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FACILITY CHANGES, TESTS AND EXPERIMENTS

CONDUCTED WITHOUT PRIOR APPROVAL

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UNDER FOR PROVISIONS OF 10 CFR 50.59

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

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SEV-1019
CONTAINMENT STRUCTURAL MODIFICATIONS

The following is a discussion of new changes to this safety evaluation which was previously transmitted as part of the 1995 10 CFR 50.59(b) submittal.

Containment Construction Openings

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

- Trolleys will be installed on the containment painter's monorail at Elevation 361'8". Once the plant has reached Cold Shutdown, a fabric dust collector, such as herculite, will be suspended from these trolleys to control the debris/dust generated during the concrete chipping operation from reaching the lower elevations of containment.

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:

- The fabric dust collector will be suspended from the monorail trolleys and hoisted to the approximate elevation of the painter's monorails (El. 361'8").

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:

- A concrete pump and stationary delivery line will be installed to deliver replacement concrete to the dome area. The concrete pump line will be routed up the containment from ground level to the construction openings.

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:

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

- The fabric dust collector suspended from the painter's monorail at Elevation 361'8" will be lowered and the trolleys and dust collector will be removed from containment.

To facilitate installation of the fabric dust collector, trolleys will be installed on the existing painter's monorail at Elevation 361'8" prior to Cold Shutdown. Pick boards and ladders will be installed on the bridge crane scaffold support steel to access the monorail. To preclude any adverse impact due to a seismic event during the period when the pick boards and ladder are in place, these items will be installed to satisfy the requirements of RG&E Procedure A-1406.1. In addition, during installation, the pick boards, ladder or trolleys will not be carried directly over the steam generators. Movement of the bridge crane or trolley to facilitate installation of the trolleys on the painter's monorail will be performed in accordance with existing station procedures. Installation of the fabric dust collector will occur after the plant reaches Cold Shutdown.

Interfering Commodities

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. In addition, flanged connections may be installed at the piping cut locations to facilitate reinstallation of the spray piping.

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. The ducting may either be severed and removed from the liner plate prior to liner plate removal or the duct section may remain attached to the liner plate during rigging operations.

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. The containment spray piping, supports and the containment liner have been evaluated for the flange addition. The results of these evaluations have demonstrated that code allowables have not been exceeded. This ensures that the systems are available during plant modes in which they may be expected to perform their safety function.

Dome Access Manlifts and Liner Plate Attachments

To provide access to the containment dome openings and interfering commodities, manlifts will be temporarily installed on the bridge crane scaffold support platform. To secure the manlifts, support steel will be installed on the bridge crane scaffold support platform. The manlifts and support steel will be installed, used and removed only with the reactor in a defueled condition.

To provide dome access in addition to the manlifts, lugs will be welded directly to the containment liner plate to allow attachment of hanging scaffolding. The lugs attached to the liner plate will be installed in the defueled condition. The hanging scaffolding will be installed, utilized and removed in the defueled condition. Prior to the start of refueling, the hanging scaffolding will be removed from containment and the liner attachment lugs will be abandoned in place.

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 manlifts, the manlift support steel, the hanging scaffolding and the liner attachment lugs will be installed once the reactor vessel is completely defueled and essential safety-related systems isolated from containment. The manlifts, manlift support steel and hanging scaffolding will be removed prior to fuel reload.

All heavy load movements during installation and removal of the manlifts, the manlift support steel, the liner attachment lugs and the hanging scaffolding will satisfy the requirements of RG&E Administrative Procedure A-1305. With the plant in the defueled condition and essential safety-related systems isolated from containment, the impact of a construction incident is minimized.

The manlifts and support steel, as well as the liner attachment lugs and hanging scaffolding have been evaluated and have been determined to have no adverse structural impact on either the containment bridge crane or the containment liner plate.

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

SEV-1020
STEAM GENERATOR RIGGING AND HANDLING

The following is a discussion of new changes to this safety evaluation which was previously transmitted as part of the 1995 10 CFR 50.59(b) submittal.

Haul Routes

An erection crane (Manitowoc 4100W tower crane) will be assembled North of the Service Building to support the installation of a construction crane. The assembly and movement of the erection crane into the operating location may be performed during any mode of reactor operation. This section of the Safety Evaluation addresses the attachment of the tower/boom assembly to the erection crane crawler, raising of the tower assembly, and the movement of the erection crane into its operating location.

The impact of an erection crane boom failure on the Screen House or the buried service water piping does not adversely affect the safety related function of the systems.

Temporary Cranes and Rigging Components

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

- Assemble the construction crane in the Southwest bay of the Turbine Building (may be performed in all modes of plant operation).
- Stage material in the alleyway (to include movement of the equipment hatch personnel airlock) to support the steam generator replacement outage (may be performed with the plant in Cold Shutdown, Refueling or Defueled modes of plant operation).

Prior to Cold Shutdown, a temporary erection crane (Manitowoc 4100W tower crane) will be assembled and positioned North of the Service Building and the erection crane boom will be raised. This crane will be used to install a construction crane (Liebherr 281HC climbing tower crane) in the Southwest bay of the Turbine Building. Prior to erection of the crane, a construction opening will be created in the Southwest bay of the Turbine Building roof. This opening will be approximately 15 feet by 22 feet and will involve the removal of roofing membrane, insulation, decking, roof purlins, and the horizontal diagonal bracing at the bottom of the Turbine Building roof truss. The crane tower will be supported on the operating floor of the Turbine Building. In this mode of plant operation, the temporary crane tower will be constructed to an elevation above the turbine building roof and the jib will be constructed and attached to the tower. In this position, the jib will be oriented in the East/West direction with the main jib over the Service Building. The construction crane will be lashed off to prevent rotation of the jib under high wind loads or collapse of the crane into the Facade Structure during a seismic event. The construction crane will remain in the storage configuration until the plant reaches Cold Shutdown. Once the plant has reached the Cold Shutdown condition, the self-climbing assembly on the construction crane will be used to raise the crane to its required operating elevation by installing additional tower sections.

After the plant reaches Cold Shutdown and the construction crane has been raised to its operating elevation, the crane will be used to rig material to and from the containment dome and the facade during the steam generator replacement outage including the liner plate sections. After completion of all Steam Generator Replacement construction activities requiring use of the construction crane, and prior to Cold Shutdown, the self-climbing assembly will be used to lower the crane assembly to below the top elevation of the facade. During or after Cold Shutdown, the temporary erection crane (Manitowoc 4100W tower crane) will be positioned North of the Service Building and will be used to disassemble the construction crane.

To allow movement of material into the containment, a material handling system will be installed in the alleyway and extend into containment. This material handling system will consist of material handling carts which will travel into and out of containment on a rail system. The existing material track system used during normal refueling outages will be modified to extend further east into the alleyway and further into containment. The material handling system will be installed once the plant reaches Cold Shutdown and, with the aid of mobile construction cranes, will be used to stage material into and out of containment during the steam replacement outage.

Installation of the material handling system and staging of material in the alleyway in support of the steam generator replacement outage will be performed with the plant in a cold shutdown, refueling or defueled mode. Use of the material handling system will occur during the Cold Shutdown or defueled mode.

The following activities may be performed in a mode above Cold Shutdown:

- A construction opening will be created in the Southwest bay of the Turbine Building roof which will allow the construction crane tower to be supported on the operating floor of the Turbine Building.

- The boom of the 4100W erection crane will be raised into the operating position.

- The temporary construction crane tower will be assembled to an elevation above the Turbine Building roof, where the jib will be constructed and attached to the tower.

Once the plant has reached the Cold Shutdown condition, the self-climbing assembly on the construction crane will be used to install additional tower sections to raise the crane to the required operating elevation. During Cold Shutdown, Refueling, and the defueled mode of reactor operation the construction crane will be used to rig material to and from the containment dome and the facade, to include the liner plate sections.

After completion of all Steam Generator Replacement construction activities requiring use of the construction crane the construction crane will be disassembled. Disassembly may be performed during any mode of plant operation. Should disassembly of the construction crane occur in a mode above cold shutdown, prior to leaving Cold Shutdown, the self-climbing assembly will be used to lower the crane assembly to below the top of the facade structure. The temporary erection crane will be positioned North of the Service Building and will be used to disassemble the construction crane.

With exception of the construction crane, the lifting and handling activities addressed in this safety evaluation are performed with temporary cranes which have no direct interface with permanent plant systems or components.

The mounting of the construction crane within the turbine building, and the evaluation of the turbine building structure for the loads resulting from this installation, have been evaluated in accordance with the existing design basis and have been determined to have no adverse impact on the existing structure. In addition, erection, storage and operation of the construction crane has been evaluated to ensure that, in the inadvertent event of a failure during installation, storage or use, no adverse impact to safe shutdown or decay heat removal equipment occurs.

For the material handling system used to stage material into and out of containment, the system has been designed to allow reinstallation of the equipment hatch personnel lock or closure of the equipment hatch roll-up door, should containment closure be required.

SEV-1029

INSTRUMENT SETPOINT CHANGES ASSOCIATED WITH STEAM GENERATOR REPLACEMENT AND REDUCED RCS AVERAGE TEMPERATURE

Following the 1996 Refueling Outage, Ginna will operate at a reduced full power RCS Average Temperature (RCS Tavg) (EWR 10154), with replacement steam generators (EWR 10034) and changes to the main feedwater control valves (EWR 10311). These modifications result in several control/protection system setpoint changes. This safety evaluation addresses the acceptability of the following changes:

Programmed RCS Tavg (Tref) is controlled as a function of plant power as determined by turbine impulse chamber pressure. It currently varies linearly from 547°F at 0 psig (0% power) to 573.5°F at 489.2 psig (100% power). This will be modified to vary linearly from 547°F at 0 psig (0% power) to 561.0°F at 489.2 psig (100% power).

Programmed pressurizer level is a function of RCS Tavg. It currently varies linearly from 19.5% at 547°F (0% power) to 49% at 573.5°F (100% power). This will be modified to vary linearly from 35% at 547°F (0% power) to 50% at 561°F (100% power).

Reactor Coolant System flowrate is monitored by differential pressure producing elbow taps in the loop crossover legs. The flow indicators are calibrated to read 100% flow for 100% power conditions. Since 100% power conditions change, these calibrations change.

The condenser steam dump system actuates on, among other signals, the difference between actual RCS Tavg and Tref. Due to the changes in Tref (see 1.1) a change in the deviation deadband without turbine trip (TM-401I) from 6°F to 5°F is being made, and the RCS Tavg input lead time constant is changed from 16 seconds to 20 seconds.

The following changes to the feedwater control system (ADFCS) are being made:

Steam generator narrow range water level setpoint currently ramps from 39% at 0% power to 52% at 20% power, and remains constant at 52% to full power (power as sensed by turbine impulse pressure). The setpoint will be changed to a constant 52%.

The no load steam generator wide range level setpoint changes from 340" to 347" (65.5% to 66.8%), and the gain from 3.0%/% to 9.0%/%.

The main feedwater control valve demand curve, and total demand curve change as a result of the resized valves.

The lag time on the turbine first stage pressure input changes from

60 seconds to 5 seconds.

OPdT and OTdT setpoints are based in part on a nominal full power RCS Tavg. This changes from 573.5°F to 561°F.

Various inputs to the PPCS and SAS change to reflect the above changes.

The high Tavg alarm setpoint (MCB Annunciator) is being changed from 578°F to 566°F.

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 instrument setpoint changes evaluated here do not increase the probability of failure of any equipment important to safety. By maintaining the design basis of all systems the potential consequences of accidents evaluated in the UFSAR are unchanged.

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

The proposed instrument setpoint changes do 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. Therefore the margin of safety is not reduced.

SEV-1052
RWST MODIFICATIONS

The change involves modifications to the Auxiliary Building structure and the Refueling Water Storage Tank (RWST) as summarized herein.

The proposed modifications are planned to be performed either during normal plant operation, or during a refueling outage.

The modification will remove an existing block wall fire barrier on the Auxiliary Building operating floor level between the RWST and the 480-V electrical system bus 14 equipment. The Appendix R Conformance Verification Checklist associated with this modification assess the acceptability of this removal.

A 6-inch high reinforced concrete curb extending around the edge of the RWST floor opening on the Auxiliary Building operating level, a design feature of the original plant, will be removed. The design basis for the curb was determined, and the acceptability of its removal, was assessed.

The modification will add a total of 16, equally spaced, vertical stiffeners to the exterior of the RWST. Each stiffener is made up of a pair of structural steel channel sections, placed end-to-end and joined with a splice weld. Each channel section is 12 inches wide and 12'-4" long weighing approximately 370 pounds (approximately 12,000 lbs. total). Each will be rigged and suspended in place while being welded to the RWST.

The modifications will add a welded steel support skirt extending 360 degrees around the perimeter of the RWST at elevation 271'-0" (near the vertical mid-point of the tank). The ring will also be attached to the Auxiliary Building operating-level floor slab via at least forty 3-inch diameter steel pins, set into 12-inch deep drilled holes, equally spaced around the RWST perimeter. The completed support will provide the RWST with improved horizontal support against seismic loads.

The structural aspects of the modification design are specified to be in accordance with the generic Design Criteria for Ginna Station Structural Modifications /Additions/ Evaluations.

In accordance with the generic structural Design Criteria the placing of unprotected structural steel (the welded steel support ring) in a fire load zone has been assessed. The placement of kaowool in the gap meets this criteria.

The evaluated design of the 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 alter only the pressure-boundary of the RWST (while preserving its pressure-retaining function) and are independent of accident mitigation features of the SI, CS, CVC and RHR Systems which draw water from the RWST.

The evaluated design of the 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 boundary of the RWST, and are independent of accident mitigation features of the SI, CS, CVC and RHR Systems which draw water from the RWST.

This modification will not reduce the margin of safety as defined in the bases for any technical specifications. The modification design does 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 design does not involve an unreviewed safety question.

SEV-1054
AUTOMATIC LUBE OIL COOLING FOR TURBINE DRIVEN AFW PUMP AND PIPING UPGRADE FOR MOTOR DRIVEN AFW PUMP LUBE OIL COOLER

This modification will install automatic initiation of Service water cooling to the Turbine Driven Auxiliary Feedwater Pump Lube Oil Cooler. Automatic cooling will be accomplished by installing a 3-way thermostatic valve on the oil side outlet to the TDAFW pump lube oil cooler. A new bypass line will be installed from the cooler inlet to port B on the 3-way valve located on the outlet of the cooler. The cooler outlet will then be connected to port C of the 3-way valve and port A(mixing port) will supply L/O to the common discharge line going to the turbine inboard and outboard bearings. The TDAFW pump bearings are cooled by self contained oil reservoirs with the thrust bearing cooled by a separate SW feed which will remain in-service. The 3 way valve will bypass oil around the cooler via port B when oil temperature is less than 120 F. As the temperature increases above 120 F, the valve will throttle lube oil through the cooler via port C proportional to the temperature differential. Ports B and C are actuated by a plug which expands on increasing temperature. The valve plug is filled with a specially formulated wax which melts at the desired set point causing the plug to expand.

All or portions of the TDAFW lube oil skid's pickled carbon steel piping will be replaced with stainless steel (SS) piping to facilitate replacement of leaking threaded joints and avoid having to procure pickled piping. Pickling of the carbon steel was done to remove mill scale from the internal diameter of the pipe in order to avoid particles in the oil. SS piping does not have mill scale.

Several corrosion related failures have occurred in the copper tubing used for Service Water cooling to the motor driven AFW pump lube oil coolers. This modification will also upgrade the piping material to stainless steel(SS) tubing for enhanced corrosion resistance in a mild chlorinated environment. The SS material will also reduce galvanic corrosion of the SW carbon steel piping since SS has a lower electrical potential compared to copper. In order to facilitate the use of compression tubing fittings and procurement of qualified valves, all the associated maintenance isolation valves will be replaced with stainless steel ball valves and the instrument root valves will be replaced with stainless steel needle valves.

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 TDAFW pump automatic L/O cooling or SW piping for the motor driven AFW pumps will not affect any fire protection features nor will it increase any combustible loading. No other accidents are initiated by the AFW system. The automatic L/O cooling will not affect the TDAFW performance or ability to remove decay heat from the reactor.

The new components are all located on the L/O skid do not require electrical power to operate. Increasing the normal standby L/O temperature for the TDAFW turbine bearings will not effect its ability to operate with a loss of SW since the oil reaches a steady state temperature lower than the maximum allowable bearing operating temperature which is independent of the initial oil temperature. Consequently, the system will retain its ability to be AC independent for at least 2 hours and isolatable from the control room. The SW piping material upgrade and valve replacements to the motor driven AFW pump lube oil coolers will not affect pump or system performance. Therefore, the consequences of an accident or malfunction of equipment will not be changed.

The possibility for an accident or malfunction of a different type than evaluated previously in the UFSAR will not be created by the proposed modification. The automatic L/O cooling valve is similar in design to the existing L/O cooling bypass valve in both Diesel Generators. Both valves are manufactured by AMOT corporation. The only difference between the valves is their size, body material, and set point. The new bypass piping will utilize equivalent materials of construction as existing equipment in the AFW system and the construction and design standards utilized will be equivalent or better than those used in the original design of the system. The new 3-way valve is considered an active control valve, similar to any motor operated valve. The new valve has the potential for an active failure such that the temperature element does not allow flow through the L/O cooler. A single active valve failure for the AFW system, such as the case of an MOV, has already been considered. The new cooling method does introduce any new types of equipment or materials to the plant or operate any system differently than previously analyzed.The SW piping material upgrade and valve replacements to the motor driven AFW pump lube oil coolers will also use materials and equal codes and standards of construction that are presently used in the SW system.

The margin of safety as defined in the basis of any Technical Specification is not reduced by the proposed modification. The TDAFW automatic L/O cooling and the SW piping material upgrade and valve replacements to the motor driven AFW pump lube oil coolers will not affect the AFW pump performance or their ability to be tested.

SEV-1056

TURBINE TRIP SOLENOID VALVE TEST BLOCK INSTALLATION

In response to the Salem Turbine over speed trip failure, Westinghouse initiated AIB 9301 which included recommendations to allow for the (unit) on-line testing of the 20/ET and 20-1/AG, 20-2/AG turbine trip solenoid valves.

This modification will fabricate and install manifold type valve blocks and electrical control switches for testing and maintenance at the inlet and outlet ports of the turbine trip solenoid valves. This change will allow for periodic isolation and testing of the individual solenoid valve to verify their ability to relieve system pressure as originally designed without causing a turbine trip. This modification will also allow for on-line replacement of a failed solenoid valve.

Industry experience has typically shown that in this application, the spool/poppet type solenoid valve may fail to reposition on demand with infrequent use. The normal testing interval of once per year does not sufficiently provide adequate indication of operability.

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. Malfunction of the new test devices due to operator error or test device failure may slightly increase the probability of a turbine trip or loss of load. However, the probability for a malfunction of the these SOV's during periods while the Unit is on-line will be significantly reduced since these devices will provide the capability for more frequent schedule of testing. The benefits derived from the reduction in the malfunction of the turbine trip SOVs is judged to out weigh the potential increase in a turbine trip or loss of load. Consequently, the overall probability of a turbine protective feature malfunction will be decreased. While performing testing or maintenance, if a turbine trip and resulting loss of load were to occur, the consequences of such a malfunction would not be increased since a loss of load trip is mitigated by the Reactor protection system. The loss of load case in chapter 15 of the UFSAR accident analysis does not assume a reactor trip from a turbine trip signal. The Pressurizer high pressure signal is credited for tripping the reactor to mitigate the event. The subject modification does not effect the Reactor protection system.

The possibility for an accident or malfunction of a different type than evaluated previously in the UFSAR will not be created by the proposed modification. The new test devices meet the turbine EH system design requirements and utilizes material equivalent to those presently used in the system. The test devices will not effect any of the turbine protective features set points or control methods. The new test devices will remove one of the turbines trip SOVs during testing and maintenance. If an automatic turbine trip were called for during the test, the redundant SOV (to the one in test) would still provide the necessary protective feature. If a failure of the redundant device were also to occur, the turbine could experience damage due to over speed, high back pressure, or low bearing oil pressure. Each of these malfunctions have previously been analyzed and found to result in a potential for turbine missiles.Turbine missiles have already been analyzed in the Ginna UFSAR.

The margin of safety as defined in the basis of any Technical Specification is not reduced by the proposed modification. Technical Specification table 3.3.1-1 identifies stop valve closure and low auto stop oil pressure signals as an input to the Reactor Protection system to indicate a turbine trip. The isolation and testing of the trip solenoids will not block or bypass the stop valve closure limit switches or the low auto stop oil pressure signals. Per the bases of the Technical Specifications, the only required function is stop valve position detection (open or closed) and low auto stop oil pressure signal. Turbine trip initiation is not a Technical Specification requirement. Therefore the installation, testing and isolation for maintenance will not effect any Technical Specification margin of safety.

SEV-1057
18 MONTH FUEL CYCLE

For economical operation of an 18 Month Fuel Cycle core peaking factors needed to be increased. This requires reanalysis of several of the UFSAR Chapter 15 transients. Since the transients were being reanalyzed and the steam generators are being replaced it is appropriate to include the characteristics of the new steam generators in the analysis. Since the new steam generators (RSGs) produce higher steam pressure operation at a reduced Tavg would be economically beneficial. Therefore, the changes being incorporated into the 18 Month Fuel Cycle are:

- increased core peaking factors
- incorporation of the RSGs
- Tavg window

Increased peaking factors, Tavg window, and characteristics of the BWI SG do not affect the probability of occurrence of an accident or malfunction. They are assumptions used in calculating the consequences of an accident. The Tavg window would allow operation at a Tavg of up to 15°F lower than the current Tavg. This small reduction in Tavg does not affect the probability of an accident. The changes associated with this evaluation have been incorporated into the calculation of accident or malfunction consequences. The consequences meet the required acceptance criteria, thus the consequences are acceptable.

The nature of the changes addressed by this safety evaluation can not cause an accident or malfunction of a different type than previously evaluated. The changes only effect the consequences.

The changes addressed by this safety evaluation do not reduce the margin of safety as defined in the basis for any technical specification because the analysis of the accident consequences meet the required acceptance criteria. Since all acceptance criterion are met there is no reduction in the margin of safety.

SEV-1058

ADDITION OF AUTOMATIC TURBINE TRIP ON HIGH VIBRATION

The purpose of this safety evaluation is to determine if there are any unreviewed safety questions related to the work scope for this modification. This modification is necessary to assure that damage to the Turbine is minimized in the event of high, sudden, ramping of turbine vibration. In addition to the turbine trip function, the (2) turbine supervisory monitoring racks and the reactor coolant pump (RCP) vibration monitoring rack are being updated.

The existing configuration of the turbine supervisory instrumentation and RCP vibration monitoring consists of the following:

1) Bently Nevada 7200 Turbine Supervisory rack (top), safety class NS, consisting of a power supply and eight monitors for turbine bearings #1 through #8.

2) Bently Nevada 7200 Turbine Supervisory rack (bottom), safety class NS, consisting of a power supply and eight monitors for turbine bearing #9, differential expansion (generator end), differential expansion (governor end), case expansion, thrust position and rotor eccentricity.

3) Bently Nevada 7200 RCP Vibration rack, safety class NS, consisting of a power supply and four monitors for RCPA shaft vibration, RCPA seismic vibration, RCPB shaft vibration and RCPB seismic vibration.

When any of the turbine bearing vibration levels reach 7 mils, annunciator I-27, Rotor Eccentricity or Vibration alarm is lit. If bearing vibrations reach 14 mils or greater (15 mils for bearing #9 only), the turbine is manually tripped. If reactor power is >50%, or >P7 and steam dump is not available, a turbine trip will automatically initiate a reactor trip.

This modification shall modify this configuration as follows:

1) Bently Nevada 7200 Turbine Supervisory racks (2), and RCP Vibration rack shall be replaced by Bently Nevada 3300 series racks. The 3300 series is designed to be fault tolerant and highly immune to EMI effects. Common faults in the monitors and field equipment such as short, open and grounded circuits do not generate signals likely to produce false alarms in this equipment. Transient or rapidly varying faults, such as EMI, are mitigated by the Timed OK/Channel Defeat circuits inherent within each vibration monitor. This allows the monitor to differentiate between a transient fault (eg. EMI) which is not allowed to produce an alarm, and an actual high vibration amplitude which will be allowed to produce an alarm.

2) An automatic turbine trip function will be added, based on a high vibration signal from any of the eighteen bearing transducers (two transducers per bearing). Bently Nevada research and field experience has shown that machine damage can occur due to excessive vibration in one plane. Therefore, AND logic (2/2) is "not only unnecessary" but " it is technically inappropriate" per Bently Nevada. The trip setpoint for all nine bearings will be 14 mils. Upon receiving a high vibration signal, for three seconds, the danger contacts of the bearing vibration relays will close thus energizing the Turbine Auto Stop Trip Solenoid (20/AST), the Turbine Emergency Trip Solenoid (5501S3), annunciator ANN-K10 and transmitting a turbine trip signal to the Plant Process Computer System (PPCS). Turbine Supervisory racks will be upgraded to safety class SS. A first out annunciator will be added at ANN-K10 for automatic turbine trip on high vibration. A key block switch will be installed to bypass the trip function for maintenance and testing purposes, however, a turbine trip signal will still be sent to annunciator ANN-K10 and the PPCS. Spare annunciator ANN-J19 will be engraved and wired to indicate when the block switch is in the blocked position. The ANN-I27 annunciator function, and automatic reactor trip if reactor power is >50%, or >P7 and steam dump is not available, will remain unchanged.

The proposed modification will not increase the probability or the consequence of, any accident or malfunction of equipment important to safety. The proposed modification is not fed from any safety related power sources therefore, the installation would not cause a loss of power to 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 affects RCP and turbine vibration monitoring and initiates a turbine trip on high vibration 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 RCP vibration and Turbine Supervisory reliability and minimizes the probability of severe turbine damage due to sudden, high, vibration.

The margin of safety as defined by plant Technical Specifications is not affected by this modification. This modification adds another criteria for an automatic turbine trip (turbine bearing vibration ≥14 mils). Section 15.2 of the Ginna UFSAR discusses accident analysis with respect to a decrease in heat removal by the secondary system.

"A" DIESEL GENERATOR SOV 5907A DISABLE

Fuel recirculation SOV 5907A for the 'A' Diesel Generator has failed periodic testing. A replacement solenoid valve is not readily available. This Temporary Modification will disconnect SOV 5907A from the 125 VDC control circuit, thus rendering it in the open position. 125 VDC feed conductors for SOV 5907A shall be insulated and taped back.

Level Transmitter LIT 2050A senses the 'A' Diesel Generator Fuel Oil Day Tank (TDG04A) level. While level is above setpoint, SOV 5907A is actuated open to recirculate from the Fuel Oil Transfer Pump (PDG02A) to the 'A' Diesel Generator Fuel Oil Storage Tank to provide (TDG01A) for pump protection. SOV 5907 is actuated closed to prevent over filling of the day tank. When the day tank level drops below setpoint, SOV's 5907 and 5907A "swap" states with 5907 opening and 5907A closing. This allows 100% of the fuel oil to flow to fill the day tank.

Since SOV 5907A has mechanically failed, it is postulated that further degradation could cause electrical failure. An electrical failure could cause failure of control fuses FUDGACP/1A1, thus disabling the entire 'A' Diesel Generator fuel oil day tank fill control circuit. This Temporary Modification will disconnect power leads at SOV 5907A, thus isolating the SOV from the control circuit and eliminating any possibility of fuse failure due to further coil degradation. De-energizing SOV 5709A will maintain the valve in an open position. If sufficient flow cannot be developed with SOV 5907A open, the upstream manual isolation valve 12398G will be throttled to provide the necessary flow restriction. The throttle position of 12398G will be set by closing the supply isolation SOV 5907 to the day tank and throttling 12398G to develop 35 psig discharge pressure on the Fuel Oil transfer pump. This will ensure sufficient recirculation without lifting the 40 psig relief valve 5959.

After disconnecting conductors at SOV 5907A and taping back, PT-12.6 and PCN 96-T-0019 will ensure that SOV 5907 functions normally and will ensure that the diesel day tank can be filled at greater than 2.62 GPM.

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 since the Fuel Oil transfer system will still be operated within its design parameters and no new equipment or modes of operation are being introduced

The possibility for an accident or malfunction of a different type than evaluated previously in the UFSAR will not be created by the proposed modification since no new equipment or materials are being introduced nor will the equipment be operated outside their design parameters.

The margin of safety as defined in the basis of any Technical Specification is not reduced by the proposed modification since the fuel oil transfer system will perform its intended function to deliver fuel oil to the day tank greater than the consumption rate.

SEV-1062

REVISION 1 TO TECHNICAL REQUIREMENTS MANUAL (TRM)

The purpose of this safety evaluation (SEV) is to evaluate proposed changes to the Ginna Station Technical Requirements Manual (TRM). These changes are the result of the first use of the TRM following the implementation of the Improved Technical Specifications (ITS) for Ginna Station. Specifically, this SEV will evaluate the following changes to the TRM:

a. Remove the requirement for entering LCO 3.0.3 if Required Actions are not met or provided. Instead, attempts to perform the specified Required Actions would continue until they are met.

b. Revise the Applicability requirements for the TR 3.1.2 (Boron Injection System - MODES 5 and 6) and provide alternative means for meeting the associated operability requirements. Specifically, charging will no longer be required in MODE 6 while credit for the reactor coolant system (RCS) boron concentration may be taken in place of requiring an operable charging system in MODE 5.

c. Revise the shutdown requirements for inoperable Circulating Water Flood Protection Instrumentation (TR 3.3.2) to allow use of a "continuous flood watch" in the event that inoperable channels are not restored to operable status within the required time limits.

d. Revise the Applicability of the refueling manipulator crane interlocks (TR 3.9.3) and revise the Required Actions to allow use of administrative controls in place of inoperable interlocks during refueling operations.

e. Revise the frequency for verifying flow through the reactor vessel head vents using liquid or gas (TSR 3.4.1.2) from every 18 months to every 36 months.

f. Remove the restriction on MODE changes if equipment is inoperable (LCO 3.0.4) or if required surveillances are not met (SR 3.0.4).

Operation of Ginna Station in accordance with the proposed changes does not involve an increase in the probability or consequences of an accident previously evaluated. Analyzed events are assumed to be initiated by the failure of plant structures, systems, or components. Revising the Required Actions for inoperable Circulating Water Flood Protection Instrumentation and refueling manipulator crane interlocks in lieu of requiring a plant shutdown or stopping refueling operations does not have a detrimental impact on the integrity of any plant structure, system, or component. Allowing the plant to remain in its existing MODE without requiring a plant shutdown, no longer restricting MODE changes with inoperable equipment, and increasing the surveillance interval of the reactor vessel head vents also does not have a detrimental impact on the integrity of any required plant structure, system, or component. Changing the Applicability requirements for CVCS and refueling manipulator crane interlocks since necessary requirements already exist will not alter the operation of any plant equipment, or otherwise increase its failure probability. The probability that equipment failures resulting in an analyzed event will occur is unrelated to the proposed changes since they do not result in any assumed accident or transient. As such, the probability of occurrence for a previously analyzed accident is not increased.

The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, and the availability and successful functioning of the equipment assumed to operate. The CVCS in MODES 5 and 6, reactor vessel head vents, and the refueling crane interlocks are not assumed to mitigate any analyzed event. Extending the time for which systems may continue to remain inoperable or allowing MODE changes with inoperable equipment also does not result in an increased consequence of an analyzed event. The proposed compensatory actions for inoperable Circulating Water Flood Protection Instrumentation ensure that no flooding analysis assumptions are violated since appropriate actions will be taken in the event of a circulating water system leak or failure. Based on this evaluation, there is no increase in the consequences of a previously analyzed event.

The proposed changes do not involve a physical alteration of the plant. No new equipment is being introduced, and no installed equipment is being operated in a new or different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints which initiate protective or mitigative actions. No change is being proposed to the procedures governing normal plant operation or those procedures relied upon to mitigate a design basis event. The changes do not have a detrimental impact on the manner in which plant equipment operates or responds to an actuation signal. As such, no new failure modes are being introduced. In addition, the change does not alter assumptions made in the safety analysis and licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The margin of safety is determined by the design and qualification of the plant equipment, the operation of the plant within analyzed limits, and the point at which protective or mitigative actions are initiated. The proposed changes do not significantly impact these factors. There are no design changes or equipment performance parameter changes associated with this change. No setpoints are affected, and no change is being proposed in the plant operational limits as a result of this change. Therefore, this change does not involve a reduction in the margin of safety.

SEV-1063
USE OF CONTAINMENT EQUIPMENT HATCH ENCLOSURE BUILDING

The purpose of this safety evaluation (SEV) is to evaluate the use of the containment equipment hatch enclosure building under the following conditions:

a. Meeting LCO 3.9.4 Required Action A.4 and LCO 3.9.5 Required Action B.3;

b. Meeting NUMARC 91-06 guidelines with respect to containment closure, except during reduced inventory conditions (i.e., mid-loop operations).

The use of the equipment hatch enclosure building to meet the requirements of LCO 3.9.3 is being addressed separately by a proposed technical specification amendment. Therefore, this SEV will not address LCO 3.9.3.

It should be noted that this safety evaluation is based on the assumption of the completion of the modification to the Containment Equipment Hatch Enclosure Building. While the modification does not require a safety evaluation in order to be implemented, actual use of the equipment hatch enclosure building in place of other currently accepted means for the above conditions must be evaluated since the UFSAR does not address this use.

Operation of Ginna Station in accordance with the proposed changes does not involve an increase in the probability or consequences of an accident previously evaluated. Containment is only used to mitigate the consequences of an accident, and as such, the use of the equipment hatch enclosure building will not increase the probability of an accident. Since containment must remain operable for all previously analyzed accidents per LCO 3.6.1 and 3.9.3, the use of the equipment hatch enclosure building during other periods will not result in the increase in the consequences of an accident.

The proposed use of the equipment hatch enclosure building will follow a plant modification per PCR 96-019. However, this modification will not add any new equipment and no installed equipment is being operated in a new or different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints which initiate protective or mitigative actions. No change is being proposed to the procedures governing normal plant operation or those procedures relied upon to mitigate a design basis event. The changes do not have a detrimental impact on the manner in which plant equipment operates or responds to an analyzed event. As such, no new failure modes are being introduced. In addition, the change does not alter assumptions made in the safety analysis and licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The margin of safety is determined by the design and qualification of the plant equipment, the operation of the plant within analyzed limits, and the point at which protective or mitigative actions are initiated. The proposed uses of the equipment hatch enclosure building do not impact these factors. There are no equipment performance parameter changes associated with this change. No setpoints are affected, and no change is being proposed in the plant operational limits as a result of this change. Therefore, this change does not involve a reduction in the margin of safety.

SEV-1064
MOV 4616 50% OPEN THROTTLING

The valve portion of MOV 4616 had been replaced in 1994 with a valve having a longer stroke time than the original (114.8 seconds vs. 30 seconds). The longer stroke time is now, by analysis, determined to be undesirable and the stroke time is to be reduced to the quickest practicable without access to the valve due to service water unavailability. This is to be accomplished by adjusting the open limit switch to limit the opening stroke to approximately 50% open.

The probability of occurrence or the consequences of an accident or the malfunction of equipment important to safety previously evaluated in the SAR is not increased since the resultant reduction of closure time of MOV 4616 better ensures that the affected loads will receive their required service water flows.

The possibility for an accident or malfunction of a different type than any previously evaluated in the SAR is not increased since the increased pressure drop across MOV 4616 is negligible and will not impact service water flow to the component cooling water and spent fuel pool heat exchangers. Additionally, since the increased pressure drop across MOV 4616 is negligible and due to the flow characteristics of the twenty-inch gate valve, the valve portion of MOV 4616 will not be subjected to increased flow induced degradation.

The margin of safety as defined in the basis for any technical specification is not reduced since the ability of MOV 4616 to perform its safety function to isolate service water flow upon SI actuation and to provide service water flow to its design loads when SI is not actuated will not be adversely impacted by throttling MOV 4616 to 50% open.

10CFR50.59 SAFETY REVIEW FOR UFSAR CHANGE NOTICE UCN 13/051 CHANGE THE CALIBRATION FREQUENCY OF THE METEOROLOGICAL TOWER INSTRUMENTS FROM SEMI-ANNUAL TO ANNUAL

No specific references to the calibration of instruments exist in Tech Specs as related to the Met Tower.

The UFSAR and regulatory documents were reviewed to assess the proposed change's significance. UFSAR sections 1.8.1.21, 1.8.1.23, 2.3.3, and 2.3.4.1.1 discuss the meteorological tower and the meteorological measurements program. The UFSAR does not directly specify calibration of the tower instruments, but section 2.3.3 does cite conformance with Reg. Guide 1.23 for system accuracies. Safety Guide 23 (originally applicable to Ginna), para. C.5 states that "the instruments should be calibrated at least semi-annually to ensure meeting system accuracies." A revision to Reg Guide 1.23 in April 1986 rewrote that section of the Guide to refer to ANSI/ANS-2.5-1984 (to consolidate the guidance into one standard) as acceptable in providing guidance in satisfying NRC regulations pertaining to the meteorological measurements program. It was recognized in 1986 that "Reg. Guide 1.23 was originally issued as Safety Guide 23 in Feb. 1972. Much of the information provided in the guide is now obsolete, having been made so by changes in the state of the art in meteorological measurement technology." Safety Guide 23 and Reg Guide 1.23 Sept. 80 were prescriptive and do not conform to the performance based approaches utilized today. ANSI/ANS-2.5-1984 Section 6.2 contains the semi-annual system calibration requirement and system accuracies which are consistent with Reg Guide 1.23.

Calibration frequency does not affect design of the system, does not change its function or method of performing its function, but could potentially impact degree of accuracy and confidence of parameters measured. A demonstrated set of performance data showing accuracy and consistency of reading can therefore be used as a basis to revise the existing semi-annual calibration frequency.

A review was performed of the regulatory documents associated with the meteorological measurements program, which was handled under SEP Topic III-2.B and later under TMI Item III.A.2 (NUREG 0737). Under SEP Topic III-2.B in a letter from D.K. Davis, SEP Branch, Division of Operating Reactors to D.L. Ziemann, Operating Reactors Branch #2, dated May 29, 1979, it was concluded that the measurements system met Reg Guide 1.23 for system accuracies.

Under TMI Item III.A.2, in a letter from RG&E to NRC, subject: Meteorological Assessment Capability in Response to NUREG-0737, dated April 28, 1981, RG&E committed to performing semi-annual primary and back-up weather tower calibrations, in response to NRC suggested quarterly calibrations. Other correspondence of significance includes RGE letters dated July 1, 1981, (NUREG 0737)

June 11, 1982 (GL 82-10), and Information Notice 84-91, Quality Problems of Meteorological Measurement Programs, dated Dec. 10, 1984. Therefore, an annual calibration frequency constitutes a change to RG&E's commitment. Applying the guidance of RG&E administrative procedure IP-SEV-1, this commitment may be changed based on the following:

1) The change is not significant to safety.

2) The original commitment is not necessary to meet an "obligation", i.e., 10 CFR, or Notice of Violation. The semi-annual frequency was based on Safety Guide 23 and Reg Guide 1.23 Rev. 1 standards, which were largely based on obsolete equipment and meteorological methods of measurement. Rev. 2 of Reg Guide 1.23 and Reg. Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents," Section C.11.c Calibration (1974), gives guidance which is not prescriptive but is performance based. The calibration data examined under the subject UCN demonstrate that the as-found values are within tolerances. The data reviewed is typical since replacement of the met tower system under EWR 4531 in 1992. Primarily due to the current state of the art measurement equipment, there is a basis to extend the calibration interval to annually.

3) The NRC did rely upon the original commitment under their review of the onsite meteorological measurements program. NRC review was based on the Reg Guide 1.23 Rev. 1 standard that existed at that time. Clearly, conformance to tolerances for system parameters, which ensure the capability to measure and predict offsite doses in a radiological emergency, is more important than prescriptive calibrations that offer no benefit.

4) The original commitment was not made to minimize recurrence of an adverse condition. It simply defined calibration in accordance with standards at that time.

Calibration frequency is a measure to ensure that meteorological instrument accuracies are highly reliable for assessment of offsite dose calculations during emergency plant events. The frequency itself does not affect the design. Existing calibration data supports an increased (annual) calibration frequency. The equipment involved is not used in the mitigation of any accidents.

Existing data supports and justifies an annual calibration and therefore meets the originally intended equipment accuracies. The Reg. Guide 1.23 guidance is outdated relative to calibration frequency. The original intent is still satisfied. System accuracies are still required to be met per Reg. Guide 1.23 and ANSI/ANS-2.5-1984. Future annual calibrations that found the accuracy to exceed the existing limits would necessitate reverting back to semi-annual calibration.

Existing design requirements established as part of EWR Design Criteria for EWR 4531 continue to be applicable to ensure system accuracies and design standards. The existing acceptance criteria for the met tower instruments are still to be applied to the calibration, therefore, the design parameters will still operate within design limits as originally intended.

10CFR50.59 SAFETY REVIEW FOR CHANGE TO MATERIAL HANDLING EQUIPMENT IDENTIFICATION AND INSPECTION FREQUENCY

RG&E's response to the NRC on "Control of Heavy Loads," dated March 2, 1983, included the following commitments:

1. "All slings used at Ginna Station have load ratings marked on them."

2. Annual inspections of special lifting devices, "not to exceed 15 months."

Whenever possible, load ratings are imprinted on, or attached to slings. Due to size constraints or the danger of tags falling into critical equipment, individual identification may not be possible or prudent. Determining the capability of a piece of rigging equipment is the responsibility of the user, that is, the rigger. RG&E Nuclear Directive ND-MHE commits to establishing a program for training riggers and assigns them the responsibility to make appropriate determinations as to sling capacity prior to making a lift. This precludes the need to have load ratings marked on "all slings."

ND-MHE commits to the inspection of Special Lifting Devices "prior to first use during outages." The replacement of the steam generators will result in extended outage cycles exceeding the 15 month limit. The intent of the RG&E's commitment is met by performing inspections prior to use during any outage, regardless of the frequency or time elapsed.

The change affects only the inspection frequency for Special Lifting Devices. The equipment, its function and design remain unchanged.

This change allows for extended outage cycles while maintaining the same level of safety in the use of Special Lifting Devices.

The intent of the original commitment was to assure the inspection of Special Lifting Devices at the beginning of each outage, not to exceed 15 months. The change as found in ND-MHE assures that inspections occur prior to use during any outage, regardless of the frequency.