New weld locations in the main steam and feedwater piping, if required due to fit up, and the safe end to elbow weld will be performed in accordance with approved procedures which will satisfy the original design requirements. Any changes in piping/tubing configuration will be reanalyzed as required to document the acceptability of the new configuration.

All new welds will receive nondestructive examinations in accordance with ASME Section III and will be hydrostatically pressure tested, as required. All operating design conditions shall remain unchanged.

Based on the above, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

Prior to severance of the RCS piping and until weld-out is complete, a designated deadweight support configuration will be established for the RCS loops that ensures no adverse effects will result to any safety-related components. The RCS piping will be severed using a machine cutting method to prevent debris from entering the piping during the cutting process and debris dams will be installed during final weld end preparation. Loop inspections for debris will be performed following completion of the cutting and milling.

Secondary plant piping systems including main steam, feedwater, blowdown, shell drain, steam generator/main steam instrumentation tubing, and the loose parts monitoring system may be severed from the steam generators while fuel is in the reactor vessel. In this configuration, the RHR system remains available for decay heat removal. Prior to piping severance, a designated support configuration shall be established that ensures no adverse effects, including gravity missiles and sway interaction concerns, will result to any safety-related components. This configuration will be maintained until defueling is complete or until the support is no longer required. During the cutting and welding process for secondary piping, debris control/protection will be provided to minimize the introduction of debris into the piping systems. Procedural controls will be established to provide for a foreign object search following completion of piping activities.

All reactor coolant system and secondary side piping/tubing will be restored to satisfy the existing design requirements using material which meet or exceed the original design requirements. No operational changes occur as a result of the piping/tubing replacements.

Based on the above, this activity will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

The activities required to remove and reinstall the piping/tubing connected to the steam generators will not require any changes to the technical specifications. All operating design conditions will remain unchanged and the affected piping systems will continue to be capable of performing their required safety-related functions.

No activities will be performed on permanent plant systems until the system has been released by operations. This ensures the technical specification operability requirements are maintained.

Therefore, the margin of safety as defined in the Bases section of the Technical Specifications will not be reduced.

INSULATION

The external features of the replacement steam generators are sufficiently different from the existing generators to warrant the replacement of the existing steam generator thermal insulation.

To facilitate severing of the steam generators from the connecting piping systems, a portion of the piping insulation must be removed and replaced. Because some of this insulation contained asbestos, the asbestos insulation was disposed of and new insulation was installed in its place during the 1995 outage. The piping insulation replacement was evaluated in Safety Evaluation SEV-1022.

This Safety Evaluation addresses the installation of insulation on the replacement steam generators which will be completed during the steam generator replacement.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

The steam generator insulation does not perform any safety function. Failure of the replacement insulation during postulated accidents as described in the UFSAR have been evaluated. The insulation will be designed and installed to Seismic II/I criteria to ensure that the insulation remains in place following a seismic occurrence. In addition, the insulation has been procured to meet or exceed the containment

post-accident environmental conditions. A debris analysis has been prepared to address the effect of the replacement insulation on the containment sump analysis and has determined that no adverse impact on the capability of safety-related components to perform their safety function has occurred. Therefore, the capability of the safety-related components will not be adversely affected by the replacement insulation.

Based on the above, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

The replacement insulation performs the same function as the original insulation and is installed to Seismic II/I criteria. Failure of the SG insulation does not, in itself, initiate an accident nor does the thermal insulation perform a safety function. Therefore, this activity will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

The activities required to install the thermal insulation on the steam generators will not require any changes to the Technical Specifications. All operating design conditions will remain unchanged and plant systems will continue to be capable of performing their required safety-related functions.

Therefore, the margin of safety as defined in the Bases section of the Technical Specifications will not be reduced.

LASER TEMPLATING

Upon completion of the steam generator replacement, the target nests installed in the 1995 outage will be removed and the target nest anchor bolts will be cut off flush with the surface of the cubicle wall, driven into the wall slightly, and the holes grouted. Grouting will be performed in accordance with RG&E Specification CE-153 and is classified as safety significant.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

The removal of the target nests and grouting of the drilled bolt holes will be performed in accordance with approved procedures to ensure the structural integrity of the cubicle wall is not compromised.

Based on the above, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

As a result of the procedural controls associated with the removal of the target nests, this activity will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

The removal of the target nests will not require any changes to the technical specifications. All operating design conditions will remain unchanged and plant systems will continue to be capable of performing their required safety-related functions. Therefore, the margin of safety as defined in the Bases section of the Technical Specifications will not be reduced.

STEAM GENERATOR LOWER SUPPORT TEMPORARY RESTRAINTS

The steam generator lower supports are pinned columns which will require stabilization during the steam generator replacement outage before being disconnected from the steam generators. The following is a description of the planned activities associated with the lower supports which will be performed in the 1995 refueling outage in preparation for the steam generator replacement (SGR) outage:

- Brackets will be installed (clamped) onto the support columns and tube steel will be welded to the existing primary manway access platforms. Temporary stabilizing braces with integral turnbuckles will be loosely fit-up (trial fit) to the column brackets and tube steel. The integral turnbuckle assemblies will not be connected to the pipe clamps, new tube steel or platforms at any time during the 1995 outage. The column brackets and tube steel will remain in place during plant operation and will not be removed until the end of the SGR outage.

- The column brackets and the tube steel will be installed in the cold shutdown or refueling mode. The temporary stabilizing braces, including turnbuckles, will be trial fit in the cold shutdown or refueling mode and removed from containment prior to entry in a mode above Cold Shutdown.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

The column brackets and tube steel are classified as safety significant, Seismic Category II/I and have been analyzed to ensure they will not fail and adversely impact safety related equipment during a seismic event. This design approach ensures that the installation of the brackets and tube steel will not adversely impact safety-related structures, systems or components.

During the 1995 refueling outage, the temporary restraint bracing will be trial-fit after the plant has entered Cold Shutdown or Refueling. However, the turnbuckle assemblies associated with the temporary restraint will not be connected during the 1995 outage. This administrative control ensures that the steam generator lower supports will continue to satisfy all design basis loading configurations.

The impact of the additional structural steel which will be added to containment during the 1995 to 1996 operating cycle has been qualitatively evaluated in this safety evaluation and determined to have an insignificant impact on the passive heat sinks and consequently the LOCA analyses.

Based on the above, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

Administrative controls and design of the brackets and tube steel to Seismic Category II/I criteria has been provided to ensure no adverse impact to safety related structures, systems or components in the event of a design basis earthquake. In addition, the temporary stabilizing bracing will only be trial fit to the column bracing. The integral turnbuckle assemblies will not be connected to the bracing and will be removed prior to entering a mode above Cold Shutdown. Thus, this activity will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

The Seismic Category II/I design of the brackets combined with the limits imposed regarding the plant conditions under which trial fitup of the temporary stabilizing braces can be performed ensures that the structural integrity of the reactor coolant system and all design basis loading configurations are maintained. Therefore, the margin of safety as defined in the Bases section of the Technical Specifications will not be reduced.

AUXILIARY CRANE ANCHOR BOLTS

A temporary auxiliary crane will be installed and used in the containment during the steam generator replacement (SGR) outage. To support the installation of the auxiliary crane, anchor bolts will be installed in the Elevation 278'-4" floor slab during the 1995 refueling outage.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

The anchor bolt installation will be in accordance with existing station technical specifications and has been evaluated to ensure no adverse impact the Elevation 278'-4" floor slab. Installation of the anchor bolts in accordance with RG&E Technical Specification CE-153 will ensure that the existing design basis for the floor slab is maintained.

Based on the above, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

No new accidents or malfunctions have been postulated since no new failure modes have been created.

No adverse change in the structural integrity of the affected structure will result for the implementation of the modifications addressed in this safety evaluation. The impact of the anchor bolt installation on the safety-related floor slab has been evaluated and determined to be acceptable. The final configuration of the floor slab following the installation of the anchor bolts will satisfy the existing design basis.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

The activities associated with installation of the anchor bolts to support the auxiliary crane and tower will not reduce the margin of safety as described in the basis for the Technical Specification 3.6. The installation of the anchor bolts have been evaluated to be acceptable and will be installed in accordance with existing station Specification CE-153 to ensure the Elevation 278'-4" floor slab is not adversely affected.

CONTAINMENT HVAC DUCT MODIFICATIONS

A portion of the containment HVAC duct installed on the containment dome is an interference to the steam generator B construction opening and must be temporarily removed during the SGR outage. Two flexible duct connections containing asbestos will also be removed and discarded as part of this modification. To facilitate the removal of this duct and eliminate concerns associated with asbestos abatement during the SGR outage, the expansion joint fabric connections will be disassembled and removed during the 1995 refueling outage (RFO). Existing plant drawings do not clearly identify the as-built configuration of the expansion joints. However, based on a visual examination of similar accessible expansion joints within the containment, an inner metal sleeve may be installed which extends the full length of the expansion joint. Consequently, replacement of the expansion joint fabric will be performed under one of the following two scenarios:

- If the expansion joint does not contain an inner metal sleeve, new non-asbestos bearing fabric will be installed during the 1995 RFO.
- If the expansion joint contains an inner metal sleeve, new non-asbestos bearing fabric will not be installed until the end of the 1996 SGR outage.

To provide flexibility, each of these replacement scenarios must be evaluated and determined to be acceptable.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

The design bases of the affected system will not be altered. All of the affected equipment will be re-installed to satisfy the current design basis.

The containment recirculation and cooling systems will not be modified until the plant has reached Cold Shutdown. With the plant in the cold shutdown condition, the design basis events for which the containment ventilation system is required to be operable will not occur. Thus, after reaching Cold Shutdown, the containment ventilation system is no longer required to perform a safety function.

The containment recirculation and cooling system expansion joint replacement is a like-for-like replacement, with the exception of the material requirements. The materials have been selected in accordance with the existing RG&E Technical Specification SP-5342. The effects of potentially not re-installing the expansion joint fabric material until the end of the 1996 SGR outage have been evaluated and determined to be acceptable.

Based on the above, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

As a result of the controls imposed regarding when the system can be modified, and since the system will be reinstalled in accordance with the current design requirements, this activity will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

As a result of the controls imposed regarding when the system can be modified, the margin of safety as defined in the Bases section of the Technical Specifications will not be reduced. All requirements of Technical Specification Section 3.3.2 will be adhered to during the implementation of this modification.

LASER TEMPLATING

To achieve accurate fit-up of the new steam generators to the existing RCS piping, laser templating techniques will be employed to obtain accurate measurements of the pertinent features of the existing and new steam generators and the severed ends of the RCS pipes. The laser templating technique requires the mounting of fixed target nests (reference points) on the steam generator cubicle walls and laser reflecting targets placed at strategic points on the cubicle walls, generators and piping. The target nests will remain in place during plant operation and will be removed upon completion of steam generator replacement. The laser reflecting targets are only installed during the measurement process and will be removed prior to plant restart.

The anchor bolts used to install the target nests have sufficient capacity to support the nests, which weigh less than one pound.

Acid etching and mechanical buffing may be performed to expose the existing steam generator nozzle to RCS piping weld. Low stress punch marks may also be applied to mark reference locations on the steam generator nozzle to RCS piping welds.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

Mechanical buffing, acid etching, and the application of low stress punch marks to the RCS piping will be performed in accordance with approved procedures. Performance of these activities will be procedurally controlled to limit the effect on the RCS piping. The amount of base metal removed by buffing and etching and the depth of the punch marks are insignificant when compared to the thickness of the RCS piping and will have no effect on the capability of the RCS to maintain the reactor coolant pressure boundary. Minimum wall thickness for the reactor coolant piping will be maintained.

The amount of aluminum added has been evaluated and determined to have a negligible impact on the hydrogen generation within containment following a loss-of-coolant accident.

Cutting of rebar during installation of the anchor bolts to support the target nests is not permitted unless previously approved by engineering, and the structural adequacy of the affected structure with cut rebar has been confirmed by analysis or evaluation. Therefore, this activity will have no effect on the structural integrity of the steam generator cubicle wall. In addition, the target nests will be installed with anchor bolts sized to ensure the target nests remain intact during a seismic event. The additional loads of the target nests on the steam generator cubicle walls are insignificant.

Based on the above, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

As a result of the procedural controls and design of the nests to seismic II/I criteria, this activity will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

The laser templating activities will not require any changes to the technical specifications. All operating design conditions will remain unchanged and plant systems will continue to be capable of performing their required safety-related functions.

Therefore, the margin of safety as defined in the Bases section of the Technical Specifications will not be reduced.

ASBESTOS ABATEMENT AND INSULATION REPLACEMENT

To facilitate replacement of the steam generators in the 1996 outage, asbestos-bearing and non-asbestos insulation in the area of the proposed piping cut locations will be removed and replaced with non-asbestos bearing blanket insulation during the 1995 refueling outage. Replacing the existing insulation with fiberglass blanket insulation in the area of the piping cut locations during the 1995 refueling outage will minimize the need for asbestos abatement during the 1996 steam generator replacement (SGR) outage.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

The thermal insulation performs no safety function. Failure of the replacement insulation during postulated accidents as described in the UFSAR have been evaluated. The insulation will be designed and installed to Seismic II/I criteria to ensure that the insulation remains in place following a seismic occurrence. In addition, the insulation has been procured to meet or exceed the containment post-accident environmental conditions. A review of the replacement insulation with respect to the impact on the containment sump has been performed. This review has determined that replacing the existing insulation with blanket insulation in the vicinity of the piping cut locations is no worse than the existing insulation or has been evaluated to have a negligible impact. Therefore, the capability of the safety-related components will not be adversely affected by the replacement of insulation in the vicinity of the piping cut locations.

Based on the above, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

The replacement insulation performs the same function as the original insulation and is installed to Seismic II/I criteria. Failure of the piping insulation does not, in itself, initiate an accident nor does the thermal insulation perform a safety function. Therefore, this activity will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

The activities required to remove and reinstall the thermal insulation on the connected piping will not require any changes to the Technical Specifications. All operating design conditions will remain unchanged and plant systems will continue to be capable of performing their required safety-related functions.

Therefore, the margin of safety as defined in the Bases section of the Technical Specifications will not be reduced.

TEMPORARY BRIDGE CRANE WORK PLATFORM

In order to remove the steam generators through openings in the containment dome, portions of the containment spray piping, containment HVAC duct, and painter's trolley rails must be removed and re-installed. To perform these and other related activities, worker access to the inside of the containment dome is required during the 1995 and 1996 refueling outages. For the 1995 refueling outage, worker access to the dome is required to support decontamination of the liner plate and to support the containment HVAC ductwork modifications.

To provide this access, a temporary work platform will be erected on top of the bridge crane trolley. The work platform will consist of a structural steel frame and scaffolding. The structural steel frame will be bolted to the bridge crane trolley which will act as a base for the scaffold structure above.

Upon completion of the 1995 refueling outage, the structural steel frame will remain attached to the bridge crane trolley during the 1995-1996 operating cycle to allow reuse during the steam generator replacement outage. The scaffolding connected to the steel frame will be removed upon completion of the 1995 refueling outage and reinstalled to support the 1996 steam generator replacement outage. Upon completion of the steam generator replacement outage, the structural steel frame and scaffolding will be removed from containment.

In addition to the installation of the structural steel frame and scaffolding, the existing jib crane mounted on top of the bridge crane trolley may also be modified, if required, to support installation of the structural steel frame and scaffolding.

This section of the safety evaluation addresses the safety impact of erecting and utilizing the temporary work platform and jib crane modifications on top of the bridge crane trolley.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

The bridge crane temporary work platform, to include the scaffolding, and the jib crane modifications have been classified as safety significant, seismic Category II/I and have been analyzed to ensure that the components will not fail and adversely impact safety related equipment during a seismic event. All installation and removal activities conducted with fuel in the containment building will satisfy the requirements of RG&E

Administrative Procedure A-1305.1, as applicable, and the additional administrative controls imposed by this safety evaluation. The scaffolding associated with the temporary work platform will be removed prior to leaving the cold shutdown mode.

The addition of the structural steel base during the 1995-1996 operating cycle will not adversely affect the containment spray coverage during a postulated accident. The spray system will continue to cover a maximum portion of the free volume space. The additional steel added to containment is negligible when compared to the total amount of structural steel in containment. Therefore, the addition of the base during the 1995-1996 operating cycle will not adversely affect the heat sinks used in the accident analyses.

The structural steel which remains within containment will be coated with coatings that satisfy the requirements of CE-125 to ensure that a coating system approved for in-containment usage is applied.

This design approach ensures that the installation of the bridge crane work platform and the jib crane modifications, and operation of the plant through the 1995-1996 operating cycle with the platform base installed, will not adversely impact safety-related structures, systems or components. The temporary platform will be completely removed upon completion of the 1996 steam generator replacement outage.

Thus, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

As a result of the load handling administrative controls imposed by this safety evaluation and the design of the work platform and jib crane modifications to withstand seismic loadings, this activity will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

Neither the bridge crane temporary work platform nor the jib crane modifications involve a change to any Ginna Technical Specifications. Adequate precautions in the design and installation/removal of the temporary work platform and jib crane modifications have been implemented to minimize adverse impact to structures, systems or components which are required to be operable while the bridge crane temporary work platform is installed.

The bridge crane is not specifically addressed in the existing station Technical Specifications.

All technical specification requirements applicable during cold shutdown or defueling/refueling will be adhered to.

Therefore, the margin of safety as defined in the Technical Specifications will not be reduced.

SEV-1023

STEAM GENERATOR REPLACEMENT PRE-SGR OUTAGE EXCAVATION AND CONSTRUCTION ACTIVITIES

To accomplish the steam generator replacement, a Transi-Lift crane will be utilized to lift the old steam generators from the containment and install the replacement steam generators. To facilitate these heavy load movements, engineered foundations will be constructed for the Transi-Lift at a working location east of the containment. The construction of the foundations will include excavation, backfill, and concrete placement. Construction of the foundations is intended to be performed with the reactor at power but may be accomplished during any mode of reactor operation.

To withstand the surcharge loadings associated with the Transi-Lift and the site transporter/prime mover utilized to transport the steam generators to and from the downending/upending location, the existing 10-inch thick concrete slab located above the 34 kV underground ductbank plus associated control sections and the 115 kV underground ductbank will be replaced with a 2'-0" thick concrete slab. This modification will include excavation, backfill, and concrete placement. This modification is intended to be performed with the reactor at power but may be accomplished during any mode of reactor operation.

During excavation activities associated with the foundations and ductbank slab modifications, soil removed from the excavated areas may be contaminated. Any excavated soil removed during the construction activities will be identified and handled in accordance with existing site procedures.

To support installation of the Transi-Lift foundations, existing utilities either underneath the foundation or located above grade within the foundation's footprint area will be permanently relocated. The relocation activities include the following components: 1) rerouting the yard loop fire protection line between valves 8570 and 8573, 2) relocating the stepdown transformer, the 480/208/120 volt distribution center and power pole #436-H-4 serving the outage support trailers.

To support installation of the Transi-Lift foundations, existing utilities either underneath the foundation or located above grade within the foundation's footprint area will be temporarily disconnected, removed and reinstalled following the replacement outage. These commodities include the 300kVA transformer supplying power to outage support trailers and the equipment hatch personnel lock track including the rail stops and perforated underdrain.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

As described in the above sections, movement of construction equipment in support of the construction activities has been evaluated and determined to have no adverse impact to the buried commodities. The construction vehicles utilized during the construction activities associated with the Transi-Lift foundations will remain bounded by the analysis performed in EWR 3681 for the GSU transformer removal and replacement. Since these vehicles will be bounded by this analysis they will have no adverse impact on the underground utilities beneath the 10-inch slab. The accidents and transients evaluated are not directly impacted by the construction equipment or by the construction activities. A review of the accidents in the UFSAR has been completed and the mitigating measures described in the UFSAR have not been altered or affected.

The performance of existing fire protection equipment is not adversely affected by the rerouted piping. A comparison of the rerouted fire protection piping to the existing piping layout has been performed. This comparison has determined that, due to the type and size of fittings removed and replaced, there is no decrease in the systems capability. Therefore, there is no adverse impact on the capability of the existing system.

Administrative controls will be invoked to ensure that heavy equipment remains a minimum distance of 10 feet away from the Auxiliary Building or safety-significant structures, systems, or components identified in Table 1. In those instances where this minimum distance cannot be maintained, an evaluation, on a case-by-case basis, will be performed to determine a revised minimum distance and/or additional controls required to prevent inadvertent damage to the Auxiliary Building or safety-significant structures, systems, or components identified in Table 1. Furthermore, as stated above, the backhoe does not directly bear on the duct banks or associated cables and the integrity of the duct banks/cables will not be adversely affected. The concrete slab will be broken up into pieces approximately six inches in diameter. These pieces are of sufficient size such that the inadvertent drop of rubble from the backhoe will not adversely affect the integrity of the duct bank. Therefore, the administrative controls, as well as, a combination of machine and manual excavation will be employed to minimize the potential for a construction incident which could lead to initiation of event.

Based on the above, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

Movement of construction equipment has been reviewed to identify the most severe surcharge loadings associated with these components and the buried commodities have been reviewed for these loadings. The loadings associated with the construction vehicles is either bounded by the surcharge loadings associated with the Transi-Lift/transporter and prime mover or will be bounded by the EWR 3681 loads previously evaluated for the underground ductbank. Therefore, there is no adverse impact to the buried commodities as a result of the construction activities.

Above ground commodities have also been reviewed and relocated, if necessary. Relocation of above ground commodities will be performed to satisfy the current design basis to ensure the relocated commodity continues to perform its design function. Temporary protective measures, such as security watches, will be invoked, where appropriate, to ensure no adverse reduction in the degree of protection or function of systems or components during the commodity relocation.

As stated above, during construction activities, administrative controls will be established to ensure that construction equipment/personnel remain a sufficient distance away from energized components. In addition, during removal of the 10-inch slab, the backhoe will not directly bear on the duct banks/cables. Therefore, the integrity of the duct banks/cables will not be adversely affected.

During construction activities, temporary barriers will be utilized, as required, to limit surface water runoff towards the protected areas. Temporary provisions will be utilized, as necessary, to limit sediment infiltration into the storm drain system. After the construction activities are completed, the site will be graded to conform to the contours of the site prior to the construction activities. Where this is not possible, the soil will be graded such that surface water runoff is directed away from the protected area.

Based on the above, this activity will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

This activity will not affect compliance with requirements of the Technical Specifications nor will these activities require a change to the Technical Specifications. The service water systems and the electrical power duct banks have been evaluated and protected, as required, to ensure the requirements of Technical Specification 3.3.4 and 3.7, respectively, continue to be met.

Therefore, the margin of safety as defined in the Bases section of the Technical Specifications 3.3.4 and 3.7 will not be reduced.

While the plant is in operation and prior to achieving Cold Shutdown, miscellaneous items will require installation either on the containment dome or within the confines of the facade. These items include, but are not necessarily limited to, the following:

- Crown platform on the containment dome;
- Craft break area on the facade truss;
- Walkways from the break area to the crown; and
- Miscellaneous small structural members for facade upgrades.

The crown platform is a structural steel structure which will be temporarily erected on the containment dome. The crown platform is addressed in the Safety Evaluation for DC-10034A, Containment Structural Modifications.

To accomplish the rigging and handling associated with these components with the reactor in a mode above Cold Shutdown, the following temporary modifications are required.

An air winch will be installed at elevation 275'-0" on the concrete foundation for the moisture separator relief valves and exhaust piping. This winch will be installed and utilized to lift material and equipment from the ground elevation to the tendon access platform at elevation 343'-0".

A support beam assembly will be installed at approximately elevation 378'-0" on the eastern facade structure to accommodate the hoist snatch block required for lifting material and equipment from the ground elevation to the tendon access platform at elevation 343'-0". The beam will be attached to a new vertical member spanning between existing facade truss chord members at elevations 360'-0" and 386'-0". This beam assembly will be utilized in conjunction with the air winch.

A protective barrier will be installed above the existing cable tunnel, routed east of containment and within the confines of the facade structure, to protect the cable tunnel from a postulated load drop during the rigging activities associated with lifting material and equipment from the ground elevation to the tendon access platform. The barrier will be placed over the cable tunnel and will be constructed of oak timbers to a 4'-0" thickness, overlaid with a 1-inch thick steel plate.

To facilitate installation of the temporary decking, enclosures, and walkways on the facade structure, the existing facade structure horizontal trusses located on the West side of the facade at elevation 386'-0" will be utilized as lift points for the rigging required to lift material from the tendon access platform into position at elevation 360'-0" for the facade modifications. Column "G4" will be extended vertically using an extension clamped to the existing column flange to support a lift point.

To facilitate the installation of the crown platform on the containment dome, the existing facade structure horizontal trusses located on the South side of the facade at elevation 386'-0" will be utilized as lift points for the rigging required to lift material from the tendon access platform into position for the erection of the crown platform.

To facilitate the installation of the crown platform on the containment dome, a crown platform erection crane will be utilized on the containment dome. This crane consists of a structural steel support frame and an intermediate-capacity truck crane, Model TM-4000 by Air Technical Industries. This crane assembly will be attached by two, 3/4-inch diameter wire ropes to a dome anchorage located at the apex of the containment dome. Cable dollies will be located between the dome anchorage and the crane to provide clearance between the wire ropes and the containment concrete.

Installation and use of the winch, crown platform erection crane, support beam assembly, cable tunnel protective barrier and use of the lift points on the western and southern portions of the facade structure may be performed in any mode of plant operation.

Use of the lift points and removal of the winch, crown platform erection crane, support beam assembly, column extension, and the cable tunnel protective barrier may be performed during any mode of plant operation.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

The design, material, and construction standards applicable to permanent plant SSCs are unaffected by the temporary installation of these load handling components. The winch, facade modifications and the crown platform erection crane have been evaluated to ensure that the components will not fail and adversely impact safety related equipment during a postulated seismic event. The defense-in-depth philosophy described in this safety evaluation, which includes identification of safe load paths, localized protection of key shutdown safety functions, and administrative and design controls of lifting activities ensures the continuity of key shutdown safety functions. The temporary installation of the equipment for the associated load handling operations do not change, degrade, or prevent actions described or assumed in any accident discussed in the UFSAR. Therefore, the rigging activities associated with the load handling activities will not increase the probability of occurrence of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

As a result of the load handling administrative controls imposed by this safety evaluation and the design of the rigging equipment to not adversely affect safety related equipment during a postulated seismic event, this activity will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

The load handling associated with the rigging activities will not reduce any margin of safety as defined in the basis of any Technical Specifications. The installation and use of the temporary rigging equipment does not adversely affect the operation of any safety-related structures, systems, or components. Furthermore, the defense-in-depth philosophy includes identification of safe load paths, localized protection of key shutdown safety functions, and administrative and design controls of lifting activities ensures the continuity of key shutdown safety functions.

Temporary construction power and distribution systems will be required to support temporary facilities prior to, during, and after the 1996 steam generator replacement outage for construction activities and facilities located inside the containment, on the containment dome, and outside the containment. 480 Volt temporary power sources will be located within close proximity to all temporary facilities (i.e., mockup/fabrication/warehouse facility, dome mock-up area, old steam generator storage facility (during construction), containment access facility, containment dome area, and the contaminated storage area). The temporary power sources outside containment are described below.

Three temporary stepdown transformers will be provided outside of containment to service steam generator replacement loads. Two transformers will be fed from one of the existing 34 kV offsite transmission sources (Circuit 751) with the third transformer fed from an existing 12kV source (Circuit 5202). One of the two transformers (I-3) fed from Circuit 751 is located in the vicinity of the Containment Access Facility and the second transformer (I-1) is located near the steam generator replacement temporary offices and construction facilities. The transformer fed from the Circuit 5202 (I-2) is located near the old steam generator storage facility and the dome mockup facility.

The temporary distribution system inside containment will utilize the 4160V, 3 Phase, 60Hz Reactor Coolant Pump (RCP) "B" power feed supplied from Bus 11B. The existing RCP "B" motor leads will be disconnected and coiled back. A temporary stepdown dry type transformer (4160/480V, with close coupled power distribution switchboard) will be temporarily connected to the 4160V bus/RCP feed to supply in containment steam generator replacement electrical loads. From the temporary power distribution switchboard, four power panels will be installed which will supply power to support in-containment steam generator replacement loads. These loads include additional lighting, pipe cutting and welding equipment, pipe end decontamination equipment, portable ventilation units, post weld heat treatment equipment, the auxiliary crane as well as other temporary loads required to support the replacement.

All of the temporary power distribution commodities installed prior to or during the steam generator replacement, both inside and outside containment, will be removed after completion of the work activities related to the steam generator replacement.

This safety evaluation addresses the impact of connecting loads to the temporary transformers described above.

Will the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

The RCP "B" motor feeder will not be disconnected from the pump motor and utilized as the feeder for the temporary power distribution system until the plant has reached Cold Shutdown. De-termination, re-termination, and functional testing of the reactor coolant pump will be performed.

The design bases of the affected system will not be altered by the temporary modifications to the non-Class 1E distribution system. In addition, the temporary electrical system will not interface with Class 1E or associated 1E circuits. All of the affected equipment will be re-installed to satisfy the current design basis.

Two of the three stepdown transformers will be provided power from the 34 kV offsite power source (Circuit 751). Fused disconnects at the stepdown transformer as well as additional protection at the load point have been provided to protect the 34kV source. Protection is coordinated to ensure that faults on temporary circuits will not cause a loss of Circuit 751 to the station bus. The additional loading on Circuit 751 has been evaluated and does not affect the ability of Circuit 751 to maintain voltage.

Movement of the temporary transformer and main power distribution switchboard with fuel in the containment will be performed to satisfy the requirements of RG&E Administrative Procedures A-1305 and A-1305.1, as applicable. Furthermore, the Elevation 274'-6" floor slab is capable of withstanding the temporary transformer and main power distribution switchboard loading.

Safety-related structures, systems, and components will not be adversely affected by the temporary power distribution system since:

- the location of the temporary power distribution system equipment inside containment will be such that components will not impact safety-related structures, systems, and components that are required to function during Cold Shutdown, Refueling, or the defueled mode of operation;

- those portions of the containment dome temporary power distribution system whose failure could adversely impact adjacent safety-related equipment during a seismic event are anchored; and

- the other portions of the distribution system located outside of containment, but not on the containment dome, are located at a sufficient distance away from safety-related structures, systems, and components.

Based on the above, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

Following completion of the steam generator replacement, the temporary power distribution system will be removed and reactor coolant pump "B" will be returned to its pre-outage condition. No adverse change in the functional integrity of affected components will result by the implementation of the temporary modifications addressed in this safety evaluation. The impact of the temporary transformer and power distribution switchboard on permanent plant systems have been evaluated and determined to be acceptable. Therefore, no new accidents or malfunctions have been postulated since no new failure modes have been created.

As a result of the controls imposed regarding when the systems can be modified, and since the systems will be reinstalled in accordance with the current design requirements, this activity will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

The temporary modifications do not involve a change to any Ginna Technical Specification.

All requirements of Technical Specification Sections 3.7 and 3.11 will be adhered to during the implementation of this modification. Protective devices both at the stepdown transformers as well as the load point have been provided to protect upstream sources from potential faults due to steam generator replacement loads. Therefore, as a result of the controls imposed regarding when the systems can be modified, the margin of safety as defined in the Bases section of the Technical Specifications will not be reduced.

SEV-1036
LIMITED CONSTRUCTION PACKAGE CARDREADER INSTALLATION

This change is a modification to the plant for installing new security card readers. The limited construction package includes the installation of conduits from RMT 3 to IMUX 3 and RMT 4 to IMUX 4 and pulling in cable from the Steam Generator Building to the East DP, from the Steam Generator to the Guard Tour Reader in the South East corner of the site, from Projects Building to the West DP and from Projects Building to the Guard Tour Reader in the South West corner of the site.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased. There is no new components being added by the proposed change. Only new conduit in non-safety related areas in the turbine building basement and service building basement. The conduit will not contain safety related cables. The conduits and cables perform no safety related or control function. Therefore, the probability of an accident or malfunction are not increased.

Since the modification does not interface with anything that could cause an accident or used to mitigate an accident the possibility of an accident or malfunction of a different type than evaluated previously in the SAR is not increased.

The margins of safety as defined in the basis for any Technical Specification is not reduced because the proposed change does not increase the consequences of any accident.

SEV-1037
AMPTECTOR SETPOINT CHANGES ON DB TYPE BREAKERS

This safety evaluation reviews the modification of over current protection setpoints for both safety related and non-safety related equipment DB-Breakers.

Overcurrent tripping of DB breakers is accomplished by the following breaker sub-compartments, 1) three current transformers, 2) an Amptector and 3) an actuator. The current transformers sense the current flowing through the breaker and provide input into the Amptector. The Amptector monitors the level of current and provides an electrical trip signal to the actuator when the current exceeds a predetermined level and maintains that current for a predetermined time. The actuator is an electrical coil which provides the motive force to trip the breaker.

Setpoint changes are indicated per design analysis DA-EE-93-104-97 "480 Volt DB Breaker with Amptector Retrofit Coordination and Circuit Protection Study", Revision 1, EWR 4225.

This study documented setpoint criteria for breaker coordination and equipment overcurrent protection and analyzed existing setpoints based on this criteria.

Setpoint criteria was based on recent industry standards in equipment overcurrent protection. The analysis also used actual field measured loading data for all non-safety and safety related motors. The analysis also considered the postulated condition of degraded bus voltages. Setpoint criteria was established for both safety related motors and safety related motor control centers for the condition of degraded bus voltages. The analysis demonstrates that satisfactory overcurrent protection margin exists for safety related motors and motor control centers based on normal bus voltages, although setpoint changes are recommended for postulated degraded voltage conditions.

In addition to equipment protection the analysis evaluated the setpoints for coordination of upstream and downstream breakers to ensure selective tripping occurs. Selective tripping is the setting of breaker trip setpoints to ensure the breaker closest to the overloaded equipment or electrical fault will trip first. Selective tripping prevents a single fault or single overloaded piece of equipment from tripping the main voltage source causing the loss of an entire bus.

Included in the selective tripping analysis was the consideration of a fault on safety related buses 14, 16, 17 or 18 causing an undervoltage relay to trip prior to the bus feeder breaker tripping. Selective tripping ensures a fault on one of these buses will result in the feeder breaker tripping first preventing the Diesel Generators from closing in on a faulted bus.

Completion of the suggested setpoint modifications will 1) establish consistent overcurrent protection margins for similar type loads, 2) ensure safety related motors and motor control centers will remain energized during conditions of degraded bus voltages, 3) ensure cable and transformer thermal design limits are not exceeded, 4) ensure selective tripping occurs.

For all affected loads this modification will increase the equipment reliability and maximize its potential availability by ensuring equipment is not lost until overcurrent tripping is required and by ensuring selective tripping results in the breaker closest to the overloaded equipment or fault trips first.

The causes of over current conditions affecting equipment important to safety are independent of the mitigation features used to isolate affected loads. The probability associated with the occurrence of an overcurrent condition affecting equipment important to safety is therefore unrelated to the setpoints of the equipment used to mitigate overcurrent conditions.

The consequences of an overcurrent condition will be unchanged as a result of this modification because the modification will not alter the essential protection processes used to secure vital equipment availability. The setpoint changes proposed for this modification are within standards that were established to ensure that overcurrent protection and breaker selective tripping preserves the integrity of the vital buses.

It can be concluded that the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report will not be increased as a result of this proposed modification.

After completion of the proposed modification the electrical distribution system will function in the same manner as before the change. Because no components are being added or removed and because the functions of the existing equipment will remain unchanged the possibility for introducing an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created.

Overcurrent protection device setpoints are below the level of detail discussed in the Updated Final Safety Analysis Report and Plant Technical Specifications. The overcurrent trip features of breakers are not utilized in defining the margin of safety for any technical specification. Because the amptector setpoints are not factored into the Technical Specifications, changes to these setpoints will not result in the reduction in a margin of safety.

SEV-1038
REMOVAL OF THE RADWASTE COMPUTER SYSTEM

This safety evaluation examines the proposed removal or abandonment of the Radwaste Computer System. This system was designed to provide remote monitoring of select waste processing parameters under post accident conditions.

The Radwaste Computer System contains operating and display terminals in the Technical Support Center and at the drumming station in the Auxiliary Building. System interface racks and multiplexers are also located in the Auxiliary Building while the control processing unit is in the technical support center.

The scope of this proposed modification is the removal or abandonment of the Radwaste Computer System operating terminals (two), central processing unit, interface rack, multiplexers, and associated cabling. After completion of this change all waste disposal system local indications will remain functional.

Completion of this modification will result in deletion of a regulatory commitment and several Updated Final Safety Analysis Report (UFSAR) changes.

The Radwaste Computer System does not interface with Nuclear Safety grade equipment, nor is information displayed on the system used as the basis for manual operation of equipment important to safety. Because the system provides no information or control with respect to equipment important to safety abandonment or removal of the Radwaste computer can not have any affect on the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety.

Removal or abandonment of the Radwaste Computer System cannot create the possibility for an accident or malfunction of a different type than any previously evaluated in the safety analysis report because no Radwaste or Electric power system functions are changed. The system is not utilized in the mitigation of any accident sequences and is not now included in emergency operating or long term necessary procedures. Therefore, its abandonment in place, cannot create the possibility of a new malfunction.

The Radwaste Computer is not related to any technical specification and therefore the margin of safety as defined in the basis for any technical specification is not reduced.

SEV-1039
5A/5B FEEDWATER HEATERS

EWR 4812 was initiated as a modification to procure and replace the number 5A and 5B feedwater heaters. The current configuration of the 5A/5B high pressure feedwater heaters (EFW05A/EFW05B) utilizes copper based alloy tubes. Copper has been determined to be a contributing factor in the corrosion mechanism which causes steam generator tube wastage at Ginna Station. Tube material in the 5A/5B heaters has degraded through the years due to normal operation and chemical cleaning.

The number 1, 2, 3A/3B and 4A/4B low pressure feedwater heaters were previously removed and replaced with heaters which utilize stainless steel tubing. The installation of new high pressure feedwater heaters is the next step in a planned phaseout of copper based materials in the condensate/feedwater systems.

It has been concluded in Design Analysis ME-93-026 that the existing 5A and 5B Feedwater Heaters will be replaced with heaters manufactured by Marley Heat Transfer, a division of the Engineers and Fabricators Corporation (EFCO).

The replacement heaters have been procured within the intent of Mechanical Engineering Equipment Spec ME-312. This specification requires the replacement heater to be equivalent to the existing heaters in fit form and function with the exception of using increased surface area of tubing to account for the lower heat transfer characteristics of stainless steel tubes as compared to the existing copper based tubes.

The proposed modification would not increase the probability of occurrence of an accident evaluated previously in the UFSAR because:

a. The equipment proposed would not introduce any likely ignition sources that could start a fire.

b. All feedwater heater related events had one change that affected the accident analyses. This change was an increase in flow resistance over feedwater heaters 5A and 5B. This change had either a positive affect or little to no affect on the related events.

c. Some accidents are conservative with high feedwater temperature and some are conservative with low feedwater temperature. The overall importance of feedwater temperature as previously evaluated accidents is expected to be minor, with accident results being relatively insensitive to changes in feedwater temperature. Performance tests will be run to document outlet feedwater temperatures. The results will be provided to NS&L for review.

d. Other analyzed incidents such as floods, turbine missile and station blackout were not thought to be affected by the proposed modification.

The proposed modification would not increase the consequences of an accident previously evaluated in the UFSAR because:

a. The feedwater heaters 5A and 5B have turbine extraction steam on the shell side and main feedwater on the tube side. Main feedwater and secondary steam systems are not required during a design basis accident since they are not safety related systems.

The proposed modification would not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR because:

a. The proposed equipment would be sized, specified and installed in accordance with existing piping and heat exchanger classifications and code designations for design, material and construction.

b. The proposed modification is not safety-related and does not perform any safety function.

The proposed modification would not increase the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR because:

a. The only type of accident that would allow radioactive material in the secondary system is a steam generator tube rupture. All other accidents and normal operations would not have radioactive materials in the secondary system, where the proposed modification will take place. Even though the flow resistance in the feedwater heaters has increased, the total flow resistance and the total flow will remain the same and hence cause no more or less contamination of the secondary system.

The proposed modification would not create the possibility of an accident of a different type than any previously evaluated in the UFSAR because:

a. The proposed modification equipment is not safety-related and will not be relied upon to mitigate an accident. All plausible accidents that involve the feedwater system were evaluated in the UFSAR. Since the proposed modification is a suitable replacement, the proposed modification would not create the possibility of an accident of a different type than any previously evaluated.

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:

a. Replacement of the feedwater heaters 5A and 5B will not create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated since the feedwater heaters are technically improved and functionally equivalent to the existing feedwater heaters.

The proposed modification would not reduce margins of safety as defined in the basis for any technical specification because:

a. The feedwater heaters 5A and 5B are replacements for existing feedwater heaters and are not required to meet system design bases criteria. The feedwater heaters are technically improved due to the tube material being stainless steel instead of copper based, which will improve the steam generator reliability. Therefore main feedwater pump delivery will not be significantly altered. The proposed feedwater heaters have slightly more flow resistant than the existing feedwater heaters and consequently a higher differential pressure.

b. The proposed modification does not change any of the setpoints or limits that are written in the Ginna Technical Specifications.

c. Based on the information contained in Section 5.9, the margin of safety should remain the same due to the proposed modification being equivalent to the existing feedwater heaters 5A and 5B. The feedwater heaters 5A and 5B have little impact on the overall margin of safety since the equipment is non-safety related.

Based on the preceding, the proposed modification does not involve a change in the Technical Specification or the USFAR and is not an unreviewed safety question.

SEV-1040
CARDREADER INSTALLATION

This change is a modification to the plant for installing new security card readers. The construction package includes the removal of existing card readers and door control panels, installation of new card readers and new door control panels with the existing Hoffman enclosures, removal of the internals of old security MUXs and reusing the enclosures as terminal boxes, disconnecting security doors and fire doors from the old security computer system and reconnecting to the new security computer system.

The probability of occurrence of the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased. The modification provides/denies access to plant areas. Access control does not affect the probability or consequences of an accident or malfunction.

Access control can not create the possibility of an accident or malfunction of different type than previously evaluated.

If access control has no affect on the accident analysis, it can not affect the margins of safety as defined in the basis for any technical specification.

SEV-1041
RHR PUMP PIT LEVEL SENSOR/ALARM

The Auxiliary Building Sub-Basement sump is currently dewatered by an original plant sump system. The dewatering portion of the system is comprised of two sump pumps controlled by associated float type level switches. Indication of sump pump operating status is displayed on the Main Control Board (MCB) via read and green status lights, MCB annunciator L10 (Aux. Bldg. Sump Pump Auto Start), and the Plant Process Computer System (PPCS) via auxiliary motor starter contacts. Two diaphragm type level switches located in the sump provide a common sump high level alarm on MCB annunciator L9 (Aux. Bldg. Sump Hi Level). The existing sump dewatering system is currently classified as Safety Significant and possesses no environmental qualification.

During an NRC safety system functional inspection, a potential flooding issue related to the Auxiliary Building Sub-basement was identified. In the event that a residual heat removal pump seal failure occurs twenty four (24) hours into the recirculation phase of a loss of coolant accident, a leak of fifty (50) gallons per minute creating a harsh environment would exist in the Auxiliary Building Sub-Basement. The only dewatering/alarm system available in the sub-basement to remove water and alert operators to the condition is the existing sump dewatering system. This sump dewatering/alarm system is comprised of commercial grade components possessing no environmental qualification and, therefore, cannot be relied upon to function in a harsh environment. Consequently, the sub-basement could flood in the event of a RHR pump seal failure and preclude the RHR pumps from performing their intended safety function.

It is required per 10CFR50.49 Section B(2) that non-nuclear safety electric components whose failure under postulated environmental conditions could prevent satisfactory accomplishment of a safety function be included in the licensed facility's environmental qualification program. The components of the existing sump dewatering system located in the Auxiliary Building Sub-Basement are not currently included in the Ginna Station Environmental Qualification Program.

The basic scope of this modification involves upgrading the sump pump motors (MO/ASP1A and MO/ASP1B) and associated float type level control switches (LS-2042 and LS-2043) to an environmentally qualified status. Minor modifications will also be made to the sump pump motor control circuitry to achieve a configuration for which the level switches controlling the pump motors provide sump level status (normal or high level alarm) on the Plant Process Computer System (PPCS). Currently, the level switches provide only the function of controlling the sump pump motors.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, previously evaluated in the Safety Analysis Report (SAR), is not increased as a result of this modification.

The probability of occurrence of an accident or malfunction of equipment important to safety is unrelated to the changes proposed in this modification. The equipment this modification affects has no failure modes that can lead to initiation or mitigation of an accident. The modified equipment is utilized to provide long term assurance of flooding protection for decay heat removal post LOCA, well beyond the time the accident is mitigated. The Auxiliary Building Sub-Basement dewatering equipment is physically unrelated to any system or equipment whose failure could cause an accident or transient.

The consequences of an accident or malfunction of equipment important to safety are not changed as a result of this modification. Should a failure occur which places a demand on the dewatering equipment, the equipment will function as is described in the SAR. Because this modification does not make any functional changes, the consequences associated with any accident or malfunction are previously analyzed.

The proposed modification does not create the possibility for an accident or malfunction of a different type than any previously evaluated in the SAR.

The changes made for this modification are not functional changes. This modification does not create any new interactions with safety related equipment, nor does it remove any existing ones. Because this modification does not change any interfaces with plant safety equipment, it is not possible for it to create a new type of accident or malfunction.

The margin of safety as defined in the basis for any Technical Specification is not reduced as a result of this proposed change.

The Auxiliary Building Sub-Basement dewatering and leakage detection equipment are not utilized as the basis for any Technical Specification.

SEV-1042
STEAM GENERATOR LOWER PLATFORM MODIFICATION

EWR #10216 is a request to design, procure and install permanent steel grating platforms around each of the two steam generators inside of the Containment Building at Ginna Station. The platforms will be constructed at the working elevation of the reactor coolant pump platforms to facilitate access to the steam generator handholes used for camera inspection and water lancing of the tube sheets. The platforms will cover the entire SG cavity at that level.

The addition of these platforms does not increase the probability of an occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in that the installation and design requirements ensure that the structures cannot interact with equipment important to safety.
The results of the design analysis demonstrates that the installation of the two (2) steel grating platforms does not create the possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR in that there are no active functions performed and, therefore, the passive functions cannot initiate any accidents or malfunctions.

The addition of these platforms does not reduce the margin of safety as defined in the basis for any Technical Specification in that the installation of these platforms is not within the scope of the Technical Specifications.

Therefore, the addition of these platforms does not invalue any unreviewed safety question.

SEV-1043

CLARIFICATION OF RULES OF USAGE FOR CRITICAL SAFETY FUNCTION STATUS TREES FOR INTERMITTENT INDICATIONS

This document evaluates the proposed change to procedure A-503.1, rev. 12, "Emergency and Abnormal Procedures Users Guide", step 3.4.1.8 as proposed in PCN 94-3558. Specifically this PCN changes the rules of usage for Critical Safety Function Status Trees (CSFST's) so that Function Restoration (FR) procedures are not entered for ORANGE or RED priority conditions that come in and clear intermittently. This is different from the current procedure only for the INTEGRITY Status Tree, where currently the FR procedure is entered even for intermittent ORANGE or RED conditions.

Will the probability of occurrence or the consequences of any accident or malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

The procedure change evaluated here does not increase the probability of failure of any equipment important to safety. By allowing for the most optimum recovery for accidents or transients the change may reduce the potential consequences of accidents evaluated in the UFSAR.

Will the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR be created?

The proposed procedure change does not increase the probability of any system failure that could initiate an accident, and therefore the possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR cannot be created as a result of this change.

Will the margin of safety as defined in the basis for any Technical Specification be reduced?

The proposed change does not affect the basis of any Technical Specification as discussed in section 5.3. Therefore the margin of safety is not reduced.

SEV-1044
RE-CONFIGURATION OF THE PRESSURIZER MISSILE SHIELD BLOCKS

The design of the proposed modification addressed in this Safety Evaluation is documented in EWR 10259, "Pressurizer Blocks Tie Down."

The pressurizer cubicle is enclosed on the top by three concrete slabs. The purpose of the slabs is to protect vital equipment in containment from the effects of internally generated missiles and released high energy fluid or steam should a piping failure occur within the pressurizer cubicle. The primary missiles associated with the pressurizer compartment top blocks are valves and parts of valves, specifically from the power operated relief valves (PORVs), their motor operated block valves and the pressurizer safety relief valves.

The modification proposed under EWR 10259 is to reconfigure the placement of the slabs such that the pressurizer cubicle does not have 100% of the available space covered.

The proposed reconfiguration is engineered to accomplish two objectives. The first is to allow ingress to the cubicle without requiring missile shield slab movement. The second is to allow air flow through the pressurizer cubicle.

The change proposed for this modification satisfies these two objectives while maintaining the original function of the slabs. After the proposed reconfiguration the pressurizer cubicle missile shield slabs will still protect vital containment equipment from the effects of a high energy line break in the pressurizer compartment.

The basic scope of the modification involves the repositioning of the slabs such that one slab is placed on top of, and perpendicular to, the other two. The top slab is positioned over the potential missiles that are not impeded by the bottom slabs.

After the slab positions are reconfigured the slabs will be restrained against the possible steam jet thrust forces that could develop should a pipe break occur in the pressurizer cubicle.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, previously evaluated in the Safety Analysis Report (SAR), is not increased as a result of this modification.

The equipment this modification affects is used to ensure that, should a LOCA occur, the features necessary to mitigate the consequences of that accident survive the accident unaffected by the jet thrusts and missile generated as a result of that accident.

The reconfigurement of the slabs does not detectably lower the level of protection they supply. It can therefore be concluded that the change will not alter the consequences of any accident.

The probability of LOCAs is independent of the features used to mitigate the consequences of the accident. Because the missile slabs can not cause an accident they can not increase the probability an accident occurs. Their function is simply to limit the potential consequences of an accident.

The proposed modification does not create the possibility for an accident or malfunction of a different type than any previously evaluated in the SAR.

The changes made for this modification are not functional changes. The original concrete slabs are being reconfigured from three blocks lying side by side to two blocks side by side with one block overlaying on the top, perpendicular to the bottom ones. This configuration reduces coverage of the top of the pressurizer compartment from 100 percent to approximately 75 percent. Although total coverage is reduced because the blocks are strategically placed over all potential missiles the ability of the slabs to interrupt the trajectory of potential missiles has been preserved.

The slabs restraints ensure that jet forces can not lift the blocks. Therefore during a LOCA in the pressurizer cube they will not be moved into a position where they could drop into the pressurizer compartment and exacerbate the accident. The restraint system will also preclude slab movement during seismic events, thus removing the potential for the slabs to fall and cause an accident (LOCA).

This modification does not create any new interactions with safety related equipment, nor does it remove any existing ones. Because this modification does not change any interfaces with plant safety equipment, it is not possible for it to create a new type of accident or malfunction.

The margin of safety as defined in the basis for any Technical Specification is not reduced as a result of this proposed change.

The pressurizer missile protection scheme is not utilized as the basis for any technical specification.

SEV-1045
CONTAMINATED STORAGE BUILDING - USE EVALUATION

EWR #10279 is a request to evaluate proposed changes to the use of the Contaminated Storage Building (CSB). The changes to the use of the CSB consist of initiating inventory control of contaminated tools and supplies, providing storage for tool cabinets dedicated for use only within Containment, and the use of one portion of the floor plan in the southwest corner as a designated decontamination area.

The proposed change of use does not increase the probability of an occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in that this change has no interfaces with equipment important to safety.

The changes do not create the possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR in that any possible events are bounded by current analyses.

The change of use does not reduce the margin of safety as defined in the basis for any Technical Specification, in that the CSB is not within the scope of Technical Specifications.

Therefore, the change of use of the CSB does not invalue any unreviewed safety question.

SEV-1046

DELTA T RUNBACK LOGIC CHANGE

This document evaluates the modification to the Delta T Runback Logic circuitry. Currently, Over Temperature delta T (OTDT) and Overpower Delta T (OPDT) alarm relay logic for turbine runback is 1 out of 4 channels, and this relay logic causes spurious turbine runbacks due to single channel failures. This TSR will change the OTDT and OPDT alarm relay logic to 2 out of 4 channels thus eliminating spurious turbine runbacks due to single channel component and power failures. In addition to the alarm relay logic change, the Rod Drive Control System, both the Block of Manual Withdrawal and Stop Auto Rod Withdrawal schemes will be changed from a 1 out of 4, to a 2 out of 4 logic scheme. To accomplish the change to a 2 out of 4 logic scheme, it will be necessary to rewire 16 BF66F relays (a total of 120 relay contacts), no new relays are required.

The proposed modification will not increase the probability of occurrence of an accident previously evaluated in the UFSAR because the modification does not effect any components that could initiate an accident. This modification eliminates spurious turbine runbacks due to a single channel failure; it can not cause a turbine runback.

The proposed modification will not increase the consequences of an accident previously evaluated in the UFSAR because there are no accidents evaluated in the UFSAR that may have their radiological consequences altered as a result of this modification.

The proposed modification will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR because the modification removes a spurious turbine runback initiation due to a single channel failure. This reduces unnecessary transients on equipment and could reduce the probability of equipment malfunction.

The proposed modification will not increase the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR because this modification does not adversely alter the performance of any engineered safety feature as assumed in the accident analysis. Consequences of a malfunction of equipment important to safety is reduced by this modification because unnecessary spurious turbine runbacks will be eliminated due to single channel failures.

The proposed modification will not create a possibility for an accident of a different type than any evaluated previously in this UFSAR because this modification does not affect any accident initiators. This modification is a relay logic change only, it prevents a turbine runback due to single channel failures.

The proposed modification will not create a possibility for a malfunction of equipment of a different type than any evaluated previously in this UFSAR because there are not failure modes of a different type that are created by this modification. This modification prevents initiation of a turbine runback due to a single channel failure only.

The proposed modification will not reduce any margin of safety as defined in the basis of any Technical Specification because the overpower and overtemperature delta T setpoints will not be affected by this modification. The only changes involved with this modification are that a turbine runback initiation will require actuation of two overpower and overtemperature delta T channels.

SEV-1047
TEMPORARY INSTALLATION OF PROTOTYPE RCS METER FOR TEMPERATURE AND FLOW MEASUREMENTS

The design of the proposed temporary modification addressed in this Safety Analysis is documented in Temporary Modification Package 95-010 "Installation of Prototype RCS Motor", Rev. 0.

The system proposed for installation is an experimental prototype whose purpose is to demonstrate the feasibility of using acoustic technology to measure fluid flow and temperature. Should the technology prove viable it is expected that it will lead to an improvement in the accuracy of calorimetric calculations.

This proposed modification installs ultrasonic transducers on the hot and cold legs of both reactor coolant loops (Figure 1). Basically, a strip of the reactor coolant system (RCS) insulation will be removed, then at each metering location, the transducers will be mounted to the pipe using special-purpose stainless steel fixtures similar to the one shown in Figure 3. Each fixture is a clamp made from two 1/4" x 6" stainless steel bands and tightened by two threaded studs. Stainless steel pads for mounting transducer assemblies are welded to the bands at 45°F intervals. Transducer assemblies are bolted to the mounting pads and pressed against the pipe with a force of approximately 1000 lbs. On each hot leg there will be one fixture for temperature measurement (cross-path) transducers. Each cold leg will have three fixtures: one for cross-path transducers, and two for flow measurement (diagonal-path) transducers. The transducers are mounted on ceramic wave guides. Zinc foil is used to acoustically couple the wave guide to the RCS piping. The special-purpose fixtures hold the wave guide/transducers units in place. For the hot leg units the weight of the fixtures and transducers is less than 275 pounds for the cold leg units the weight is less than 300 pounds. Prior to plant return to power the RCS piping insulation around the assemblies is replaced.

The twenty eight transducers send and receive signals from two electronic units (one for each loop) mounted on existing unistruts outside the primary shield wall. The electronics units communicate outside containment via existing containment telephone jacks. The transducers and electronics units communicate via teflon jacketed twin axial cables installed as part of this modification. Two telephone cables from the existing phone jacks to the electronics units will also be installed.

The electronics units are powered from non safety 120 VAC containment convenience outlets.

The data acquired from the proposed modification will not be used for safety related on safety significant purposes. Should data from this modification be considered for future applications at Ginna a new safety evaluation will be performed which examines the data uncertainties associated with this modification.

The temporary modification described in this safety evaluation will be used to gather RCS flow and temperature data using non invasive acoustic technology. As detailed in Section 5.2, functional impact, the equipment being installed has no failure modes which can lead to the initiation of an accident. Likewise the new equipment has no interaction with equipment important to safety.

The equipment proposed for installation under this temporary modification has been designed to operate, without functionally interacting with equipment important to safety. Because this modification can not have any influence on safety related equipment completion of this change will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR.

The proposed modification does not create the possibility for an accident or malfunction of a different type than any previously evaluated in the SAR.

The changes made for this modification are not functional changes. This modification does not create any new interactions with safety related equipment, nor does it remove any existing ones. Because this modification does not create or change any interfaces with plant safety equipment, it is not possible for it to create a new type of accident or malfunction.

The margin of safety as defined in the basis for any Technical Specification is not reduced as a result of this proposed change.

The temporary prototype RCS flow and temperature metering equipment is not related to the basis for any Technical Specification.

SEV-1048

HOUSE HEATING STEAM SYSTEM PIPE REMOVAL IN CONTAINMENT AND INTERMEDIATE BUILDING, CAPPING PENETRATIONS 301 AND 303

The House Heating Steam (HHS) system valve 6151 has developed excessive leakage and has allowed steam to pass to the out-of-service Containment heaters. Valve 6151 is obsolete and difficult to repair due to lack of spare parts and steam piping isolation. The portion of HHS system supplying steam to the Containment heaters was only used during the Construction phase of Ginna and serves no operational or maintenance use today or in the projected future.

The piping and piping components will be removed. Sections of piping in the intermediate building and Containment will be permanently removed. Penetrations 301 and 303 will be sealed with a welded cap and redesignated as spares.

Containment isolation is currently maintained with redundant manual locked closed valves (1": 6152 and 6175, 2": 6151 and 6165). The removal of the process piping eliminates any possibility of Containment boundary breech. The manual locked closed valves will removed and the remaining process pipe will be sealed with a welded cap.

Containment isolation will be maintained after the process pipe removal by permanently welded ASME Class MC installed caps on the penetrations (301 and 303). This will, in effect, turn the once process pipe into an extension of the Containment.

The proposed change does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report in that the purpose and function of the penetration remain the same and its ability to accomplish the purpose and function remains the same.

This change does not create the possibility for an accident or malfunction of a different type than any evaluated in the safety analysis report in that the purpose and function of the penetration remain the same and its ability to accomplish the purpose and function remains the same.

This change does not reduce the margin of safety as defined in the basis for any Technical Specification in that the ability of the penetration to accomplish its purpose and function remains the same.

SEV-1049

TEMPORARY USE OF TE-450 AS A SUBSTITUTE FOR TE-402B IN THE T_{Avg} REACTOR PROTECTION CIRCUITRY

The proposed temporary modification addressed by this safety evaluation entails substituting a thermowell mounted Reactor Coolant System (RCS) cold leg resistance temperature detector (RTD), TW-450, for a direct immersion cold leg RTD, TE-402B, for use in the "A" RCS loop average coolant temperature (T_{Avg}) and delta T calculation instrumentation and control circuitry.

The proposed change will result in a configuration different then the configuration detailed in the Safety Analysis Report (SAR). The SAR credits the direct immersion RTDs as being high accuracy, fast response devices providing the T_{Avg} and delta T signals necessary for various reactor control and protection functions.

After the proposed temporary change one loop "A" cold leg RTD (TW-450) will have a slower response than the direct immersion RTD (TE-402B) described in the SAR.

This Safety Evaluation examines whether or not this proposed change constitutes an unreviewed safety question and consequently whether the proposed change is inimical to public interest.

The Reactor Protection system is currently being operated with the affected RCS temperature channel in a tripped condition. This configuration changes the protection logic such that if another channel is tested, faults, or receives an actual safety signal the reactor will scram.

The probability of occurrence of an accident has not increased due to substituting RTDs. The pressure boundary has not changed. There is no other characteristic of the RTDs that can cause an accident..

The RTD substitution can not cause malfunction of equipment important to safety because the RTD is electronically identical to the RTD it is replacing.. The only difference is well mounted vs. immersion.

The consequences or an accident are not increased because the replacement RTD causes an earlier trip on the limiting accident.

The RTD substitution can not cause an accident of a different type or different malfunction because the RTDs are electronically identical. The RTD only measures temperature, measuring temperature can not cause an accident. Failure of the RTD has been accounted for in the plant design and is not a new event.

The consequences of the accidents have not increased. The accident criteria have not been exceeded. Therefore, the margin of safety has not been reduced.

CONCLUSIONS:

All of the above were reviewed by the PORC committee with respect to the Technical Specifications and the committee has determined that no Technical Specification changes or violations were involved.

Additionally, these changes were reviewed in committee to determine if they presented an Unreviewed Safety Question and the general summations of these reviews are as follows:

1. These changes do not increase the probability of occurrence, or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the UFSAR, because:

   These changes were performed to ensure continued operability/availability of plant equipment and will not result in any equipment being operated outside of its normal operating range. This results in continued operability/availability of equipment important to safety. These changes additionally will not result in a change of operating characteristics of equipment used in the transient/accident mitigation which precludes an increase in the probability of occurrence of an accident. Because these changes ensure continued availability of plant equipment, the limits shown in the Technical Specifications, and the assumptions of the safety analyses of the Updated Final Safety Analysis Report continue to be met. As a result there is no increase in the consequences of any presently postulated accident.

2. These changes do not create the possibility for a new or different kind of accident, or a malfunction of a different type from any accident previously evaluated in the UFSAR because:

   The changes do not present new failure mechanisms outside of those presently anticipated, and are bounded by the events contained in the Updated Final Safety Analysis Report.

3. The changes do not reduce the margin of safety because:

   Present margins as contained in the Technical Specifications are valid, and these changes are performed within those limits. These changes will not result in violating the baseline assumptions made for equipment availability in the Technical Specifications and the Updated Final Safety Analysis Report.